19. PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION

19.0 Background

The purpose of the U.S. Nuclear Regulatory Commission (NRC) staff's review of the probabilistic risk assessment (PRA) and severe accident evaluation is to ensure that GE-Hitachi Nuclear Energy Americas LLC (GEH) or the applicant has adequately addressed the Commission's objectives. The NRC derived these objectives from Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"; 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," dated April 2, 1993, 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 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 (ALWR) 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

8. SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," dated February 19, 1997, and the related SRM, dated June 30, 1997

The first four NRC policy statements provide 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, which was in effect at the time GEH submitted the ESBWR application for design certification (DC), required a DC application to include PRA and severe accident information in accordance with the following NRC regulations:

- 10 CFR 52.47(a)(8), which provides information with respect to compliance with a number of the technically relevant positions of the Three Mile Island (TMI) requirements in 10 CFR 50.34(f)

- 10 CFR 52.47(a)(21), which outlines 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 within 6 months before the docket date of the application and technically relevant to the design

- 10 CFR 52.47(a)(27), which describes a design-specific PRA

19.1 Probabilistic Risk Assessment

19.1.1 Introduction

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

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 Commission's objectives. These objectives include the following:

- Use the PRA to do the following:

  – Identify and address potential design features and plant operational vulnerabilities for instances in which 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 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 1×10^{-4} per year (yr) for core damage frequency (CDF) and less than 1×10^{-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 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, identify the systems, structures, and components (SSCs) included in RTNSS.

- Use the PRA in support of programs associated with plant operations (e.g., technical specifications (TS), 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, TS, and combined license (COL) action items and interface requirements.

[1] The reference to existing operating plants applies to the light-water reactor (LWR) plant technology that existed at the time the Commission issued its Severe Accident Policy Statement on August 8, 1985.

19.1.2 Quality of Probabilistic Risk Assessment

19.1.2.1.1 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. The levels correspond to the modeling of the three major phases of a severe accident, initiation to core damage (Level 1); core damage to containment failure and release (Level 2); and, assessment of radiological consequences (Level 3). The PRA also covers both internal and external events for at-power and shutdown operations.

The ESBWR Level 1 PRA uses a linked fault tree methodology. Fault trees have been developed and evaluated for the major ESBWR frontline and support systems to determine the probability that the 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: 1987–1995,” 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 do not directly apply 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 ESBWR design specifics 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.

Preinitiator and postinitiator 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., data regarding 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) advanced light-water reactor (ALWR) Utility Requirements Document (URD), Revision 4, 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), applies. 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) assesses 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 DHR.

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 external events analyses use the fault trees and event trees developed for the internal events evaluations to the maximum extent possible, employing 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 Probabilistic Risk Assessment

The applicant described the PRA maintenance and update program in the Design Control Document (DCD), Tier 2, Revision 6. This section summarizes the key elements of this program.

The applicant treated the ESBWR PRA model documentation as a controlled document containing the detailed information for the model. The applicant established the following set of requirements and design controls that COL applicants referencing the ESBWR 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 applicant 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.

The maintenance process requires an independent review of the model or model elements by a qualified reviewer or reviewers. 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 ensure that the PRA remains current with industry practices.

19.1.2.2 Regulatory Criteria

No specific regulatory requirements govern the quality of PRAs used to support DC. However, Regulatory Guide (RG) 1.174, Revision 1, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” issued November 2002; RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” issued January 2007; and Section 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” of NUREG-0800, “Standard Review Plan for the Review of Safety

Analysis Reports for Nuclear Power Plants (LWR Edition)," issued March 2007 (hereafter referred to as the SRP Revision 2), provide guidance on how to ensure quality in PRA applications for commercial nuclear power facilities. These documents articulate the fundamental objective 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 DC.

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 300 RAIs to the applicant during its review of Chapter 19 of DCD, Tier 2 and its 6 revisions, and NEDO-33201 and its 4 revisions which documents the ESBWR PRA (NEDO-33201 is hereafter referred to as the PRA report). The staff's initial review of these documents and subsequent review of the responses to the RAIs 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 all of these RAIs, and the staff has found the responses to be acceptable. The applicant has incorporated information provided in these RAI responses into Revision 4 of the PRA report and Revision 6 of DCD, Tier 2, as appropriate.

The staff considered PRA results in the DCD, as well as results of the applicant's 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, and addressed 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 PRA models consider T-H uncertainties.

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 that varied key T-H parameters for each of the passive systems to determine the effect on the criteria for a successful event sequence following a limiting initiating event. The studies addressed a number of passive systems,

including 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 applicant performed these studies with the Modular Accident Analysis Program (MAAP 4.0.6) 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 <2,200 °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 <2,200 °F	No. of valves valve size MAAP 4.0.6 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 MAAP 4.0.6 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 4.0.6 code by comparing simulations of LOCAs performed with MAAP 4.0.6 and with those using the GEH version of the Transient Reactor Analysis Code (i.e., the TRACG code). However, these benchmark calculations may not reflect T-H conditions in the reactor vessel during such accidents. The applicant applied the design-basis accident (DBA) analysis assumptions (i.e., the single-failure criterion) regarding availability of passive mitigating systems rather than the assumptions made for the PRA, which are substantially more limiting. In RAI 19.1.0-1, Supplement 1, the staff 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 4.0.6 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 requested that the applicant provide the rationale for the accident scenarios selected, including any criteria applied in making the selections and the results of any parametric studies used to identify limiting scenarios.

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[2] The applicant based these results on the sensitivity study.

[3] The applicant did not perform sensitivity analysis because it used the design-basis criteria assumed in the PRA.

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. To understand the uncertainty in the determination of minimal success criteria, the staff requested, in RAI 19.1.0-1, Supplement 1, that the applicant identify the key parameters and describe how the analysis treated each one (e.g., as nominal values or bounding values) and, in cases in which nominal parameter values were used, discuss the impact on the results of the analyses if bounding parameter values had been used.

In the analyses, the applicant applied a limit of 2,200 degrees Fahrenheit (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. The staff tracked RAI 19.1.0-1 as an open item in the safety evaluation review (SER) with open items.

The applicant provided its response to RAI 19.1.0-1, Supplement 1, in a letter dated June 26, 2008. In this response, the applicant compared the performance of the TRACG and MAAP 4.0.6 codes for simulating medium- and large-break LOCA events in which the core becomes uncovered and heats up substantially before emergency cooling is started. Such conditions represent a challenge to successful mitigation of severe accidents. The results of the analyses show that the two codes predict similar behavior of key T-H parameters during the LOCA transients. The applicant provided adequate explanations for the few notable differences between the simulation results. These results adequately address the staff's concern with the original benchmark calculations.

In its response to RAI 19.1.0-1, Supplement 1, the applicant also provided an adequate rationale for its selection of limiting scenarios, including a discussion of the criteria used for selection. The applicant also identified the key T-H parameters and described how the analysis treated each one and why it was treated in that way. In cases in which nominal parameter values were used, the applicant adequately discussed the impact on the results of the analyses, if bounding parameter values were to have been used.

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. In its response, the applicant identified the key T-H phenomena, physical processes, and core parameters which directly determine the core heatup process and peak cladding temperature and provided references to topical reports which describe how TRACG models these phenomena and processes and the qualification of these aspects of TRACG using a wide range of test data. The staff finds this to be adequate justification for the use of TRACG in the study of PRA success criteria. Therefore, RAI 19.1.0-1 and the associated open item are resolved.

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 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. This study indicated 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 it used the MGL method 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. The staff considers the use of this method in the context of the general approach for treating CCFs, as described in NUREG/CR-4780, to be acceptable. The staff also finds the referenced methods for estimating CCF parameters to be acceptable.

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 which the NRC has endorsed (e.g., American Society of Mechanical Engineers (ASME)-RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications"). The applicant stated that, "where applicable, ASME-RA-Sb-2005 Capability Category 2 (CC-II) attributes are included in the analysis." In RAI 19.1-117, the staff requested that the applicant: (1) identify those high-level requirements or CC-II attributes of the standard that the ESBWR PRA did not embody, (2) address the impact on the qualitative and quantitative results of the PRA of excluding those high-level requirements or CC-II 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. The staff tracked RAI 19.1-117 as an open item in the SER with open items.

The applicant provided its response to RAI 19.1-117 in a letter dated January 16, 2008. In its response, the applicant presented the results of its assessment which showed the extent to which the ESBWR PRA incorporated CC-II attributes of ASME-RA-Sb-2005. These results included a list of the ASME standard's supporting requirements (SR), which are not considered to be applicable to the ESBWR design PRA; adequately explained why each item was not applicable; and discussed the capability level satisfied by requirements considered to be applicable to the ESBWR PRA. The applicant identified two SR which did not satisfy CC-II. The SRs considered not applicable to the ESBWR design PRA included those that pertained to treating plant operational programs that are not defined at the design stage and those that are not consistent with unique objectives of a design PRA. For each of the two SRs that did not

incorporate CC-II attributes, the applicant evaluated the impact of this condition on the qualitative and quantitative results of the PRA and discussed the results of the evaluation in the response. The staff has reviewed the information provided by the applicant and finds it to be adequate to address the concern reflected in RAI 19.1-117. Therefore, RAI 19.1-117 and the associated open item are resolved.

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. These elements include the following:

- Monitor PRA inputs and collect new information
- Ensure cumulative impact of pending plant changes are considered
- Maintain configuration control of the computer codes used in the PRA
- Identify when PRA needs to be updated based on new information or new models/techniques/tools
- Ensure peer review is performed on PRA upgrades

The staff has reviewed the applicant's proposed program and determined that the program includes the key elements described in RG 1.200. The staff finds the program described by the applicant in DCD, Tier 2, Revision 6, acceptable.

19.1.2.4 Conclusion

Based on its review of the information provided by the applicant, the staff finds that the quality of the applicant's PRA is sufficient for the PRA to be used to address the Commission's objectives, described in Section 19.1.1 of this report, that govern the treatment of severe accidents for DC. In addition, the staff finds that the applicant's PRA maintenance and update program includes the key elements described in RG 1.200 and is therefore acceptable.

19.1.3 Special Design Features

19.1.3.1 Summary of Technical Information

19.1.3.1.1 Design and Operational Features for Preventing Core Damage

Revision 2 of the applicant's PRA report and appropriate sections of the ESBWR DCD, Tier 2, Revision 6, describe the design and operational features of the ESBWR aimed at preventing core damage. These features include the following:

- For prevention and mitigation of an 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 the SLCS under conditions indicative of an ATWS

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

DCD, Tier 2, Revision 2, 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 to provide isolation in a passive way 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 when the condensate return valves open to initiate the system.

- The GDCS provides passive emergency core cooling after any event that threatens the reactor coolant inventory. Once the ADS has depressurized the nuclear boiler system (NBS), 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 (SDC) water. The FAPCS can provide cooling water during the long term using a pipe connection to convey water to the IC/PCCS pool for post-LOCA heat removal after 72 hours.

- During a total loss of offsite power, the onsite, non-safety-related diesel generators automatically power the safety-related electrical distribution system. 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 CHR system that maintains the containment within its design pressure and design temperature limits for DBAs, including LOCAs and postblowdown 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 and Operational Features for Mitigating the Consequences of Core Damage and Preventing Releases from Containment

Revision 4 of the applicant's PRA report and appropriate sections of DCD, Tier 2, Revision 6, 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 withstand a higher ultimate pressure than used for currently operating BWRs. The 95-percent confidence structural capacity (fragility) of the ESBWR primary containment system to overpressurization for the 260 degree Celsius (C) (500 degree F) steady-state thermal condition is 1.095 megapascals (MPa) (gauge) (159 pounds per square inch gauge (psig)) limited by leakage at the drywell head flange as the result of bolt yielding. Under normal operating (ambient) conditions, the structural pressure capacity 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 degree C (1,000 degree 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 are: the wetwell, including the suppression pool; an upper drywell (UDW) region surrounding the RPV; and a lower drywell (LDW) region below the RPV. Vacuum breakers are located 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 smaller than 1 square centimeter (cm^2) (0.155 square inch ($in.^2$)).

  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 deinerting value of 5 percent is significantly greater than 24 hours, which allows ample time for implementation of 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 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 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 LDW, 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 indicate 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 (BiMAC) device 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 ensures long-term coolability and stabilization of the resulting debris.

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

Revision 4 of the applicant's PRA report and appropriate sections of DCD, Tier 2, Revision 6, 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 device to flood the LDW when the temperature in the LDW increases enough to indicate 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 the use of the PRA in the design process and discuss 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 Section 18 of the PRA report, Revision 4, which DCD, Tier 2, Revision 6, references.

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 as 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 RCPB pressure.

- The probability of a loss of CHR is significantly reduced because the redundant heat exchangers and completely passive component design of the PCCS make it highly reliable.

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:

- The design includes additional redundant, physically separated flowpaths to the low-pressure injection and suppression pool cooling lines in response to fire analysis.

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

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

- The design includes additional redundant supply valves for ICS and PCCS pool makeup.

- The design includes additional redundant drainline valves for the ICS to eliminate a dependency on power supplies.

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

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

- The design includes the BiMAC device to reduce the consequences of severe accidents.

19.1.3.2 Regulatory Criteria

The staff has considered the special design features of the ESBWR design with respect to the Commission's objectives for new reactor designs, as stated in Section 19.1.1 of this report. 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.

No specific regulatory requirements govern the special design features used to support DC.

However, the staff has used applicable guidance from SRP Section 19.0, Revision 2 in its review.

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, and core uncovering during LOCAs and ISLOCAs, as well as fires and floods. All of these events have contributed significantly to risk in current operating plants and required design and operational changes after the facilities were built and operating. In addition, the ESBWR design includes features that address the following specific containment failure modes:

- DPVs and structural improvements to the containment to address DCH from high-pressure melt ejection (HPME)
- GDCS deluge and the BiMAC device to address potential melt-through of the containment
- fewer containment penetrations to reduce the likelihood of containment bypass

The staff finds that the applicant has provided an adequate balance between design features that prevent accidents and those that 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 previously, indicates that the applicant has included several features in the ESBWR design to address the major contributors to core damage in the current generation of BWRs (i.e., SBO, ATWS, and LOCA). In addition, the ESBWR design includes features to address specific containment failure modes. Therefore, RAI 19.1-73 is resolved.

19.1.3.4 Conclusion

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 currently operating BWRs. 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 at-power operations 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 listed below in response to RAI 19.1-68 and incorporated this information into Revision 4 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

The applicant's response provided adequate detail to resolve the staff's concern. Therefore, RAI 19.1-68 is resolved.

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 1.68×10^{-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 key backup non-safety systems, the focused Level 1 PRA results are reduced by almost two orders of magnitude. However, the impact to the CDF can be minimized by about one order of magnitude if the analysis credits only the availability of the DPS (including surrogate logic for the DPS signal for isolation of the main steam isolation valve (MSIV)).

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

Section 18 of Revision 4 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 DCD, Tier 2, Revision 6, and Section 7 of the PRA report, Revision 4, describe the significant accident sequences leading to core damage. The 10 most significant sequences, which constitute approximately 65 percent of the CDF, are summarized below:

- General Transient with ATWS (approximately 13 percent of CDF)

  - Scram fails
  - SLCS fails

- Inadvertent Opening of a Relief Valve (approximately 10 percent of CDF) where the following actions occur:

  - Scram is successful
  - High-pressure injection fails
  - Depressurization is successful
  - Low-pressure injection fails

- Inadvertent Opening of a Relief Valve (approximately 10 percent of CDF) where the following actions occur:

  - Scram is successful
  - High-pressure injection fails
  - Depressurization fails

- Medium Liquid LOCA (approximately 5 percent of CDF)

  - Scram is successful
  - Vacuum breakers pressure suppression is successful
  - Depressurization is successful
  - Low-pressure injection fails

- General Transient with ATWS (approximately 5 percent of CDF)

  - Scram fails
  - One or more SRVs sticks open
  - Maintenance of RPV water level fails

- Medium Liquid LOCA (approximately 5 percent of CDF)

  - Scram is successful
  - Vacuum breakers pressure suppression is successful
  - Depressurization is successful
  - Low-pressure injection fails

- Medium Liquid LOCA (approximately 5 percent of CDF)

  - Scram is successful
  - Vacuum breakers pressure suppression is successful
  - Depressurization fails

- Small Steam LOCA (approximately 4 percent of CDF)

  - Scram is successful
  - Vacuum breakers pressure suppression is successful
  - Depressurization is successful
  - Low-pressure injection fails

- CRD injection fails

- Small Liquid LOCA (approximately 4 percent of CDF)
  - Scram is successful
  - ICs are successful
  - Depressurization is successful
  - Vacuum breakers pressure suppression is successful
  - Low-pressure injection fails
  - CRD injection fails

- LOPP (approximately 4 percent of CDF)
  - Scram is successful
  - ICs fail
  - SRV open and reclosure is successful
  - Depressurization fails
  - CRD injection fails

19.1.4.1.1.2 Leading Initiating Event Contributors to Core Damage from the Level 1 Internal Events Probabilistic Risk Assessment. Transients contribute the most to CDF (approximately 59 percent). The most significant groups of transient initiators are the following:

- inadvertent stuck-open relief valve (22 percent)
- general transients (19 percent)
- loss of offsite power transients (10 percent)
- loss of feedwater transients or instrument air (5 percent)

LOCAs that occur inside containment contribute approximately 39 percent to the CDF. The most significant LOCA initiators with respect to CDF contribution are the medium liquid LOCA, small steam LOCA, and small liquid LOCA, which, together, represent 35 percent of the overall CDF, thus becoming the third, fourth, and sixth most important initiating events, respectively. The large contribution of these LOCA events is caused primarily by feedwater isolation, which occurs by design in scenarios in which high drywell pressure exists, and CRD isolation, which occurs by design in scenarios in which high drywell pressure and high LDW level exist. Finally, breaks outside containment (BOCs) represent less than 2 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 65 percent of the CDF. Core damage events occurring at high RPV pressures with the containment initially intact account for approximately 18 percent of the CDF. Core damage events that involve a failure to insert negative reactivity account for about 16 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 1 percent of the CDF.

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 the PRA results to individual system failures. Based on this study, the applicant identified the following systems as the most important from a risk perspective:

- ADS
- ICS
- CRDS
- SLCS
- safety-related and non-safety-related I&C 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, failure to open the vent in the ICS when required, preinitiator valve positioning errors in the CRDS, and failure to recognize the need to makeup the ICS and PCCS pool levels. The human factors engineering program incorporates information on important operator actions.

Section 19.1.3 of this report 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 (1) develop a better understanding and provide insights related to CDF generated through model analysis and (2) 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	All actions fail; all actions succeed	< 100-fold change
Common-Cause Failure	All CCF eliminated	1,000-fold decrease
Squib Valve Failure Rates	Failure rates increased by factor of 2 in key systems	Substantial increase
T&M Unavailability	All T&M activities fail; 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	Six of 12 versus four 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 Postcontainment-Failure	CRD assumed to fail if containment fails	No change
Accumulators	All accumulators supporting pneumatic components fail	100-fold increase

[4] The applicant provided this assessment.

Vacuum Breakers	Failure rate increased by factor of 10	Slight increase
System Importance	Importance measures computed for 40 systems	20 systems with FV > 0.01 (risk significant)
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 as follows:

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

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

- Changes to squib valve failure data, particularly when used for the ADS and GDCS functions, have a significant impact because of their contribution to passive safety features.

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, as stated in Section 19.1.1 of this report. The following four objectives for the applicant's use of the design PRA are most 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 1×10^{-4}/yr for CDF.

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

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used applicable guidance from SRP Section 19.0, Revision 2 to conduct its review.

19.1.4.1.3 Staff Evaluation

The applicant has reported a CDF of 1.68×10^{-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, unlike current operating plants, injection into the reactor vessel with a diesel-driven fire pump is part of the ESBWR design.
Transients with Loss of DHR Capability	• degree of dependency of DHR systems on support systems; low dependency reduces CDF • the capability of the ECCSs to pump saturated water; reduces CDF • use of the RWCU as an alternative DHR system; reduces CDF • ability to replenish water sources	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.

Event Category	Design Features in Existing BWRs That Significantly Affect CDF (NUREG-1560)	Relevant ESBWR Design Features
	outside containment for use in long-term cooling	
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 to specifically address issues important to risk in previous BWR designs. It is reasonable to expect, based on these changes, that the CDF for the ESBWR 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 of this report documents the staff's evaluation of passive system success criteria.

19.1.4.1.4 Conclusion

The staff has reviewed the results and insights derived from the Level 1 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 (i.e., 1.68×10^{-8}/yr) 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 1×10^{-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 and 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 (1.4×10^{-9} per reactor-year for at-power internal events and 4.7×10^{-9} per reactor-year for all at-power events, respectively). Accident sequences leading to an LRF are unlikely but have broad bands of uncertainties 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 for internal events approaching 0.08, and an overall CCFP for all at-power events of about 0.11.

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 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 applicant evaluated the containment response 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 input into the CET evaluation. The results of the CET evaluation are then grouped into a set of "release categories" for use as source terms for the offsite consequence analysis and, subsequently, risk integration.

The applicant created a Level 2 PRA quantification model 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). Each of the Level 2 CETs models the nodes 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 PRA as a single, one-time quantification model allows the results to correctly reflect all dependencies and initiator impacts.

19.1.4.2.1.1.1 Containment Event Trees. To determine the conditional system failure probabilities values used on the CET branches, the 138 listed Level 1 quantified accident sequences above the cutoff level of 1.0×10^{-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 (less than 1 MPa) (65 percent of CDF).
- Class II: Containment failure precedes core damage (0.2 percent of CDF).
- Class III: Vessel failure occurs at high pressure (greater than 1 MPa) (18 percent of CDF).
- Class IV: Vessel failure occurs at low pressure; core damage results from failure to insert negative reactivity in ATWS conditions (16 percent of CDF).
- Class V: Core damage occurs with the RPV open to the environment because of BOCs (0.5 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 assumed 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[5] If the water level is below 0.7 meters, the applicant determined that a steam explosion impulse would not challenge the containment structure.

[5] "Behavior is physically unreasonable and violates well-known reality. Its occurrence can be argued against positively." (Theofanous and Yang, 1993)

The applicant used the CETs 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 analysis used the Level 1 sequence bins 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. The source term evaluation used release categories, which represent meaningfully different outcomes to the containment challenge.

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	1.1×10^{-8}	Sequences with RPV failure at low pressure	Low/Dry	I_LD	8.0×10^{-9}	0.48
			Medium	I_M	1.8×10^{-9}	0.11
			High	I_H	1.1×10^{-9}	0.07
Class II	3.1×10^{-11}	Containment failure preceding core damage	No CET required as the containment is failed in these sequences before core damage			
Class III	3.0×10^{-9}	Sequences with RPV failure at high pressure	Low/Dry	III_LD	3.0×10^{-9}	0.18
Class IV	2.7×10^{-9}	Sequences involving failure to insert negative reactivity	Low/Dry	IV_LD	2.7×10^{-9}	0.16
			Medium	IV_M	3.9×10^{-12}	< 0.01
			High	IV_H	1.1×10^{-11}	< 0.01
Class V	7.9×10^{-11}	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 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 CCI.

The applicant conducted a complete Level 2 fault tree analysis 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 represents each of the accident classes for detailed modeling. This allows evaluation of the containment response to the complete spectrum of accidents contributing to the CDF.

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:

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 74 percent of the Class I frequency is associated with stuck-open relief valve (T-IORV), large feedwater LOCA (LL-S-FDWA/B), or small and medium LOCA (SL-, ML-) 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. The applicant used a similar approach 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 break. 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. RPV 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 not initially depressurized (ADS inhibit successful). SLCS ineffective or unavailable. Feedwater runback successful. No short- or long-term coolant injection. PCCS available, but no active CHR (FAPCS). GDCS/BiMAC function successful. RPV depressurization assumed to be successful before RPV failure.
V	None	No representative sequence assigned for containment evaluation because 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, the applicant developed an ESBWR simulation model using MAAP 4.0.6 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 are used as input to the Level 3 PRA calculations. Table 19.1-6 shows the results of MAAP 4.0.6 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 4.0.6 to estimate the probability of containment failure from DCH, EVE, or BMP 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 Core Damage (MPa)
T_nIN _TSL	618	0.50	0.8	7.8	7.8	0.58
T_nIN_nCHR_FR	614	0.49	0.9	7.7	7.7	0.91
MLi_nCHR	123	>72	>72	>72	NA	NA
T_nDP_nIN_TSL	NA	0.92	1.4	6.2	6.2	0.57
T_nDP_nIN_nCHR_FR	NA	0.93	1.5	6.7	6.7	1.01
T-AT_nIN_TSL	1,123	0.1	0.3	5.7	5.7	0.57
T-AT_nIN_nCHR_FR	1,124	0.1	0.3	5.8	5.8	1.04

Accident Class I involves sequences in which the RPV fails at low pressure, and Accident Class III involves sequences in which the RPV fails at high pressure. Accident Class IV includes sequences that are initiated by an ATWS and followed by failure to achieve subcriticality. Transient sequences in which there is no core injection dominate all three classes. The analysis used sequences T_nIN, T_nDP_nIN, and T-AT_nIN 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 boiloff 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 in which no short- or long-term coolant injection to the RPV by the feedwater system, CRDS, FPCS, 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 618 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 follow. 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, preventing significant CCI. 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 because 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 strength. 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 33 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 5.3 hours. Further shroud level decrease occurs until flow through the equalizing line begins at about 6.2 hours. Flow from the suppression pool maintains the RPV level above the top of active fuel (TAF) beyond 72 hours.

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 during which mitigating actions can be implemented 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)

An LOPP is the initiating event for the sequence T_nDP_nIN. 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 SRV setpoint.

Actuation of the GDCS deluge line and successful BiMAC function prevent significant 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 in which CHR has failed, the containment pressure increases, and controlled venting is implemented to limit the pressure rise and control the radiological releases. The drywell pressure reaches 0.62 MPa (about 81 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 approximately 28 hours after accident initiation, which is about 4 hours before containment failure caused by overpressurization would be expected.

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.

Actuation of the GDCS deluge lines and successful BiMAC function prevent significant CCI from occurring in the LDW (CCI is limited to the protective layer of concrete on top of BiMAC). The dispersed core debris to the UDW regions would not result in significant CCI because of the large cooling potential of the core debris when dispersed over a large area. Continued heating of water by the core debris in the LDW results in protracted generation of steam and containment pressurization. The PCCS condenses the debris-generated steam from the containment, thus preventing containment failure by 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. Radiological releases to the environment occur only through potential containment leakage at the TS limits because 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 one 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 TSL 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 occurs outside of containment.	Feedwater, main steam, RWCU/SDC line breaks	RPV to environment
BYP	Loss of isolation occurs.	CIS function failure	Drywell to environment
CCID	LDW corium debris not flooded; CCI noncondensable gas ruptures drywell.	Unsuccessful GDCS deluge	Drywell to environment
CCIW	LDW corium debris bed flooded but not effectively cooled; CCI gas ruptures drywell.	Unsuccessful GDCS deluge	Drywell to environment
DCH	DCH event (RPV failure at high pressure) overpressure ruptures drywell or fails liner.	physically unreasonable; no failure 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	Leakage allowed 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 1.4×10^{-9} per reactor-year. The Level 1 CDF is 1.68×10^{-8} per reactor-year, so that the ESBWR CCFP for all non-TSL failure modes is 0.08 (the ratio of these two numbers), which is consistent with the NRC's containment performance objective of 0.10.

19.1.4.2.1.1.5 Source Term Evaluation. The applicant performed the source term evaluation using the MAAP 4.0.6 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 4.0.6 code.

The analysis typically incorporated conservative assumptions 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 values represent the expected ESBWR operating conditions.

In Section 9 of Revision 4 of the PRA report, GEH, in response to RAI 19.1-177, revised the source terms for release categories CCID, CCIW, FR, OPVB, OPW1, and OPW2 to account for the reduction in the containment ultimate capacity. The applicant provided two sets of release fractions: 24-hour and 72-hour release fractions. Consistent with the previous revisions, the release fraction at 24 hours represented early release source terms, and the release fraction at 72 hours represented release variations and uncertainties at least 24 hours up to 72 hours after the event.

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 (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 BMP in the ROAAM evaluation. Section 9 of the PRA discusses containment overpressure failure as a consequence of system failures. The following sections briefly explain these failure modes, as pertinent to the ESBWR.

19.1.4.2.1.2.1 Containment Failure from Direct Containment Heating. 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 deentrainment occurs, is in the UDW.

The ROAAM analysis demonstrated that the ESBWR containment can withstand bounding DCH pressure loads and concluded that catastrophic containment failure as the result of DCH is physically unreasonable.

The applicant stated 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, the PRA does not identify DCH as a containment rupture failure mode.

However, the calculations also show short periods of potentially very high temperatures in the LDW atmosphere (up to 4,000 kelvin (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. The staff considered the design of the lips and concludes that the applicant's assumption is reasonable.

19.1.4.2.1.2.2 Containment Failure and BiMAC Failure Resulting from Ex-Vessel Steam Explosions. 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 for low-pressure scenarios, are especially prone to 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 (413 pounds per square inch (psi)), explosive-level pressures acting on a time scale of milliseconds can produce concrete cracking, along with liner stretching and tearing, sufficient to compromise the leaktightness 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, BMP, and containment pressurization by the so-generated noncondensable gases.

The ROAAM assessment in Chapter 21 of the PRA finds that failure of the ESBWR containment liner (and therefore, the leaktightness of the containment) because of EVE is physically unreasonble 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 performed by the applicant 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 (i.e., accidents involving deep, subcooled water pools), violation of the ESBWR containment leaktightness and the BiMAC function as the result of EVE is physically unreasonable. 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

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 demonstrates that, if the water level is "medium" or "low/dry," a sufficiently energetic steam explosion is physically unreasonable. PRA sensitivity studies assign a failure probability of 1.0×10^{-3} to cases involving a medium water level in the LDW.

19.1.4.2.1.2.3 Containment Failure from Molten Core-Concrete Interactions. 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. Neither significant ablation of concrete in the basemat or pedestal wall nor containment overpressurization by concrete decomposition gases would occur. The applicant stated that the following features support this conclusion:

- A layer of concrete will serve as a protective layer to eliminate impingement attack by superheated metallic jets.

- 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 dryouts 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 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 2.7×10^{-4} (citing historical data) for failure of debris cooling due to BiMAC line plugging and failure of GDCS flow following successful deluge operation based on the design of the BiMAC device.

Accident sequences that successfully supply water to the BiMAC, but with the BiMAC non-functional, 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.

Accident sequences in which no water is supplied to the BiMAC terminate with the end-state core-concrete interactions-dry (CCID). In such accident 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, GEH provided the results of sensitivity studies using MAAP 4.0.6, which it 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). The staff carried out a confirmatory assessment of CCI using MELCOR 1.8.6, confirming that concrete ablation depths in the axial direction would be of similar or somewhat smaller magnitude than those predicted by MAAP 4.0.6 for several comparable sequences involving assumed basaltic concrete under both dry and wet conditions. A representative MAAP 4.0.6 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 (RB) 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. The staff finds the applicant’s response to RAI 19.2-32 reasonable, so the issue is resolved.

19.1.4.2.1.2.4 Containment Isolation System Failure. In these events, the failure of the CIS causes the containment to be bypassed. 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. This is the condition in which 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. This is the condition in which 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. 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 limit the pressure rise to be below 90 percent of the containment ultimate pressure capacity.

19.1.4.2.1.2.8 Break Outside of Containment. 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 the possible failure of an ICS tube. The analysis of a BOC in the ICS, as an initiator, shows that the break makes a negligible contribution to the CDF. Therefore, RAI 19.2-38 is resolved.

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. 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 SDC 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. The TS limit for allowable containment leakage is 0.5 percent of containment air volume per day at rated design-basis pressure. Leakage at the TS limit, which is included in all of the modeled severe accident sequences, 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 RB 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 BOCs 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 deflagration due to presence of combustible gases cannot be excluded when the containment atmosphere is deinerted, GEH conservatively assumed that all core damage events during the deinerting period would lead to containment failure. The applicant calculated this contribution as 4.61×10^{-11} per reactor-year (i.e., CDF/365) and added it to the BYP frequency.

Table 19.1-8 Release Category Frequencies and Representative Release Fractions

Release Category	Frequency (per reactor-year) (% contribution to CF)	Representative CsI Release Fraction at 24 Hours
TSL (no CF)	1.54×10^{-8} (0)	0.00016
EVE	1.14×10^{-9} (82.3)	0.028
BOC	7.95×10^{-11} (5.7)	0.70
CCIW	2.92×10^{-12} (0.2)	0.00015
BYP	5.66×10^{-11} (4.1)	0.21
OPW1	1.97×10^{-12} (0.1)	0.0
OPVB	2.08×10^{-12} (0.1)	0.0033
CCID	1.47×10^{-12} (0.1)	0.068
FR	9.15×10^{-11} (6.6)	0.0
OPW2	8.51×10^{-12} (0.6)	0.0
DCH	0	-

Section 10 of Revision 4 of the PRA report provides release category frequencies for the external (internal fire, internal flood, and high winds) and shutdown events. For the external events during power operation, core damage sequences were assigned to various accident classes and release categories using an approach similar to that used for the internal events. For the external events during shutdown, the analyses conservatively assumed that the core damage scenarios result in large releases because the containment is open during most of the shutdown period.

The quantification resulted in a summed (all release categories except for TSL in Table 19.1-8) containment failure frequency of 1.4×10^{-9} per reactor-year. These are all termed “large releases.”

The low-pressure Class I accident sequences contribute the majority (86 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 2 percent of the containment failures, mostly as the release category FR.

The high-pressure Class III contributes about 0.4 percent of the containment failures, mostly as the release category BYP.

Class IV (ATWS type) contributes 5 percent to the CFP, mostly as the release categories BYP and EVE.

Class V contributes 6 percent to the CFP entirely as the release category BOC.

19.1.4.2.1.4 Risk-Significant Equipment/Functions/Design Features, Phenomena/Challenges, and Human Actions. The following paragraphs summarize important insights from the Level 2 PRA. These insights are organized in terms of equipment and design features, severe accident phenomena and 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 (CCI, 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 Commission's recommended goal of 10 percent for CCFP.

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 three 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 or heatup or both of the suppression pool water. Unlike the advanced boiling-water reactor (ABWR), or any other previous GEH BWR, the ESBWR containment design includes the PCCS to remove decay heat from the containment and the BiMAC device (also passive), which is intended to essentially eliminate the possibility of extended corium-melt interactions, noncondensable gas generation, and BMP.

Table 19.1-9 summarizes the containment challenges and mitigative attributes in place for the ESBWR. These attributes have contributed to reducing or eliminating the likelihood of the associated severe accident challenges 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 and CCI	BiMAC Activation Failure	Sensing and Actuation Instrumentation Diverse/Passive Valve Action
	Local Melt-Through	

19.1.4.2.1.4.2 Phenomena and Challenges. Given a severe accident, the applicant has considered the following challenges to containment integrity:

- prompt, energetic loading—explosive FCIs, HPME leading to DCH (and pressurization)
- late, gradual loading—melt ablation and penetration of the containment basemat, pressurization of containment atmosphere by steam or noncondensable gases or both
- isolation failure—errors or malfunctions that leave existing flowpaths open to the outside, activation of the containment overpressure protection system

Section 21 of the PRA report discusses the phenomenological (physics) components of these threats (namely, EVE, DCH, and BMP) as part of the ROAAM process. The discussion of BMP also provides the principal phenomenological input needed to assess containment overpressurization, which, because it is a systems-driven event, the Level 2 PRA treats. 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 (i.e., accidents involving deep, subcooled water pools) violation of the ESBWR containment leak-tightness and the BiMAC function as the result of EVE is physically unreasonable. The process also determined that the ESBWR containment can withstand bounding DCH pressure loads and that catastrophic containment failure as the result of DCH is physically unreasonable. 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 ensures 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 bounded by the ROAAM analysis because the quantities and rates of melt location from the RPV into the LDW would be significantly lower. In particular, this phenomenon

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, the containment evaluation considers operator actions 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, a long period (more than 24 hours) would be necessary 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. The staff agrees with this approach.

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), the applicant did not analyze source term uncertainties in terms of time, quantity, and chemical and physical forms of release. The ROAAM process does not cover the systems portions of the CETs, and it does not consider the propagation of the driving Level 1 PRA numbers. The Level 2 uncertainty analysis presented in Section 11 of the PRA demonstrates that the ratio of the upper bound of the LRF (95th percentile) to the mean value is approximately a factor of 3. Nevertheless, 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, as well as a large number of sensitivity studies, is such 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. Because the ROAMM approach is bounding in nature, the staff agrees that an importance analysis is not required..

19.1.4.2.1.5.3 Sensitivity Analysis. Tables 11.3-18, 11.3-8A, 11.3-19, and 11.3-19A of Section 11 of the PRA report, Revision 4, the results of four Level 2 sensitivity studies, which 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 Containment Isolation System Node Placement in the Containment Event Tree. In Revision 2 of the PRA report, the applicant describes a Level 2 PRA model sensitivity analysis used 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 the CIS in a nodal position of three or four.

Results for the CIS node sensitivity analysis showed no impact on LRF as demonstrated by a lack of change in nTSL frequency over the PRA Level 2 base model. Consequently, the placement of the CIS node earlier in the event trees has 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. The applicant performed a sensitivity study to better understand the impact to nTSL and source terms pertaining to the omission of these physically unreasonable modes from the model. These modes include EVE from a medium LDW water level and DCH.

Results for the physically unreasonable sensitivity analysis in Revision 2 of the PRA report 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} per reactor-year was obtained for the physically unreasonable 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 physically unreasonable sensitivity confirms that the exclusion of physically unreasonable events from the Level 2 PRA model is not negating any potentially significant offsite consequences.

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

Results for the vacuum breaker sensitivity showed an nTSL frequency of 2.06×10^{-9} per reactor-year at a truncation of 1×10^{-15} per reactor-year. This value for nTSL represents an increase in nTSL frequency of about 50 percent more than that of the base Level 2 model. However, the increased nTSL meets the NRC goal of 1×10^{-6} per reactor-year for LRF with considerable margin. The results show that the uncertainties associated with the primary vacuum breaker design and anticipated number of cycles increase the LRF only slightly.

19.1.4.2.1.5.3.4 Squib Valves. In the squib valves sensitivity analysis, the applicant increased the failure rates of the squib valves by a factor of 10 in the database file to account for uncertainty in general reliability and the mission time.

Results for the squib valve sensitivity showed an nTSL frequency of 1.18×10^{-8} per reactor-year at a truncation of 1×10^{-15} per reactor-year. This value for nTSL represents an increase in nTSL frequency of almost one order of magnitude compared to the base Level 2 PRA model. However, the increased nTSL meets the NRC goal of 1×10^{-6} per reactor-year for LRF with considerable margin. Based on these results, the uncertainties associated with the squib valve reliability may contribute to slightly increased LRF, but the increase is reasonable.

19.1.4.2.1.5.3.5 BiMAC Failure. Given failure of the BiMAC and continued corium-concrete interaction, there is a potential for RPV pedestal failure. The applicant performed sensitivity studies using MAAP 4.0.6 to estimate concrete ablation for both limestone and basaltic concrete. These cases involved a loss of injection with successful depressurization of the RPV. Section 19.1.4.2.1.2.3 of this report discusses the results.

The applicant does not consider it useful to perform LRF-based sensitivities for operator actions for two reasons. First, because the total CDF estimated in this sensitivity is less than 1.0×10^{-6} per reactor-year, it is not possible to raise the LRF value above the goal. Second, the LRF evaluation credits no important operator actions. 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, as stated in Section 19.1.1 of this report. 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 goal of less than 1×10^{-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 (b) 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

NUREG-1580 represents an extensive compilation of the results generated by the industry in performing its IPEs for the current generation of plants. 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, Tier 2, Revision 6, and Sections 8–11 and 21 of the PRA report 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 1.4×10^{-9} per reactor-year, and the corresponding CCFP is calculated to be 0.08 (0.11 when external events are included). The LRF is about three orders of magnitude below the Commission's safety goal, and the CCFP is acceptably low. This is a significant reduction in risk as compared to existing BWRs, which typically have LRF values in the range of 1.0×10^{-6} per reactor-year to 1.0×10^{-5} per reactor-year 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 for the following reasons.

  – The new features designed to prevent or mitigate ATWS greatly reduce the probability and 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.

  – A highly reliable ADS reduces the probability of a high-pressure core melt. 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 BMP 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.

The NRC carried out an independent assessment of the ESBWR design response to selected severe accident scenarios using the latest version of the MELCOR 1.8.6 computer code. The assessment examined 13 accident scenarios from the ESBWR PRA, which were chosen based on a combination of frequency, consequence, and dominant risk. The majority of these scenarios were similar or identical to sequences analyzed with MAAP 4.0.6 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 MAAP 4.0.6 results, particularly 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, the MELCOR 1.8.6 and MAAP 4.0.6 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 generally supports the results and conclusions of the source term analysis conducted in the ESBWR PRA.

19.1.4.2.4 Conclusion

The staff has reviewed the results and insights derived from the Level 2 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.

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

The applicant performed a Level 3 PRA to assess the calculated ESBWR public risk level results to three major offsite consequence-related goals. These goals were established in the GEH ESBWR licensing review, and are based on the NRC Safety Goal Policy Statement.

The intent of the following implemented design goals is to ensure that the radiological risk from accidents in the ESBWR is maintained 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 (3.9×10^{-4} fatalities per year).

GEH: As a design objective, the individual risk goal is set to be 3.9×10^{-7} fatalities per year within 1.6 kilometers (1 mile).

(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 (1.7×10^{-4} fatalities per year).

GEH: As a design objective, the societal risk goal is set to be 1.7×10^{-6} 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 (0.5 miles) from the reactor shall be less than 1.0×10^{-6} per reactor-year.

GEH: The design objective for the probability of receiving 0.25 Sv at 0.5 mile is set at less than 1.0×10^{-6}per reactor-year.

The staff agrees that these constitute a reasonable set of goals for establishing the level of public risk for the ESBWR, which are consistent with the NRC Safety Goal Policy Statement.

19.1.4.3.1 Level 3 Probabilistic Risk Assessment Methodology

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-mile and a 50-mile radius of the plant. The applicant used the MELCOR Accident Consequence Code System (MACCS2), Version 1.13, 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. The analysis estimated effective doses for each of 10 different release categories.

For the ESBWR Level 3 PRA, each of the 10 nonzero 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 were established based on plant-specific, T-H calculations using the MAAP 4.0.6 code.

Section 10 of Revision 4 of the PRA report lists the following input assumptions. The analysis used a meteorological condition comparable to the EPRI ALWR Utility Requirements Document (URD), Revision 4 meteorological reference data set. The Sandia siting study population density data were used to develop a uniform population density. A bounding uniform density of

305 people per square kilometer (km^2) (790 people per square mile) for the first 32 kilometers (20 miles) was 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 was assumed and no sheltering was assumed. The public was assumed to continue normal activity during the reactor accident in this bounding analysis.

The analysis modeled the following two baseline cases:

(1) The release category with 24-hour source terms was modeled to occur at ground level. The thermal content of the plume was assumed to be the same as ambient.

(2) The release category with 72-hour source terms was modeled to occur at elevated level. The thermal content of the plume was assumed to have a buoyant energy of one megawatt.

The staff reviewed these analyses and 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 is therefore acceptable.

19.1.4.3.1.1 Results. In Section 10 of Revision 4 of the PRA report, the applicant provided risk and consequence results in terms of the safety goals for external events and shutdown modes, in response to RAI 19.1-13, Supplement 1. Table 19.1-10 summarizes the baseline results for internal events and external events (i.e., internal fire, internal flood, and high winds) occurring during full-power operation and shutdown conditions and compares them to the evaluated NRC safety goals.

Table 19.1-10 Baseline Consequence Goals and Results (from Revision 4 of the PRA Report, Table 10.4-2)

Operating Status and Release Conditions		Risk Goals Criteria and Results			
		Individual Risk (0–1 Mile) < 3.9×10^{-7} (0.1%)	Societal Risk (0–10 Mile) < 1.7×10^{-6} (0.1%)	Radiation Dose Probability > 0.20 Sv (at 0.5 Mile) < 10^{-6}	Safety Goal Achieved
At-Power Internal	C 1	1.6×0^{-10}	2.0×10^{-11}	2.0×10^{-9}	Yes
	C 2	1.6×0^{-10}	2.6×10^{-11}	1.9×10^{-9}	Yes
Shutdown Internal	C 1	3.9×10^{-9}	1.4×10^{-9}	3.4×10^{-8}	Yes
	C 2	3.7×10^{-9}	1.6×10^{-9}	3.4×10^{-8}	Yes
At-Power Fire	C 1	2.9×10^{-10}	1.0×10^{-10}	3.0×10^{-9}	Yes
	C 2	2.8×10^{-10}	1.2×10^{-10}	3.1×10^{-9}	Yes
Shutdown Fire	C 1	2.2×10^{-9}	8.0×10^{-10}	1.9×10^{-8}	Yes
	C 2	2.1×10^{-9}	8.9×10^{-10}	1.9×10^{-8}	Yes
At-Power High Wind	C 1	2.3×10^{-10}	8.4×10^{-11}	2.3×10^{-9}	Yes
	C 2	2.4×10^{-10}	9.4×10^{-11}	2.5×10^{-9}	Yes
Shutdown High Wind	C 1	9.1×10^{-9}	3.3×10^{-9}	7.9×10^{-8}	Yes
	C 2	8.5×10^{-9}	3.7×10^{-9}	7.9×10^{-8}	Yes
At-Power Flood	C 1	6.7×10^{-10}	2.4×10^{-10}	5.9×10^{-9}	Yes
	C 2	7.1×10^{-10}	2.8×10^{-10}	7.0×10^{-9}	Yes

Shutdown Flood	C 1	1.2×10^{-9}	4.4×10^{-10}	1.0×10^{-8}	Yes
	C 2	1.1×10^{-9}	4.8×10^{-10}	1.0×10^{-8}	Yes
C1 = Base Case 1 (ground release); C2 = Base Case 2 (elevated release)					

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. Risk and consequence results in terms of the safety goals are not available for seismic events at power and shutdowns. 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 two 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 ESBWR design is protective of the public health and safety, 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 consistent with the Commission's safety goals for individual risk, societal risk, and radiation dose, as well as the Commission's CPG.

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 low 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 0.07 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 chosen to designate all containment failures as large releases (i.e., those in excess of technical specification leakage). The staff finds this conservative assumption acceptable.

Based on its review of Section 19 of the DCD, the staff concludes that 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 nonzero 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, respectively.

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. Similar insights are applicable to other events presented above.

The 72-hour values bound the reported 24-hour values but are not significantly greater. For example, the societal (latent fatality) risk is 2.6×10^{-11}/yr at 72 hours, compared with 2.0×10^{-11}/yr at 24 hours. The following are leading risk contributors:

- 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 noncontainment failure (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, respectively. These risk categories account for 77 percent, 6 percent, and 4 percent of the individual risk, and 61 percent, 5 percent, and 16 percent of the societal risk, respectively.

- Together, the release categories TSL, EVE, BOC, and BYP account for 99 percent of the CDF, 87 percent of the individual risk, and 83 percent of the societal risk.

Based on its review of Chapter 19 of the DCD, the staff concludes that 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 did not identify any risk-significant equipment, functions, design features, phenomena, challenges, and human actions as part of the Level 3 ESBWR PRA. Based on its review of Chapter 19 of the DCD, the staff concludes that this is acceptable.

19.1.4.3.5 Insights from Uncertainty, Importance, and Sensitivity Analyses

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

Throughout the various revisions of Section 10 of the PRA, the applicant presented sensitivity analyses of the offsite consequences, considering variations in meteorological conditions, release elevation, release energy (heat and buoyancy), and mission time.

The analysis considered 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 was considered to 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 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 times. The risk insights obtained via ground release modeling at 50 miles do not change even with elevated release modeling.

The sensitivity study showed 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.

19.1.4.3.5 Conclusion

The staff has reviewed the results and insights derived from the Level 3 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.

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

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

19.1.5.1 Results and Insights from the Seismic Risk Assessment

19.1.5.1.1 Summary of Technical Information

19.1.5.1.1.1 Methodology and Approach. 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 SSE (1.67*SSE). The analysis involves the following two major steps: (1) seismic fragilities and (2) accident sequence HCLPF analysis. The seismic fragilities of the ESBWR SSCs are based on generic industry information and ESBWR-specific seismic capacity calculations for certain structures. The MIN-MAX method is used to determine the functional and accident sequence fragilities. In accordance with 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, in accordance with 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 0.5g (acceleration due to gravity) 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 (PGA) (g level) at which there is a high confidence that the particular SSC will have a low probability of failure (i.e., 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 Significant Accident Sequences and Leading Contributors. 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.

19.1.5.1.1.3 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.

19.1.5.1.2 Regulatory Criteria

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

The staff has considered the results and insights from the SMA with respect to the Commission's objectives for new reactor designs, as stated in Section 19.1.1 of this report. The following objective is especially relevant to the evaluation of results and insights from the SMA: Identify risk-informed safety insights based on systematic evaluations of the risk associated with the design such that the applicant can identify and describe: (1) the design's robustness, levels of defense in depth, and tolerance of severe accidents initiated by either internal or external events, and (2) the risk-significance of specific human errors associated with the design.

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used the applicable guidance from SECY-93-087 and SRP Section 19.0, Revision 2 in its review.

19.1.5.1.3 Staff Evaluation

19.1.5.1.3.1 Methodology and Approach. The methodology used to perform the SMA follows a PRA-based approach as described in SECY-93-087 and associated SRM and is therefore acceptable.

The PRA-based SMA shows that the ESBWR design can meet the expected 0.84g HCLPF value if the seismic capacities of structures, systems and components (SSCs) associated with the seismic initiated accident sequences are qualified to be above the specified acceptable design value of 0.84g. In DCD, Tier 2, Revision 6, Section 19.2.6, the applicant stated the following:

> The COL Applicant will identify a milestone for completing a comparison of the as-built SSC HCLPFs to those assumed in the ESBWR SMA shown in Table 19.2-4. 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. A minimum HCLPF value of 1.67*SSE will be met for the SSCs identified in DCD, Table 19.2-4.

This COL information item (COL Information Item 19.2.6-1-A, "Seismic High Confidence Low Probability of Failure Margins") is acceptable.

19.1.5.1.3.2 Significant Accident Sequences and Leading Contributors. 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 (CB), 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 0.84g. This is the result of using an assumed value (i.e., 0.84g) 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 because many of the other mitigating systems depend on dc power 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.

19.1.5.1.4 Conclusion

The applicant performed its PRA-based SMA using an approach acceptable to the staff; therefore, the analysis is acceptable. Through the PRA-based SMA, the applicant has identified significant accident sequences and potentially dominant contributors to core damage in accordance with the Commission's objectives for DC. With COL Information Item 19.2.6-1-A,

the plant HCLPF capacity of 1.67*SSE is assured for the the DC, and therefore the seismic risk is adequately addressed for the DC as required by 52.47(a)(27).

19.1.5.2 Results and Insights from the Internal Fires Risk Analysis

19.1.5.2.1 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 approach that is conservative and bounding with respect to CDF and LRF. For example, the FPRA assumes the worst effects of fire on all 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. The analysis used bounding fire initiating event frequencies, consistent with the nature of the fire analysis.

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

The following analysis tasks, which are described in NUREG/CR-6850, apply to ESBWR FPRA model development:

- 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

The applicant performed subsequent analysis tasks using an approach simpler than that suggested in NUREG/CR-6850. This approach is acceptable because the impact of the detailed analysis will not affect the results from this simplified analysis due to the conservative assumptions used in the ESBWR PRA. Seismic-fire interaction (Task 13) is qualitatively evaluated.

19.1.5.2.1.1 Fire Probabilistic Risk Assessment Assumptions. The fire risk analysis is performed using conservative and bounding assumptions because the detailed cable routings and ignition sources have not been specified.. 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 the beginning of the event.

- Design requirements have been implemented to prevent spurious actuations induced by a single fire in the RB. However, the PRA assumes that fire propagation in the RB will lead to inadvertent opening of relief valves (IORV).

Because the insights from the FPRA analysis impact the detailed design, the FPRA analysis includes more specific assumptions about each task as a result of that process. Section 12.2 of the PRA report, Revision 4, describes the detailed assumptions.

19.1.5.2.1.2 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, Revision 3, "Physical Independence of Electrical Systems," issued February 2005, and Institute of Electrical and Electronic Engineers (IEEE) Standard 384-1992, "Standard Criteria for Independence of Class 1E Equipment and Circuits." In addition, the ESBWR design complies with the more stringent NRC policy statement of SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors," dated January 19, 1989, 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. Fires within the containment are not credible during plant operation because the containment is inerted..

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

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

- separation of redundant remote shutdown panels

The application of these separation criteria ensures 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 fire areas defined in Chapter 9 of DCD, Tier 2, Revision 6, which covers all of the protected area. The plant boundary includes all fire areas defined in the fire hazard analysis (FHA). The FHA fire areas include the RB, fuel building (FB), CB, turbine building (TB), electrical building (EB), radwaste building, and yard area.

19.1.5.2.1.3 Task 2: Fire Probabilistic Risk Assessment Component Selection Assumptions. The equipment and 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)

19.1.5.2.1.4 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 cabling is based on the preliminary design of panels and remote multiplexing units (RMUs). Because of the limited design detail available at the DC stage, detailed circuits are not available for evaluation. However, the ESBWR digital I&C system design is required to prevent spurious actuations.

19.1.5.2.1.5 Task 4: Qualitative Screening Criteria. The analysis used the following criteria to screen fire areas from consideration:

- 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; (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 TS because of invoking a limiting condition of operation (LCO).

19.1.5.2.1.6 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.

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

19.1.5.2.1.7 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.

19.1.5.2.2 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, as stated in Section 19.1.1 of this report.

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used applicable guidance from SRP Section 19.0, Revision 2 in its review.

19.1.5.2.3 Staff Evaluation

19.1.5.2.3.1 Evaluation of Methodology and Approach. The ESBWR internal FPRA is performed according to the guidance in NUREG/CR-6850. The FPRA method documented in this report reflects a state-of-the-art fire risk analysis approach. 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 NEDO/NEDE-33386, Revision 1, "ESBWR Plant Flood Zone Definition Drawings and Other PRA Support Information," issued May 2009. DCD, Tier 2, Revision 6, Section 9A (Figures 9A.2-1 through 9A.2-33), includes the plant layout drawings for fire areas and fire boundaries. Tables 9A.5-1 through 9A.5-7 in DCD, Tier 2, Revision 6, 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 assumed for the PRA fire model is based on the guidelines for separation criteria. Although the final cable routing could be different from that assumed in the PRA model, reasonable cable variations will not significantly impact the PRA results. The staff finds this approach to be acceptable.

In a number of RAIs, the staff requested specific information about the locations of the RWCU pumps and trains, a list of screened-out fire areas, and an explanation as to why the analysis did not address fires in the yard area and remote shutdown panels. The applicant addressed these questions in its responses as discussed below.

The components of RWCU trains are located in separate fire areas, as shown in Figures 9A.2-1 and 9A.2-10. Table 12.6-2 of the PRA report, Section 12A, contains a list of screened-out areas. The remote shutdown panels will be located in separate fire areas in the RB. 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 (NFPA) smoke control guidelines and removing smoke with HVAC systems. GEH indicated that a balanced HVAC system and the safety-related digital control and instrumentation system (Q-DCIS) address both heat dissipation and smoke removal issues.

GEH is preparing a balanced detailed HVAC system design (i.e., implementing separation criteria of RB HVAC subsystems, 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 considered to be negligible because balanced HVAC and Q-DCIS system designs address smoke removal issues.

The ESBWR FPRA has evaluated potential fire-induced spurious valve actuations causing LOCA or incorrect valve lineup. According to the FPRA a single fire in any fire area will not cause spurious actuation of DPVs, SRVs, or GDCS squib valves and result in a LOCA. The ESBWR I&C system is digital. A spurious signal cannot be induced by the fire damage in a fiber-optic cable. With the minimal use of the hard wires, the consequences of a postulated fire are reduced. Furthermore, two or three load drivers must be actuated simultaneously to activate 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 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 were modeled by the applicant 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. To perform online maintenance, some of the fire doors may be open for access and this is not modeled in the baseline ESBWR fire PRA model. The risk increase associated with the open doors will be controlled by the plant's risk management program of 10 CFR 50.65(a)(4) when the plant is in operation, therefore the staff finds this approach acceptable.

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. This is a conservative approach, which the staff finds 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 DCD, Tier 2, Revision 6. The staff documented its review of the applicant's PRA maintenance and update program in Section 19.1.2.3.4 of this report.

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 plant-specific fire analysis are bounded by the PRA described in DCD, Tier 2, Revision 6; otherwise, the COL applicant will perform a modified PRA fire analysis. This is acceptable to the staff.

19.1.5.2.3.2 <u>Evaluation of Significant Accident Sequences and Leading Contributors</u>. The total CDF for fire events at full power is 1.25×10^{-8}/yr. The total LRF for fire events at full power is 1.56×10^{-9}/yr.

The staff requested that the applicant provide a characterization of the dominant accident sequences and associated major contributors to CDF for each sequence. Combined, the following 10 fire scenarios, which are the leading contributors to core damage, contribute to about 80 percent of the total fire CDF:

(1) A postulated fire in F9160 (cable tunnel B) fails all the cabling for train B components of non-safety-related systems, including all the power cables.

(2) A postulated fire in F9150 (cable tunnel A) fails all the cabling for train A components of non-safety-related systems, including all the power cables.

(3) A postulated fire in FSWYD (switchyard) results in an LOPP, and no recovery of offsite power is assumed.

(4) 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, and FPS pump U43-P1B. The fire propagates to the DPS room.

(5) 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, and SLCS train A. It also fails Division I safety-related control signals.

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

(7) A postulated fire in F5350 (electrical equipment A) fails the train A 6.9-kilovolt switchgear.

(8) A postulated fire in F3302 (non-1E electrical room) fails RWCU train B, FAPCS train B, CRD pump B, condensate and feedwater system, RCCWS train B, and FPS pump U43-P1B. The fire propagates to cable tunnel B.

(9) A postulated fire in F4197 (turbine equipment) fails condensate and feedwater system, turbine closed cooling water system (TCCWS), and the instrument air and service air systems.

(10) A postulated fire in F3302 (non-1E electrical room) fails RWCU train B, FAPCS train B, CRD pump B, condensate and feedwater system, RCCWS train B, and FPS pump U43-P1B.

The most important fire sequences involve fires in the cable tunnels that disable either plant investment protection (PIP)-A or PIP-B control signals and power supplies. Postulated fire propagation between the N-DCIS A room and the DPS room also has a relatively higher contribution because it disables both the PIP-A and DPS controls. Other noteworthy fire-induced initiating events include the fires in the switchyard that result in LOPP and in the RB that disable Division I or II electrical equipment.

The quantification of the LRF 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 the event of fire propagation between the N-DCIS A room and the DPS room contributes to approximately 44 percent of the total LRF.

Based on the preceding discussion, the staff concludes that the applicant has adequately discussed the dominant accident sequences.

19.1.5.2.3.3 Evaluation of Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions. 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 except for fire propagation between the N-DCIS train A room and the DPS room in the CB. The reason for this exception is that the fire in the N-DCIS room is postulated to fail RWCU train A, FAPCS train A, CRD pump A, condensate and feedwater system, RCCWS train A, and FPS pump U43-P1B. Together with the equipment in the DPS room, these systems are important to preventing core damage.

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 leads to significant contributions from the ATWS sequences to the total fire CDF.

Fire in the control room traditionally requires the operator to take actions to control the plant manually. One relevant feature of 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.

Based on the preceding discussion, the staff concludes that the applicant has successfully identified risk significant functions and features.

19.1.5.2.3.4. Evaluation of Insights from the Uncertainty, Importance, and Sensitivity Analyses. The applicant performed a sensitivity analysis for the Level 1 fire model using focused PRA studies. The analysis evaluated the impact of failing all non-safety systems, along with the impact of failing all non-safety systems except those designated as RTNSS. The former study generated a CDF of 5.13×10^{-5}/yr, and the latter study generated a CDF of 2.95×10^{-7}/yr.

The results for the focused fire sensitivity study showed significant impact on the CDF with the failure of non-safety systems, both within the scope of RTNSS and outside the scope of RTNSS. The inclusion of the RTNSS SSCs in the model reduces the CDF by approximately two orders of magnitude compared to crediting safety-related systems only. The results of the Level 1 focused fire sensitivity study show that the NRC goal of 1×10^{-4}/yr CDF is met for the baseline Level 1 fire analysis, the focused study, and the RTNSS sensitivity analyses. The fire analysis is very conservative with no credit taken for fire suppression or fire severity factors.

The Level 2 focused fire sensitivity study, in which all non-safety systems are failed, generated an nTSL (nontechnical-specification leakage, which is equivalent to LRF) release frequency of 4.18×10^{-5}/yr. The RTNSS study generated an nTSL release frequency of 8.34×10^{-8}/yr. The results for these studies show significant impact to the nTSL release frequency with the failure of non-safety systems both inside and outside the scope of RTNSS. The results show a decrease of three orders of magnitude in the nTSL frequency with the RTNSS SSCs available

compared to safety-related systems only. The nTSL results of the Level 2 focused fire sensitivity study show that the NRC goal of 1×10^{-6}/yr for LRF is met when RTNSS SSCs are included, but not met for the focused Level 2 fire study with all non-safety systems failed.

Tables in Section 11 of the PRA report, Revision 4, present the results of the 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 through 11.3-25, 11.3-28, 11.3-30, 11.3-32, 11.3-34, and 11.3-36 through 11.3-39. Section 11 does not define "Difference." Because it could not reproduce some of the results, the staff was concerned that there may be some errors in the calculation of "Difference." The staff tracked RAI 19.1-160 as an open item in the SER with open items.

In Revision 4 of the PRA report, the applicant provided the definition of "Difference." The applicant also revised all of the tables mentioned in RAI 19.1-160 to show the correct values based on the definition of "Difference." The staff evaluated the results and verified its accuracy. Therefore, RAI 19.1-160 and the associated open item are resolved.

In addition to the focused PRA studies, the applicant conducted a series of sensitivity studies 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 cabinets from other cabinets in Room 3301. The risk increase associated with the merging of Rooms 3301 and 3140 into a single fire area is moderate. In both cases, the resulting total fire risks are still more than two orders of magnitude lower than the NRC goals for CDF and LRF (i.e., 1×10^{-4}/yr for CDF and 1×10^{-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 operation (Mode 1) and deinerted in shutdown (Modes 2, 3, or 4). The results of the sensitivity studies indicate that total baseline CDF and LRF in these modes are at least three orders of magnitude below the goals.

The results of the fire ignition frequencies sensitivity study confirmed that the 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 plant service water system (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 N-DCIS

electrical room A and the DPS room, and the barrier between the N-DCIS electrical room B and cable tunnel B.

The results of importance measures for the at-power fire CDF 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.

19.1.5.2.4 Conclusion

The staff has reviewed the results and insights derived from the fire 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.

19.1.5.3 Results and Insights from Internal Flooding Analysis

19.1.5.3.1 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:

- RB
- CB
- FB
- TB
- EB
- 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 occurring in the remaining ESBWR buildings because those flood waters 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, then the flood zone is not analyzed further.However, the failure probability of these components is considered in the PRA model.

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

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

- 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 or spraying or both 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.

- Electrical circuit fault protection is assumed to have 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.

- The doors connecting the control and RBs with the EB galleries are assumed to be watertight; flooding of the galleries up to the ground-level doors are assumed to generate an alarm in the control room, and procedures direct the immediate closure of the doors upon receipt of an alarm.

- The operation of the components located in containment are assumed to be unaffected 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.

The applicant performed a screening 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 were considered as potential flood sources at power:

- NBS
- CRDS
- SLCS
- FAPCS
- RWCU/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
- PSWS
- diesel generator
- 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 the internal events PRA analysis already models LOCA scenarios in containment, the internal flooding analysis does not model these events. Therefore, the at-power internal flooding analysis does not analyze further any flood scenarios in containment.

The applicant calculated the initiating event frequency 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, 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 applicant used the general transient initiating event category and associated accident sequence logic 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. The applicant also performed a Level 2 analysis for the flooding scenarios.

19.1.5.3.2 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, as stated in Section 19.1.1 of this report.

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used applicable guidance from SRP Section 19.0, Revision 2 in its review.

19.1.5.3.3 Staff Evaluation

19.1.5.3.3.1 Evaluation of Methodology and Approach. 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, Revision 1, provides a list of the equipment located in each flooding area the PRA credits for accident mitigation. The equipment includes safety as well as non-safety components.

The PRA report provides a list of screened flooding areas. The screened areas are those having not been considered as potential flood sources, or the areas containing PRA equipment which have no probabilistic impact. The staff agreed with this assessment..

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. The staff finds this assumption acceptable.

Flooding propagates between areas. The model includes those areas where propagation is likely, 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 in which flooding from main steam and feedwater pipes located in the steam tunnel propagates to the RB.

The mission time of the active equipment credited in the flooding risk analysis is 24 hours. The internal events PRA uses the same timeframe; therefore, the staff finds it acceptable.

The internal flooding analysis treats breaks in support systems, such as the service water system, RCCWS, and TCCWS, explicitly instead of assigning them the same consequences as the failure of the systems themselves, as described in Revision 3 of the applicant's PRA.

The analysis applied a recovery factor of 0.01 to the circulating water flooding scenario in the TB 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 is 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 DCD, Tier 2, Revision 6. The staff's review of the applicant's PRA maintenance and update program appears in Section 19.1.2.3.4 of this report.

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 flooding in these areas.

19.1.5.3.3.2 Evaluation of Significant Accident Sequences and Leading Contributors to Risk. The total CDF for full-power internal flooding events is 3.30×10^{-9}/yr. The total release frequency for internal flooding events excluding TSL at full power is 4.8×10^{-10}/yr.

The following 10 flooding scenarios are the leading contributors to core damage and, combined, they contribute to about 36 percent of the total flooding CDF:

(1) Flooding in the TB main condenser area caused by a small pipe leak of RWCU/SDC and CCF of rods to insert result in core damage.

(2) Flooding in the TB at elevation 1,400 millimeters caused by a large pipe leak in the condensate and feedwater system and CCF of rods to insert result in core damage.

(3) Flooding in the TB at elevation 4,650 millimeters caused by a large pipe leak in the condensate and feedwater system and CCF of rods to insert result in core damage.

(4) Flooding in the TB at elevation 4,650 millimeters caused by a large pipe leak of the “A” PSWS train and CCF of rods to insert result in core damage.

(5) Flooding in the TB at elevation 4,650 millimeters caused by a large pipe leak of the “B” PSWS train and CCF of rods to insert result in core damage.

(6) Flooding in the TB at elevation 1,400 millimeters caused by a large pipe leak of the “A” PSWS train and CCF of rods to insert result in core damage.

(7) Flooding in the TB at elevation 1,400 millimeters caused by a large pipe leak of the “B” PSWS train and CCF of rods to insert result in core damage.

(8) Flooding in the RB at elevation 11,500 millimeters caused by a large pipe leak of the ”A” RWCU/SDC and CCF of rods to insert result in core damage.

(9) Flooding in the TB at elevation 1,400 millimeters caused by a large pipe leak in the FPS and CCF of rods to insert result in core damage.

(10) Flooding in the TB at elevation 4,650 millimeters caused by a large pipe leak in the FPS and CCF of rods to insert result in core damage.

The CET release category frequencies are summarized as follows:

Release Category	Frequency
TSL	2.83×10^{-9}/yr
Containment bypass (BYP)	2.46×10^{-10}/yr
Filtered release (FR)	2.10×10^{-10}/yr
Overpressure because of failure of long-term CHR	1.98×10^{-11}/yr
Overpressure because of vacuum breaker failure	1.81×10^{-12}/yr

The combined release frequency excluding TSL is about 4.78×10^{-10}/yr.

19.1.5.3.3.3 Evaluation of Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions. 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 RBs
- not locating flood sources with a significant volume of water in the electrical equipment rooms located in the RB
- 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 leaks in the TB main condenser area, the EB general area, the TB’s first floor, and the service water pumphouse.

The cutsets associated with these sequences involve the common-cause software failures on the digital control systems and failures of the same single components that disable the ac power supplies or the IC/PCCS pool makeup.

During the initial phase of the ESBWR design, the applicant identified a significant flood risk in the CB because of a break in FPS piping. Based on this PRA insight, the design specifications now require that the FPS pipes and fire hose stations be relocated outside of the CB 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 estimated offsite consequences resulting from external events under at-power conditions are less than the defined individual, societal, and radiation dose limits.

19.1.5.3.3.4 Evaluation of Insights from the Uncertainty, Importance, and Sensitivity Analyses. The applicant performed a sensitivity analysis for Level 1 internal flooding using focused PRA studies involving (1) failing all non-safety systems and (2) failing all non-safety systems except those designated as RTNSS. GEH performed this sensitivity analysis using the conservative PRA flooding model developed for the PRA report, Revision 4. The flooding baseline CDF for this model is $6.95\times10\text{-}^{9}$/yr. The Level 1 focused flood analysis with all non-safety systems failed generated a CDF of 9.39×10^{-5}/yr; the RTNSS study generated a CDF of 4.36×10^{-7}/yr. The results for the focused flood sensitivity analysis showed significant impact to the CDF upon failure of the non-safety systems, both with and without RTNSS. The inclusion of RTNSS in the model reduces the CDF by approximately three orders of magnitude as compared to the CDF when crediting safety-related systems only. Based on the Level 1 focused flood sensitivity analysis results, both the focused flood model and the RTNSS sensitivity scenarios meet the NRC goal of 1×10^{-4}/yr CDF.

The Level 2 focused flood model with all non-safety systems failed generated an nTSL (equivalent to LRF) release frequency of 9.22×10^{-6}/yr. The RTNSS study generated an nTSL release frequency of 3.12×10^{-7}/yr. The results of the focused flood sensitivity study showed significant impact to the nTSL release frequency with the failure of non-safety systems, both with and without RTNSS. The results showed a decrease of about three orders of magnitude in the nTSL frequency with RTNSS available as compared to the frequency when crediting safety-related systems only. Based on the Level 2 focused flood sensitivity study nTSL results, the NRC goal of 1×10^{-6}/yr LRF is met for the RTNSS sensitivity scenarios, but not for the focused flood model which does not credit non-safety systems. By crediting the RTNSS systems, the NRC goal for LRF in the Level 2 flooding analysis are met.

The staff issued RAI 19.1-161 asking GEH to correct a typographical error in Table 11.3-30 and revise the text to reflect that the goal of 1×10^{-6}/yr has been exceeded for the Level 2 flood focused model crediting only the safety systems. The staff was tracking RAI 19.1-161 as an open item in the SER with open items. The PRA report, Revision 4, presents these corrections. Therefore, RAI 19.1-161 and the associated open item are resolved.

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.

19.1.5.3.4 Conclusion

The staff has reviewed the results and insights derived from the flooding risk analysis 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.

19.1.5.4 Results and Insights from High-Winds Analysis

19.1.5.4.1 Summary of Technical Information

The staff's review of the ESBWR high-winds risk assessment is based on the results reported in Section 14 of the PRA report, Revision 4, and DCD, Tier 2, Revision 6, Section 19.2.3.2.3. The applicant developed separate ESBWR high-winds risk assessments for tornado initiators and hurricane initiators. The risk assessment and the staff's evaluation encompass plant operation at power, in cold shutdown, and in refueling modes. Section 19.1.6.1 of this report discusses the risk from high winds at shutdown and refueling.

The applicant's high-winds risk analysis presented in the PRA report, Revision 4, is based on the robustness of the ESBWR structures. The ESBWR is designed for a tornado wind load of 147.5 meters per second (m/s) (330 miles per hour (mph)), which is assumed to be the maximum windspeed that will not challenge the safety-related structures. In addition, the ESBWR is designed for extreme windspeed (i.e., hurricanes) of 67.1 m/s (150 mph) for seismic Category I and II structures and 58.1 m/s (130 mph) for nonseismic structures. The only exceptions are the TB, service water building, and EB structures, which are nonseismic and have a design-basis hurricane basic windspeed of 195 mph.

The PRA assumes seismic Category I and II structures will be essentially undamaged by the windspeed of all hurricanes and tornadoes. The PRA assumes that hurricane and tornado missiles will not do significant damage to seismic Category I structures or equipment that is below grade. The PRA also assumes that only the most powerful tornado missiles can significantly damage seismic Category II structures. These assumptions are important because most of the equipment needed to keep the core cool during the first 72 hours of the event is located in the RB, which is a seismic Category I structure.

Because high winds are not expected to damage the most important structures housing safety and non-safety equipment, and because loss of offsite power (or LOPP) would be expected in a high-winds event, the applicant chose to model high winds in the PRA by developing the event tree for LOPP. The assessment uses the internal events PRA event tree for LOPP, system fault trees (modified for loss of certain components and structures caused by high winds), and success criteria for LOPP events to calculate the risk from extended loss of offsite power resulting from high winds.

As documented in Tables 14.6-1 and 17.1-1 of the PRA report, Revision 4, the CDF for high winds at power is estimated to be 9×10^{-9}/yr, which is approximately one-half the estimated internal events CDF. The PRA estimates LRF for high winds at power to be 1×10^{-9}/yr, which is comparable to internal events.

The insights about risk from tornadoes and hurricanes in the PRA are similar to those associated with internal event long-term LOPP sequences.

19.1.5.4.1.1 Methodology and Approach for Tornadoes. The tornado risk analysis presented in the PRA report, Revision 4, is based on the premise that (1) plant structures built to seismic Category I and II requirements are invulnerable to the direct effects of tornado winds, (2) seismic Category I structures will not experience any significant damage from tornado missiles, (3) equipment located below grade will not be damaged by tornado missiles, and (4) seismic Category II structures will only be significantly damaged by the most powerful tornado missiles. The PRA reports results for both the Fujita Scale[6] (or F-scale) and the Enhanced Fujita Scale (EF-scale). The assessment assumes that, following a strike by winds from an EF2 or greater tornado, preferred power will be lost (i.e., there will be an extended loss of offsite power that cannot be recovered). The assessment assumes equipment housed in seismic Category I and II structures will operate with normal equipment failure rates. Table 14.3-2, "ESBWR Tornado Wind—PRA Predicted Structure Damage," in the PRA report, Revision 4, provides the assumptions on the amount of damage that structures would receive from various classes of tornadoes. (Table 19.1.5.4-1 of this report summarizes this table.) The applicant assumed that EF5 tornado missiles would significantly damage seismic Category II structures but not seismic Category I. In contrast, the applicant assumed that hurricane missiles (which have a lower velocity than some tornado missiles) would not significantly damage any seismic Category I or II structures.

The applicant performed its tornado risk assessment by taking the following steps:

- Calculate the tornado hazard frequency.
- Evaluate the tornado-induced plant effects.
- Calculate the tornado-induced CDFs and release frequencies.

The risk assessment uses the data and method from NUREG/CR-4461, Revision 1, "Tornado Climatology of the Contiguous United States," to calculate the tornado strike frequency. The risk assessment segregates the data into three bins—EF2 and EF3 tornadoes, EF4 tornadoes, and EF5 tornadoes. The number of EF0 and EF1 tornadoes observed was discarded because the applicant assumed that these tornadoes would not significantly damage structures on site and would cause only LOPP. The PRA states that the frequency of such power losses is captured under the initiating events for internal events LOPP. The applicant chose to use data from the central region of the United States, which should encompass most ESBWR sites, because the frequencies of occurrence and tornado intensities in that region are the highest in the nation. In addition, in calculating the tornado strike frequencies, the applicant used a characteristic length, w_s, equal to 400 feet, which is twice that assumed in NUREG/CR-4461, Revision 1, effectively doubling the frequency of the assumed tornado strikes. This results in an at-power strike frequency for the ESBWR plant design for EF2/EF3, EF4, and EF5 tornadoes of 1×10^{-4}/yr, 4×10^{-6}/yr, and 5×10^{-7}/yr, respectively.

The applicant then entered these occurrence frequencies into the at-power internal events PRA event tree for LOPP. Fault trees, developed for the at-power LOPP event tree, are modified to take into account the effects that tornadoes will have on various components and structures. The fault trees are then input into the LOPP event tree to estimate the CDF from extended loss of offsite power due to tornadoes. The high-winds risk assessment for tornadoes assumes that equipment located in the yard or in nonseismic structures, including the TB, service water building, and EB, will always fail if an EF2 or stronger tornado strikes the site. RTNSS

[6] 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. The Enhanced Fujita Scale is an updated version of the Fujita Scale that estimates the rotational speeds of tornadoes somewhat lower than the Fujita Scale.

structures are assumed to fail for EF4 and stronger tornadoes, and seismic Category II structures are assumed to be damaged by tornado missiles for EF5 tornadoes.

Table 14.6-1 in the PRA report, Revision 4, displays the estimated CDF and LRF from high winds when the plant is at power and when the plant is in shutdown. The estimated CDF from an at-power EF2 to EF3 tornado strike on an ESBWR is 9×10^{-12}/yr. The estimated CDF for EF4 tornadoes at power is 8×10^{-10}/yr. The estimated CDF for EF5 tornadoes at power is 1×10^{-10}/yr. The estimated CDF from all tornadoes when an ESBWR is shut down is 5×10^{-11}/yr. The LRF from tornadoes when the plant is at power is 8×10^{-10}/yr for EF4 tornadoes and 1×10^{-10}/yr for EF5 tornadoes, with LRF for EF2/EF3 tornadoes a much smaller contributor. The LRF from EF4 and EF5 tornadoes is significantly larger because the applicant assumed that the TB which houses the MSIVs would be destroyed if such powerful tornadoes were to strike the plant.

19.1.5.4.1.2 Methodology and Approach for Hurricanes. Similar to tornadoes, the applicant based its risk analysis for hurricanes on the premise that plant structures built to seismic Category I and II requirements and to RTNSS standards would not be significantly damaged by hurricane winds and associated missiles. The assessment assumes that the equipment housed within these structures will operate with normal equipment failure rates during and after a hurricane. Nonseismic structures, with the exception of the TB, service water building, and EB structures, are assumed to fail for Category 3 and higher hurricanes, as is equipment located in the open. All structures are assumed to be able to withstand the winds associated with Category 1 and Category 2 hurricanes. The only impact on the site from such hurricanes is LOPP, with no additional equipment failures associated with the hurricane. Offsite power is assumed to be lost and be unrecoverable for all hurricanes. The analysis assumes that the frequency of losses of power for Category 1 and Category 2 hurricanes is subsumed in the risk assessment's treatment of LOPP for internal events. The risk assessment assumes that the maximum speed of hurricanes is greater than 155 mph.

The high-winds risk analysis presented in the PRA report, Revision 4, makes the following additional assumptions and statements regarding hurricanes:

- The classification of the hurricane winds used in the ESBWR high-winds analysis is based on the Saffir-Simpson scale.

- All at-power ESBWR high-winds analyses, including hurricane high winds, assume the plant is operating at full power. This approach is assumed to be conservative for the hurricane high-winds analysis because sufficient advanced warning and procedures would enable the plant to be placed into a safe condition (shutdown operations) before a high-winds event. Implicit in this assumption is that (1) the plant will go to Mode 4 and will not deinert in Mode 4 when the plant shuts down in anticipation of a hurricane strike, and (2) in anticipation of a hurricane strike, the plant will ensure that equipment credited in the high-winds PRA is available. These implicit assumptions are captured as important PRA insights.

- The FPS piping that provides makeup water to the ICS/PCC pool and water for reactor water coolant/inventory control is dedicated piping that has no fire hydrants, standpipes, or large piping external to a seismic Category I structure and has no piping that is exposed such that it could be damaged by a hurricane-induced missile.

- Straight winds are of lesser velocity than hurricanes or tornadoes and are assumed to pose minimal challenges to the plant design.

- 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).

The applicant performed a hurricane risk assessment taking the following steps:

- Calculate the hurricane-induced LOPP frequency.
- Evaluate the hurricane-induced plant effects.
- Calculate the hurricane-induced CDFs and release frequencies.

The ESBWR hurricane risk analysis presented in the PRA report, Revision 4, does not use structural fragility curves to evaluate the potential that hurricane winds might significantly damage seismic Category I, seismic Category II, or RTNSS structures. The analysis assumes that no significant damage would occur to these structures because of their robust design criteria; instead, the analysis uses data from NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants," Volume 1, "Analysis of Loss of Offsite Power Events: 1986–2004," to estimate the hurricane-induced LOPP frequency. The ESBWR hurricane risk assessment took the number of losses of offsite power that occurred at nuclear power plants in Florida, Louisiana, and North Carolina as the result of hurricanes during a specific 19-year period and divided it by the number of reactor critical-years (cyr) that nuclear power plants located in these States had operated during the same period. The risk assessment uses this estimate (i.e., 7.6×10^{-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 staff considers this estimate to be conservative for most sites in the United States.

The applicant then entered this occurrence frequency into the at-power internal events PRA event tree for LOPP. Fault trees, developed for the at-power LOPP event tree, were modified to take into account the effects that hurricanes will have on various components and structures, and the fault trees were then input into the LOPP event tree to estimate the CDF from extended loss of offsite power resulting from tornadoes.

In the DCD, the applicant stated that the high-winds risk assessment was conservative in that it did not credit alternative, onsite water sources beyond the condensate storage tank. However, GEH did not quantify the degree of conservatism.

Section 14.7, "Insights," of the PRA report, Revision 4, states that the estimated CDF and LRF for all analyzed scenarios, while using a bounding analysis, were similar to the internal events results. In RAI 19.1-185, the staff disagreed with the term "bounding analysis" for all sites. In its response, dated November 17, 2009, the applicant provided an acceptable draft modification to the PRA text. The staff confirmed the modification was made in Revision 5 to the PRA and considers this RAI resolved.

In RAI 19.1-165 the applicant was asked if, in light of the variation in strike frequency among different sites in the United States, they had done a sensitivity study on hurricane strike frequency. RAI 19.1-165 was being tracked as an open item in the SER with open items. In

their response dated March 8, 2008, the applicant stated that a sensitivity study was not performed because their analysis of strike frequency was bounding for all sites. As discussed above, the applicant subsequently characterized their analysis in the DCD as bounding for most sites. The staff finds that a sensitivity study is not necessary because the applicant's analysis bounds the frequency for most sites; and, COL applicants will need to provide a site specific analysis if their site is not bounded by the analysis in the referenced DCD. Therefore, RAI 19.1-165 and the associated open item are resolved.

Table 14.6-1 in the PRA report, Revision 4, displays the estimated CDF from high-winds initiators. The estimated CDF from an at-power hurricane is 8×10^{-9}/yr. The at-power CDF estimate for hurricanes is comparable to total CDF for internal events (i.e., approximately one-half the internal events estimated CDF).

The applicant estimated the expected LRF caused by hurricanes. GEH assumed that at-power events would start with the containment intact, which required estimating the conditional probability of containment failure given core damage from an extended loss of offsite power event with hurricane-induced damage to some structures. The LRF for hurricanes at power is 3×10^{-10}/yr. This estimate is a factor of 3 less than that estimated for tornadoes, although the CDF from hurricanes is higher than for tornadoes. The at-power hurricane LRF is smaller than the at-power tornado LRF because EF4 and EF5 tornadoes fail all equipment inside the TB, including the MSIVs, while hurricane winds are assumed to never reach a velocity that would significantly damage the TB.

19.1.5.4.1.3 Risk-Significant Functions and Features. Listed below are key ESBWR design features and functions identified in the PRA report, Revision 4, that significantly reduce the expected CDF associated with tornado and hurricane strikes that produce an extended LOPP as compared to the CDF for operating BWR designs from tornadoes and hurricanes. The risk-significant functions of the following features are primarily the same as those identified for LOPP for internal events:

- The ESBWR design stores a significant amount of water 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 RB are generally thicker than those of the RBs 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 the walls protecting ICs at older operating BWRs.
- The ESBWR long-term DHR design relies on more robust dc power as compared to operating reactors, where safety-related dc power generally will last only 4 to 8 hours. Such power will last 72 hours in the ESBWR design.
- Long-term core cooling for extended loss of offsite power events for the ESBWR design depends in great part on gravity injection rather than the turbine-driven pumps on which most operating BWRs depend.
- 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 the 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.

- For the ESBWR design, hurricanes should have no possibility of significantly damaging seismic Category I, seismic Category II, or RTNSS structures.

- The FPS components located outside the RB that are needed for FAPCS makeup (this system provides long-term makeup to the pools in the RB that cool the reactor) are designed to seismic Category I standards and can withstand tornado missiles and other natural phenomena such as hurricanes.

19.1.5.4.1.4 Significant At-Power Sequences and Leading Contributors. The CCFs of the following SSCs are significant for high-winds events based on the reported risk achievement worth (RAW) values (all near or in excess of 400) in Table 14.6-4 in the PRA report, Revision 4:

- 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 no. 126
- control rods insertion
- DPS load drivers

Of the top 50 cutsets for high-winds events, four were caused by tornado-induced LOPP. The rest were caused by hurricane-induced LOPP.

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

- software CCF
- CCF of check valves in the GDCS
- CCF of DPVs
- control rods failure to insert

- failure of any SRV to reclose following ATWS
- failure of squib valves
- operator failure to inject using the FPS or a fire truck
- operator failure to recognize need for depressurization
- operator failure to recognize need for low-pressure makeup after depressurization

19.1.5.4.2 Regulatory Criteria

In Section 19.1.1 of this report, the staff considered the results and insights from the high-winds risk assessment with respect to the Commission's objectives for new reactor designs.

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used applicable guidance from SRP Section 19.0, Revision 2 in its review.

19.1.5.4.3 Staff Evaluation

In Revision 4 of the PRA report, the applicant modified the data it used in the high-winds risk assessment and reevaluated the risk. Because of this, most of the staff's RAIs relating to high winds, which referenced Revision 3 of the PRA report and were described in the staff's SER with open items, are no longer pertinent to the high-winds assessment. This SER specifically discusses RAIs that do pertain to the assessment presented in the PRA report, Revision 4.

19.1.5.4.3.1 Tornado Hazard Frequency. The staff confirmed that the applicant appropriately used the data and methodology from NUREG/CR-4461, Revision 1, for estimating tornado strike frequencies. To ensure that the strike frequency was bounding for most sites in the United States, the applicant used frequencies generated from data for the central region of the United States, which is the region of the county with the highest occurrence rate of tornadoes and the highest tornado intensities.

19.1.5.4.3.2 Evaluation of the Effects of Tornado Strikes. The ESBWR high-winds risk analysis makes assumptions in its tornado risk assessment. The staff reviewed these assumptions and found them to be reasonable for estimating the CDF associated with tornadoes damaging an ESBWR design.

The staff reviewed the at-power LOPP event tree to determine whether the systems, associated support systems, and structures housing the systems and support systems were appropriately credited for tornado-strike events. The staff found the applicant's LOPP event tree appropriate for evaluating tornado strikes, given the assumptions made in the PRA. The staff's review supports the applicant's conclusion that the expected CDF from tornadoes is very low because of: (1) the robustness of the seismic Category I and II structures, (2) the low frequency of tornado occurrence, and (3) the low conditional probabilities associated with a tornado actually hitting an ESBWR site.

19.1.5.4.3.3 Hurricane Hazard Frequency. To estimate the frequency of hurricane strikes, the applicant averaged the frequency of hurricane-induced loss of offsite power at nuclear power plants located on shorelines and in areas with high hurricane return rates in Florida, Louisiana, and North Carolina during a 19-year period. The staff finds that this estimate of hurricane strikes is bounding for most sites in the United States, with the possible exception of particular coastal sites along the Gulf Coast or the Atlantic Ocean coast from North Carolina southward. The staff confirmed that the applicant used the data from NUREG/CR-6890, Volume 1, for

estimating hurricane strike frequencies. In response to RAI 19.1-185, the applicant modified the PRA report, Revision 5, to state that, if site-specific high-winds frequencies are estimated to be greater than the frequencies in the PRA, then the COL applicant should perform a departure analysis and apply the appropriate measures. Therefore, RAI 19.1-185 is resolved.

19.1.5.4.3.4 Evaluation of the Effects of Hurricane Strikes. The staff's review evaluated the assumption that seismic Category I, seismic Category II, and RTNSS structures are essentially undamaged by hurricanes, 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 hurricane strike events.

The high-winds assessment assumes that it is impossible for Category 4 or Category 5 hurricanes to significantly damage equipment in seismic Category I or Category II buildings in a manner that can cause core damage. In RAI 19.1-169, the staff asked the applicant to explain its basis for this assumption. RAI 19.1-169 was tracked as an open item in the SER with open items. In its response, dated March 8, 2008, the applicant referred the staff to the response to RAI 19.1-167, which stated that the buildings were built to withstand design-bases seismic events and therefore were assumed to be able to withstand high winds. The staff found this response insufficient and, in RAI 19.1-169, Supplement 1, asked the applicant to provide an engineering basis to explain why there is zero probability that hurricanes or tornados can damage seismic Category I or II structures. The staff later supplemented the RAI and asked the applicant to address the possibility of design flaws or construction errors that might lead to weaknesses in the as-built design that would make the plant vulnerable to such tornado missiles or winds. In its response, dated August 1, 2008, the applicant again provided a deterministic explanation rather than a probabilistic one. The rationale was presented in terms of margins of forces designed for versus forces expected. The staff found this an unacceptable response.

In RAI 19.1-169, Supplement 2, the staff again requested the applicant to either (1) provide a probabilistic defense for its use of seven orders of magnitude reduction in risk that provides an engineering basis for the reduction that links the strengths of the design to specific numerical analyses (e.g., fragility curves) that address conditional probabilities of failure, or (2) provide qualitative arguments as to why high winds do not constitute outliers in risk, qualitative arguments why high winds do not challenge the NRC's safety goals, a discussion of why the risk from high-winds events is lower than for operating plant designs, and a list of safety insights that are important for the as-built, as-operated plant to follow to ensure that the assumptions in the high-winds risk analysis are true and remain valid during the lifetime of the plant. The staff noted that a qualitative analysis would not constitute a PRA, and COL applicants may need to address high winds on a plant-specific probabilistic basis if the Commission has a high-winds risk assessment standard in place 1 year before the first fuel load.

In a November 24, 2008, response to RAI 19.1-169, Supplement 2, the applicant provided fragility curves, including statistical parameters for the lognormal curves representing the fragilities, for one- and three-story concrete buildings based on gust windspeeds over the range of hurricane wind speeds of 75 to 300 mph. The applicant stated that the three-story fragility curve is characteristic of the ESBWR RB, but did not supply any basis for this claim. In its response, the applicant referenced a paper on the fragility of concrete reinforced structures to hurricane winds, but was unable to answer staff questions about the basis for the fragility curves. The staff independently contacted the author of the paper and clarified the basis of the fragility curves cited by GEH. Clarification of how the author determined the fragility curves (i.e., based on actual damage to concrete structures due to hurricanes) leads the staff to conclude that the fragility values are conservative. In addition, the staff discussed the robust nature of

reinforced concrete structures and their ability to withstand high winds with structural experts within the NRC to confirm the insights drawn from the fragility curves. Based on this evaluation, the staff concurs in the assumption that seismic Category I and II structures have an extremely low conditional probability of catastrophic failure due to hurricane winds. Therefore, RAI 19.1-169 and the associated open item are resolved.

The staff found the applicant's LOPP event tree appropriate for evaluating hurricane strikes given the assumptions made in the PRA. The staff finds that the applicant's conclusion that the expected frequency of a hurricane strike resulting in core damage is very low to be reasonable. This is because of (1) the robustness of the seismic Category I, seismic Category II, and RTNSS structures, and (2) placement of pumps, diesel generators, and large water tanks that are capable of refilling the tanks over the core in robust structures.

19.1.5.4.3.5 High Winds—General. The applicant concluded in Revision 4 of the PRA report that the CDF resulting from high winds was not a significant contributor to ESBWR core damage risk. The staff questioned this conclusion in RAI 19.1-181. In its response dated November 17, 2009, the applicant indicated that it would modify the PRA to state that the high-winds at-power risk assessment does not produce significant core damage sequences or insights that differ from the internal events at-power LOPP results. The applicant is to make a similar modification for shutdown events (i.e., with regard to RAI 19.1-182). The staff found these responses acceptable. Therefore, RAI 19.1-181 and RAI 19.1-182 is resolved.

19.1.5.4.3.6 Risk Assessment Limitations. The risk assessment did not appear to evaluate the effect of damage from a hurricane or tornado strike to unprotected equipment located out in the open (e.g., fire hydrants), and the staff asked for clarification of this issue in RAI 19.1-168. The staff was tracking RAI 19.1-168 as an open item in the SER with open items. In its response dated March 8, 2008, the applicant stated that the PRA credits the FPS with providing makeup water to the IC/PCCS pool and water for reactor water coolant/inventory control. The response stated that the supporting equipment for these functions is to be seismic Category I or II. In addition, makeup and inventory control function independently of the fire suppression function (i.e., yard hydrant and piping).

In a followup to this question and in conjunction with the review of the FAPCS, the staff noted in RAI 9.1-16, Supplement 2, that there were apparent inconsistencies in the level of protection afforded FAPCS makeup regarding tornado missiles. The staff also documented its concern about fire hydrants, standpipes, or other large lines that could be attached at some point to the dedicated portion of the FPS connection to the FAPCS for makeup. In its response dated March 23, 2009, the applicant stated that the FPS components located outside the RB which are needed for FAPCS makeup will be designed to seismic Category I standards and will be designed to withstand tornados and other natural phenomena. The dedicated line from the FPS to the FAPCS is not designed to NFPA standards and will not fulfill a fire protection function. Fire hydrants, standpipes, or other large lines will not be attached to the dedicated portion of the FPS designed to provide long-term makeup to pools in the RB. In response to RAI 9.1-16, Supplement 3, the applicant committed to place these attributes in Tier 2 of the DCD. Therefore, RAI 19.1-168 and the associated open item are resolved.

In its review of Revision 6 to the DCD, the staff found that it could not distinguish, in the Tier 1 figures, the seismic Category I line that will have no firefighting requirements placed on it and will only be used for refill of the pools as an RTNSS backup. In RAI 9.1-142, the staff asked the applicant to identify the dedicated line on Figure 2.16.3-1 in Tier 1 of the DCD. In its response dated December 7, 2009, the applicant concurred that the FPS simplified diagrams illustrated in

DCD, Tier 2, Figure 9.5-1 and DCD, Tier 1, Figure 2.16.3-1 should be enhanced to reflect the dedicated, seismic Category I FPS piping that aligns the primary diesel-driven fire pump to the FAPCS isolation lines that provide makeup to the IC/PCCS pools and the spent fuel pool (SFP). The applicant stated it would modify in Revision 7 to the DCD the simplified diagrams in DCD, Tier 2, Figure 9.5-1, and DCD, Tier 1, Figure 2.16.3-1, to reflect separate seismic Category I piping routed from the fire pump enclosure (FPE) to the RB supplying redundant FAPCS connections to IC/PCCS pools and SFP makeup. The applicant indicated that this piping run will be routed in a seismic Category I trench from the FPE to the RB FAPCS manual isolation valves. The staff finds this acceptable and confirmed that Revision 7 to the DCD was modified as stated by the applicant. Therefore, RAI 9.1-142 is resolved.

The PRA report, Revision 2 and DCD, Tier 2, Revision 5, had contradictory statements about the effect on RTNSS structures from tornado missiles (including tornado missiles from EF2 and EF3 tornadoes). The staff raised this issue in RAI 19.1-167. The staff was tracking RAI 19.1-167 as an open item in the SER with open items. In its response dated March 8, 2008, the applicant clarified how it performed the high-winds risk assessment. However, the modifications provided in the PRA report, Revision 3, appeared to the staff to continue to contradict DCD, Tier 2, Revision 5, Section 3.3, and the applicant's response to RAI 19.1-167, dated March 8, 2008. Upon reading the augmented Section 14.5.1 of the PRA report, Revision 3, which estimates CDF due to the impact of high winds on the ESBWR SSCs, it appeared to the staff that Tables 14.3-1 and 14.3-2, "ESBWR Tornado Wind—PRA Predicted Structure Damage," implied that seismic Category II structures will suffer no significant damage from EF4 or EF5 tornados. Furthermore, neither the table nor the surrounding text made direct mention of tornado missiles and their effect on SSCs.

In addition, Section 14.4.1, "Tornado Strike Frequency," in the PRA report, Revision 3, in a discussion about the strike frequency for EF4 and EF5 tornados when the reactor is at power, stated that "EF4/EF5 tornado windspeeds would exceed the design of RTNSS and NS structures, but not seismic Category I or seismic Category II structures. Therefore, for EF4 and EF5 tornados, the equipment located in RTNSS structures and the yard will be assumed to fail." There was no mention of the effect tornado missiles would have on seismic Category II structures. In RAI 19.1-167, Supplement 1, the staff requested clarification of how the risk assessment included the effects of these tornado missiles. In its response dated November 24, 2008, the applicant stated the following:

> For the purpose of the ESBWR NEDO-33201, Revision 3 high winds risk analysis, component failures associated with extreme winds and missiles for EF4 and EF5 tornados were treated similarly. This assumption was made to reduce the complexity of the analysis and also because only a small number of seismic Category II components were credited.

Key assumptions related to tornado missiles and the ESBWR high-winds risk analysis include the following:

- Only components located at or above grade are considered to be vulnerable to tornado missile damage.
- Components classified as seismic Category I or located within a structure designated as seismic Category I are not susceptible to damage from tornado missiles.

- Components not classified as seismic Category I or not located within a structure designated as seismic Category I are susceptible to damage from tornado missiles.

- While the seismic Category II components are designed to withstand the extreme winds associated with EF5 tornados and are designed to withstand EF4 tornado missiles, they are not designed to withstand EF5 tornado missiles.

The staff finds that this explanation adequately clarifies the issue. Therefore, RAI 19.1-167 and the associated open item are resolved.

The risk assessment takes credit for systems providing long-term heat removal from the core, but did not provide sufficient information on the structures that house these systems and their support systems. In particular, the staff was interested in aboveground outdoor tanks or other structures holding significant quantities of liquids, such as water or oil, that if failed or damaged could cause a flooding issue for other important equipment on site (e.g., pumps, transformers). The staff raised this issue in RAI 19.1-166. The staff was tracking RAI 19.1-166 as an open item in the SER with open items. In its response dated March 8, 2008, the applicant provided assurance that the ESBWR flooding analysis considered the potential for important equipment to be flooded by aboveground outdoor tanks or other fluid-holding structures. The staff finds this explanation sufficient. Therefore, RAI 19.1-166 and the associated open item are resolved.

19.1.5.4.4 Conclusion

Based on its review of the high-winds risk assessment, the staff finds the risk assessment to be technically adequate to support DC and the identification of risk insights. The extremely low absolute values estimated for the expected CDF from these events is indicative of the applicant's design and engineering efforts to reduce risk outliers and known limitations in former BWR designs.

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 staff believes the high-winds CDF can be added to the full-power internal events CDF.

Table 19.1-11

ESBWR Tornado Wind—PRA Predicted Structure Damage (Summary of Table 14.3-2 from the PRA Report, Revision 4)

Tornado Category	ESBWR Plant Structures[7]			
	SC I	SC II	RTNSS	NS
EF0	No Damage	No Damage	No Damage	No Damage
EF1	No Damage	No Damage	No Damage	No Damage
EF2	No Damage	No Damage	No Damage	Failure
EF3	No Damage	No Damage	No Damage	Failure
EF4	No Damage	No Damage	Failure	Failure
EF5	No Damage	Failure from Tornado Missiles	Failure	Failure

[7] The ESBWR plant structures are identified as seismic Category I (SC I), seismic Category II (SC II), regulatory treatment of non-safety systems (RTNSS), and nonseismic (NS).

Table 19.1-12

ESBWR Hurricane Wind—PRA Predicted Structure Damage (Summary of Table 14.3-1 from the PRA Report, Revision 4)

Hurricane Category	ESBWR Plant Structures[8]			
	SC I	SC II	RTNSS	NS[9]
Category 1	No Damage[10]	No Damage	No Damage	No Damage
Category 2	No Damage	No Damage	No Damage	No Damage
Category 3	No Damage/ LOPP	No Damage LOPP	No Damage/ LOPP	Failure
Category 4	No Damage/ LOPP	No Damage/ LOPP	No Damage/ LOPP	Failure
Category 5	No Damage/ LOPP	No Damage/ LOPP	No Damage/ LOPP	Failure

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

19.1.6.1 Results and Insights from Internal Events Low-Power and Shutdown Operations Probabilistic Risk Assessment

19.1.6.1.1 Summary of Technical Information

19.1.6.1.1.1 Methodology and Approach. This shutdown 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 degrees C (200 degrees F) while the plant is cooling and shutting down. For Mode 5, the reactor mode switch is in the shutdown position. Before entering Mode 5 from Mode 4, the heat removal requirements are transferred to the RWCU/SDC system. The main condenser and circulating

8 The ESBWR plant structures are identified as seismic Category I (SC I), seismic Category II (SC II), regulatory treatment of non-safety systems (RTNSS), and nonseismic (NS).

9 This column excludes the turbine building, service water building, and electrical building, which are assumed to be undamaged by either hurricanes or their missiles.

10 The applicant assumed that the only impact to the site from Category 1 and 2 hurricanes would be an LOPP with no additional equipment failures caused by the hurricane. The internal events PRA addresses these LOPP events, which have been included under the initiating events for LOPP.

water pumps are removed from service and use of the ICs is terminated. For the entire duration of Mode 5, all DHR is through the RWCU/SDC system. Mode 6 begins when one or more of the reactor vessel head closure bolts is less than fully tensioned.

The applicant assessed the following four plant operational states (POSs) during Modes 5 and 6: Mode 5, Mode 5 (Open), Mode 6 (Unflooded), and Mode 6 (Flooded), as previously defined in Section 16 of DCD, Tier 2, Revision 6. Fuel is in the reactor vessel during each of these POSs.

GEH did not quantitatively evaluate operation in Mode 4 (i.e., stable shutdown in which the RCS temperature is less than 215 degrees C (420 degrees F) and greater than 93.3 degrees C (200 degrees 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 the power-operating values.

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 critical safety functions essential to the shutdown model are DHR and inventory control. Containment is assumed to be open. The TS for Modes 5 and 6 do not require containment integrity. 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; therefore, ATWS is not an issue. DCD, Tier 2, Revision 6, addresses reactivity control during shutdown deterministically.

19.1.6.1.1.2 Significant Accident Sequences and Leading Contributors. The applicant estimated the mean ESBWR shutdown CDF from internal events to be 1.7×10^{-8}/yr. This is a very low CDF in comparison to CDF estimates for plants currently operating. This low value represents the applicant's effort to reduce or eliminate the contributors to core damage found in previous PRAs through improvements in plant design. However, areas of shutdown risk for which modeling is least complete or nonexistent (such as operator errors of commission and rare/new initiating events) could become important contributors to risk and cause the CDF for plant-specific implementations or the ESBWR design to be higher.

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.

The analysis assumed that all evaluated shutdown core damage events would 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 1.7×10^{-8}/yr.

Three initiating events that may occur during each of the four POSs (Mode 5, Mode 5 (open), Mode 6 (unflooded), and Mode 6 (flooded)) comprise over 80 percent of the ESBWR internal events shutdown CDF. These three initiating events include LOCAs in the RWCU/SDC lines below TAF, LOCAs in instrument lines below TAF, and RPV leaks and diversions caused by operator error. For LOCAs below TAF, manual closure of the LDW hatches is required to

prevent core damage, since it is necessary to flood the drywell and the vessel up to a level above TAF to ensure core cooling. For RPV leaks and diversions in which the GDCS fails to provide automatic injection to the RCS, the significant operator actions include isolating the RWCU system and providing low pressure makeup following RCS depressurization.

LOPPs from severe weather events, grid failures, or switchyard faults contribute another 10 percent to the total internal events shutdown CDF. Losses of both trains of the RWCU/SDC system contribute 8 percent to the total internal events CDF.

19.1.6.1.2 Acceptance Criteria

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used applicable guidance from SRP Section 19.0, Revision 2 in its review. In addition, 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, as stated in Section 19.1.1 of this report.

19.1.6.1.3 Staff Evaluation

19.1.6.1.3.1 Evaluation of Methodology and Approach. To evaluate GEH's decision to not quantitatively evaluate Mode 4, 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 de-inerted, 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. To mitigate a high wind event, the plant will go to Mode 4 and will not de-inert in Mode 4 when the plant shuts down in anticipation of a hurricane strike. This risk insight is captured in Table 19.2-3 of the ESBWR DCD. The staff finds this PRA modeling of Mode 4 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 earliest versions of the PRA did not define core damage. In RAI 19.1-96, the staff requested additional information documenting the success criteria used in the shutdown PRA. The staff tracked RAI 19.1-96 as an open item in the SER with open items. By letters dated September 17, 2007, and March 17, 2008, T-H uncertainty for short term and long term core cooling in Mode 5 in the ESBWR shutdown PRA was evaluated using MAAP 4.06. Consideration of these results led to changes in the shutdown event trees/success criteria. The applicant changed the shutdown event trees and success criteria to include the addition of depressurization using four DPVs in Mode 5 LOCAs, with the exception of LOCAs in the feedwater lines. The applicant also changed the PRA success criteria to require one GDCS injection line from each of the two GDCS pools and one GDCS equalizing line. The applicant also provided the results of MAAP 4.0.6 calculations for the loss of RWCU/SDC event to support the success criteria of needing two SRVs to implement low-pressure injection. The applicant also stated that the shutdown PRA core damage definition is consistent with RG 1.200. The staff found this RAI response and the associated changes to the shutdown event trees and success criteria, to be acceptable. Therefore, RAI 19.1-96 is resolved.

In RAI 5.4-59, the staff raised questions regarding the capability of the RWCU/SDC system to operate successfully during Modes 5 and 6. The staff requested that the normal vessel levels for RWCU/SDC operation in all modes, including Modes 5 and 6, be documented in DCD, Tier 2, Revision 6. The staff also requested calculations that show the temperatures and levels at which 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 how coolant from the RWCU/SDC system flows and mixes within the vessel and within the shroud. The staff was concerned that coolant from the RWCU/SDC system heat exchanger could bypass the core region and therefore not provide the cooling capacity predicted in the GEH model. The staff tracked RAI 5.4-59 as an open item in the SER with open items.

In its response, GEH updated Section 5.4.8.2.2 of DCD, Tier 2, Revision 6, regarding the need to maintain RPV water level sufficiently above the first stage water spillover point in the steam separators. To avoid a thermal stratification condition, the applicant expects that the plant will be operated with the RPV water level sufficiently above the minimum level assumed during use of the RWCU/SDC system. The applicant also updated DCD, Tier 2, Revision 6, to discuss the mixing between the incoming cooler shutdown water and the spillover water from the separators. In its response to the RAI, the applicant provided the results of a study of the relationship between the mixing factor and the RWCU/SDC flow rates. The applicant also stated that the RWCU/SDC pump flow and NRHX cooling capacity are designed to limit the temperature difference between the supply and return flows, thereby minimizing the potential for thermal cycling stress.

The NRC staff performed audit calculations of the RWCU/SDC flows using computational fluid dynamics (CFD) to assess the applicability of the GEH approach. The applicant completed CFD predictions for two sets of conditions to predict the flow and mixing of the RWCU/SDC fluid during Mode 5 (shutdown) conditions in the ESBWR. The predictions demonstrate that the cooling system flow mixes well with the overall natural circulation flow in the system. These simulations confirm the applicability of the complete mixing assumption in the GEH model. The two CFD predictions have different flow rates and temperature differences and indicate slightly different overall flow patterns. In both cases, however, the mixing is essentially complete. The staff considers the issue of adequate coolant mixing from the RWCU/SDC during shutdown operation resolved based on the information provided by GEH and the results of the staff's independent calculations. Therefore, RAI 5.4-59 is resolved.

The staff based its review of the ESBWR internal events shutdown PRA on the results reported in Section 16, "Shutdown Risk," of the PRA report, Revision 4. In response to the follow-up activities identified by the staff during an audit of the PRA (NRC, June 2009), GEH modified the RWCU/SDC BOC event trees in Revision 4 of the PRA to include an additional top event—four DPVs actuate before GDCS actuation. This modification makes the RWCU/SDC BOC trees consistent with the RPV leak and diversion event trees, which the applicant had previously modified. GEH also proposed a revision to Section 22 of the PRA report, entitled, "ESBWR PRA Changes," to include an evaluation of the modified trees by quantifying the new sequences that were generated. The new logic does not bring in any new changes to system models. The new sequences generated an additional CDF/LRF contribution of 0.012 percent of the baseline internal events shutdown CDF. The changes do not impact the shutdown external event models (fire, flood, and high-winds analyses) and their associated focused PRA evaluations. The new sequences have no impact on the shutdown external events models since the shutdown external events initiators do not follow the RWCU BOC sequences. The staff finds

the resolution of this issue (i.e., assessing the impact of the tree changes with limited amount of requantification) to be acceptable only for the purposes of identifying risk insights to support DC.

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 TS, 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). In light of these insights, the staff was concerned that 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 asked GEH to address this issue. The staff tracked RAI 19.1-149 as an open item in the SER with open items. In response, GEH added outage planning and control consistent with NUMARC 91-06 as a key risk insight in Table 19.2-3 of DCD, Tier 2, Revision 5. Therefore, RAI 19.1-149 is resolved.

As discussed previously, RWCU/SDC drainline breaks below TAF and instrument line breaks below TAF that may occur during each of the four POSs (Mode 5, Mode 5 (Open), Mode 6 (Unflooded), and Mode 6 (Flooded)) comprise a large fraction 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. 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. The staff noted that Revision 3 of the PRA report stated that the availability control (AC) was applicable in Mode 5 and 6 during operations with the potential for draining the reactor vessel. Since Revision 3 of the PRA report indicated that LOCAs involving pipe breaks contributed over 98 percent of the shutdown CDF, the staff believes that the applicability of this AC to Modes 5 and 6 should extend during the entire outage period. In response to RAI 19.1-123, Supplement 1, GEH changed AC 3.6.2, "Lower Drywell Hatches," described in DCD, Tier 2, Revision 4, Section 19A, to require applicability in Modes 5 and 6 during the entire outage period. Since the entire duration of Modes 5 and 6 is covered by AC 3.6.2, this update is acceptable.

However, the staff noted that Availability Control Surveillance Requirement (ACSR) 3.6.2.2 and ACSR 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, are inconsistent with NUMARC 91-06 guidance and operating experience. GEH responded that the intent of 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 the ACs are not intended to satisfy NUMARC 91-06 recommendations for preventing fission product release from containment. The staff believes that this guidance should have to satisfy the NUMARC 91-06 recommendations. 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. Containment closure is necessary to prevent fission product release from containment during severe accidents initiated by pipe breaks below TAF. The ACSR frequency of 30 days in the ESBWR PRA is most likely longer than the outage itself and may not provide closure of containment in sufficient to prevent a fission product release. In RAI 19.1-123, Supplement 2, the staff requested that GEH address this issue. The staff tracked RAI 19.1-123 as an open item in the SER with open items.

In response to RAI 19.1-123, Supplement 2, GEH responded that ACSR 3.6.2.2 and ACSR 3.6.2.3 augment ACSR 3.6.2.1, which requires verification every 12 hours that the LDW hatch administrative closure is in place. The administrative closure plan, as outlined in the ACLCO bases, provides for "administrative controls [that] assure trained personnel will be continuously located in the area of the doors and appropriate administrative controls are in place to communicate awareness of potential breaches and effect decisions to secure the hatches." The staff finds that this administrative control, verified to be in place every 12 hours, satisfies the intent of the NUMARC 91-06 guideline that states, "[a] procedure should be established to assure that closure can be accomplished in a time commensurate with plant conditions," and recognizes that conditions change during the outage. Therefore, RAI 19.1-123 is resolved.

ACSR 3.6.2.2 and ACSR 3.6.2.3 require verification that the equipment hatch and airlock can be secured in place. The component capability to be secured in place is not expected to be compromised at any point in time, and the continuous attention of trained personnel provides adequate assurance of the continued capability. A 30-day periodic reverification constitutes an additional formal documented assurance of what is otherwise continuously verified. The staff finds these ACSRs to be an acceptable means of addressing the intent of NUMARC 91-06.

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 -6,400-millimeter (-21-foot) level in the RB, and manually close the equipment hatch and the personnel airlock. 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 19.1.0-4[11], Supplement 1, the staff requested GEH to address this issue. RAI 19.1.0-4, Supplement 1, contained Parts A through F. The staff tracked RAI 19.1.0-4, Supplement 1, Parts A through F as open items in the SER with open items. Closure of RAI 19.1.0-4, Supplement 1, Parts A through F, and the closure of the associated open items are discussed below.

In response to RAI 19.1.0-4, Supplement 1, Part A, GEH stated that at least 90 minutes will be available to detect, diagnose, and close the hatches. Thus, GEH maintained that outage personnel do not need to be located in the area of the doors continuously. However, GEH added, "closure of both the equipment hatch and the personnel hatch can be done from outside the lower drywell/containment" as a key risk insight in Table 19.2-3 in DCD, Tier 2, Revision 5. As previously discussed, in response to RAI 19.1-123, Supplement 2, GEH noted that ACSR 3.6.2.2 and ACSR 3.6.2.3 augment ACSR 3.6.2.1, which requires verification every 12

[11] RAI 19.1.0-4 was referred to as RAI 19.1-4 in the SER with open items.

hours that the LDW hatch administrative closure plan is in place. The administrative closure plan (as outlined in the ACLCO bases) provides for "administrative controls [that] assure trained personnel will be located in the area of the doors continuously and appropriate administrative controls are in place to communicate awareness of potential breaches and effect decisions to secure the hatches." The staff finds this approach to be reasonable. Since the ACLO bases discusses controls that assure trained personnel will be located in the area of the doors continuously. Therefore, RAI 19.1.0-4, Supplement 1, Part A, is resolved.

To mitigate losses of RWCU/SDC, RPV leaks, and 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 models the success of the GDCS, assuming that all eight DPVs will be operable and will automatically open. During the initial review of the ESBWR TS, the staff found that there was no requirement for the DPVs to automatically open, and there was no requirement for the DPVs or the SRVs to be operable in Modes 5 and 6. Instead, there was only a TS surveillance requirement to have a proper vent path for GDCS operability. The TS did not specify the size of this vent path or the number of valves. The staff was concerned that the complex task of determining an adequate size vent path was being left as an operational activity to be completed without the support of engineering analysis.

Furthermore, Revision 3 of the PRA did not model the failure of the operator to determine the adequate RCS vent path size (number of DPVs that need to be opened) to support GDCS operation. This operator error cannot be modeled with conventional HRA methodologies. The staff stated these issues in RAIs 19.1-93, 19.1-94, 19.1-95, 19.1-96 Supplement 1, and 19.1-143 and tracked them as open items in the SER with open items. In response to the staff's RAIs on these issues (i.e., RAIs 19.1-93, 19.1-94, 19.1-95, 19.1-96 Supplement 1, and 19.1-143), GEH updated the TS in DCD, Tier 2, Revision 5, for the GDCS to require six out of eight DPVs to be operable for automatic actuation until the reactor head is removed. Therefore, RAIs 19.1-93, 19.1-94, 19.1-95, and 19.1-143 is resolved.

In RAI 19.1.0-6 Supplement 2, the staff raised the concern that if one of the four DPVs fails to open, then the GDCS function fails. The staff requested GEH to perform and document a sensitivity study assuming only four DPVs were available and operable for GDCS. In their response to RAI 19.1-96 Supplement 2, Technical Specifications were updated to require six out of eight DPV valves to be operable until the vessel head is removed. Additionally, GEH provided a sensitivity study showing the shutdown PRA results with varying DPV requirements. The staff reviewed the updated Technical Specifications based on the results of the sensitivity analyses. Based on the results of the applicant's sensitivity study and the modified technical specifications. Therefore, RAI 19.1-96 is resolved.

As shown in Table 16.6.3 of the ESBWR shutdown PRA, the top 12 dominant cutsets contribute over 64 percent of the risk. These cutsets initiate by a LOCA below TAF or an operator-induced leak or diversion in each of the four POSs—Mode 5, Mode 5 (Open), Mode 6 (Unflooded), and Mode 6 (Flooded).

In Section 19.2.4.2 of DCD, Tier 2, Revision 6, 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 Section 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 the following on page 4-3:

> The results indicate that [for] source terms which involve a release of about 10 percent 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 staff believes that the consequences of a shutdown severe accident occurring during Modes 5 and 6 approximately 8 days or less after shutdown are not negligible. The staff tracked RAI 19.1-159 as an open item in the SER with open items. In response to RAI 19.1-159, GEH revised Section 19.2.4.3 of DCD, Tier 2, Revision 6, to state the following:

> The source terms for containment bypass events may not fall below the early fatality threshold until approximately 8 days after shutdown; however, the frequency of shutdown containment bypass events is very low. As a result the offsite consequences, which are the product of the source term risk and the shutdown containment bypass frequency, are not significant.

Since this DCD modification is consistent with NUREG/CR- 6595, the staff found this DCD change to be acceptable. Therefore, RAI 19.1-159 is resolved.

19.1.6.1.3.2 Evaluation of 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 addressed below by initiating event category.

19.1.6.1.3.2.1 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 (RHR) 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 drainpaths.

Although the RWCU/SDC system design has been improved to reduce the number of potential RPV drain pathways, it still has the potential to drain the RPV during Modes 5 and 6. The system is connected to the RPV during shutdown and 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). These risk-significant TS protect against postulated breaks in the RWCU/SDC system outside containment.

BOC can originate only in RWCU/SDC system piping because this is the only system that removes reactor coolant from the containment in Mode 6. The rest of the RPV piping is isolated. The RWCU/SDC system containment penetrations have redundant and automatic power-operated containment isolation valves that close on signals from the leak detection and isolation system.

An additional, diverse non-safety isolation of the RWCU/SDC system provides protection in the event of a break outside containment. This additional, diverse non-safety isolation signal of the RWCU/SDC system protects the system in Modes 1 through 4, but is not required by the TS for shutdown modes. This signal, provided by the DPS, is not credited during Modes 5 and 6 in the shutdown PRA. The staff raised this concern in RAI 19.1-178. In response to RAI 19.1-178, concerning the omission of TS for DPS in Mode 5, GEH updated the PRA model in Revision 4 of the PRA report to include RWCU/SDC BOCs. At the PRA audit, the staff reviewed the associated event trees and found that they had logic errors. The licensee evaluated the impact of these logic errors in Section 22.16 of the PRA, changes to the shutdown PRA model. In Section 22.16 of the PRA, the event trees were modified to include a top event for four DPVS actuating prior to GDCS actuation. The risk contribution from the new sequences was also quantified. The staff concluded that the additional core damage sequences have a negligible impact on the baseline results. Based on the information provided in Section 22.16 of the PRA, the applicant's response, RAI 19.1-178 is resolved.

To minimize the use of freeze seals, maintenance valves are installed on power-operated equipment and valves on lines attached to the RPV that require maintenance. Because these maintenance valves facilitate maintenance on power-operated equipment and valves on lines attached to the RPV, freeze seals will not be required (see DCD, Tier 2, Revision 6, Section 5.2.3.1.1).

Regarding penetrations in the vessel bottom head upstream of the RWCU/SDC isolation valves, GEH did not quantitatively evaluate operator-induced loss of reactor vessel inventory in Revision 3 of the PRA. In RAI 19.1.0-4, Supplement 1, Part E, the staff asked GEH to address this issue by adding key risk insights to the DCD. In response, GEH discussed the ESBWR design requirements that preclude the need for freeze seals. To minimize the use of freeze seals, maintenance valves are installed on power-operated equipment and valves on lines attached to the RPV that require maintenance. Because these maintenance valves facilitate maintenance on power-operated equipment and valves on lines attached to the RPV, freeze seals will not be required (see DCD, Tier 2, Revision 6, Section 5.2.3.1.1). This is acceptable. However, GEH did not fully address piping penetrations in the vessel bottom head upstream of the isolation valves.

In RAI 19.1.0-4, Supplement 2, Part E the staff requested that GEH (1) provide information about the sizes of these piping penetrations and associated alarm or position indication in the control room or (2) model operator-induced leaks using operating data in the shutdown PRA. In response to RAI 19.1.0-4, Supplement 2, Part E GEH evaluated operator-induced leaks in Revision 4 of the PRA. The staff reviewed the associated event trees, cutsets, and risk insights and found them to be acceptable. Therefore, RAI 19.1.0-4, Supplement 1s 1 and 2, Part E, is resolved.

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, which are 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 RB. The FAPCS and

CRDS connections are downstream of the two check valves. A postulated break in the RWCU/SDC piping system inside the RB, which would otherwise allow reactor coolant to flow backwards through the main feedwater lines and spill into the RB, will be isolated by either the redundant RWCU/SDC check valves or the feedwater check valves, even assuming a single failure of one check valve.

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 the bottom of 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, GEH stated that the COL applicant will develop contingency procedures to provide core and spent fuel cooling capability and mitigation actions during CRD replacement with fuel in the vessel. However, the staff noted that GEH did not capture these risk insights in Table 19.2-3 of DCD, Tier 2, Revision 4. In RAI 19.1.0-4, Supplement 1, Part B, the staff requested that GEH address this issue. The staff was tracking RAI 19.1.0-4 as an open item in the SER with open items. In response to RAI 19.1.0-4, Supplement 1, Part B, GEH added these assumptions to Table 19.2-3 of DCD, Tier 2, Revision 5. Therefore, RAI 19.1.0-4, Supplement 1, Part B, is resolved.

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

19.1.6.1.3.2.2 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, assumed 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 19.1.0-4, Supplement 1, Part C, the staff requested that GEH address this issue. In

response to a different RAI, RAI 19.2-121, Supplement 1, GEH updated Table 19.2-3 of DCD,Tier 2, Revision 6, to state that, "during shutdown conditions, in preparation for refueling, both trains of RWCU/SDC are running while the unit is in either Mode 5 of Mode 6 until the reactor cavity is flooded." The staff considers this issue to be resolved because the applicant has documented in the DCD that two trains would be running. Therefore, RAI 19.2-121 and RAI 19.1.0-4, Supplement 1, Part C, is resolved.

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 ICs, which offer an alternative, automated, passive, core-cooling path not available in current operating BWRs, can cool the ESBWR.

TS 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. The staff tracked RAI 19.1-144 as an open item in the SER. In response to RAI 19.1-144, Supplement 4, GEH responded that the ESBWR has an RPV head vent system that handles any noncondensable gas buildup that could inhibit natural circulation core cooling. The piping is 2 inches in diameter. After the plant reaches cold shutdown, the two valves in the vent piping leading to the equipment and floor drain sump are opened and the valve in the piping connected to the main steamline is closed. GEH stated in the proposed revision to PRA that the head vent should not impact the ICS operability because the isolation of this line is considered very likely. Based on T-H analyses, the operator has 32 hours to close the head vent if the ICS is started manually without credit for CRD and 14.5 hours if the ICS starts automatically without credit for CRD flow. The operators in the MCR can diagnose an open head vent line because the isolation valves leading to the equipment and floor drain sump have open and closed indication and down steam temperature indication in the MCR. Based on this update to the PRA, the staff considers RAI 19.1.144 to be resolved.

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. Section 19.1.6.1.3.1 of this report discusses adequate venting for the GDCS during shutdown

19.1.6.1.3.3 Evaluation of Insights from Uncertainty and Importance Analyses. The staff used the results of the applicant's importance analyses to identify (1) SSCs or human actions or both whose reported reliability contribute most to achieving the low reported shutdown CDF (RAW),

and (2) SSCs or human actions or both whose reported reliability would 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 and 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. Revision 3 of the PRA did not evaluate BOCs, which were therefore excluded from 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.

In Revision 3 of the PRA report, GEH stated that 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 19.1.0-4, Supplement 1, Part D, the staff requested that GEH address this issue. RAI 19.1.0-4, Supplement 1, Part D, was tracked as an open item in the SER with open items. In response to RAI 19.1.0-4, Supplement 1, Part D, GEH updated the PRA model in Revision 4 to model operator-induced leaks and RPV diversions. The staff found the information added to the PRA report to be acceptable. Therefore, RAI 19.1.0-4, Supplement 1, Part D, is resolved.

The staff noted that TS were omitted for DPS in Mode 5. The staff raised this concern in RAI 19.1-178. In response to RAI 19.1-178, GEH updated the PRA model in Revision 4 to include RWCU/SDC BOCs. At the PRA audit, the staff reviewed the associated event trees and found that they had logic errors. The licensee evaluated the impact of these logic errors in Section 22.16 of the PRA, Changes to the Shutdown PRA Model. In Section 22.16 of the PRA, the event trees were modified to include a top event for four DPVS actuating prior to GDCS actuation. The risk contribution from the new sequences was also quantified. The staff concluded that the additional core damage sequences have a negligible impact on the baseline results. Based on the information provided in Section 22.16 of the PRA, the applicant's response, RAI 19.1-178 is resolved.

Based on the addition of RPV leaks and diversions and BOCs to Revision 4 of the shutdown PRA, the risk achievement analyses yielded additional risk insights. LOCAs below TAF in each of the four POSs have the highest RAW values, exceeding 5×10^5. LOCAs below TAF comprise 50 percent of the internal shutdown CDF/LRF. To prevent core damage, the operator must close the drywell hatch.

Events having RAW values exceeding 1×10^3 include the following:

C63-CCFSOFTWARE	Common-cause failure of software, which represents the failure of the entire safety-related Q-DCIS platform to actuate all supported functions, including manual actuations
%M6U_RWCU_BOC	LOCAs involving RWCU BOC in Mode 6 (Unflooded)
%M5_LOCA_OT	LOCAs in lines other than feedwater or GDCS in Mode 5
%M5O_LOCA_OT	LOCAs in lines other than feedwater or GDCS in Mode 5 (Open)
%M5_ LOCA_ FW	LOCA in feedwater line—Mode 5
%M6U_LOCA_FW	LOCA in feedwater line—Mode 6 (Unflooded)
%M5_LOCA-G	LOCA in GDCS—Mode 5
%M5 LOCA-FW	LOCA in feedwater—Mode 5 (Open)

19.1.6.3.4 Evaluation of Insights from Sensitivity Studies

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 unavailability, 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.

19.1.6.1.3.4.1 <u>Focused Probabilistic Risk Assessment Sensitivity</u>. 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 some form of regulatory treatment:

- 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, RCCWS, TCCWS, plant air, nitrogen, PSWS, FMCRD groups, and PIP buses A3 and B3. To perform the focused and RTNSS sensitivity studies for shutdown internal events fires, floods, and high winds, the applicant generated two flag files: (1) fail all non-safety systems and (2) fail all non-safety systems except those systems designated as RTNSS.

PRA report Tables 11.3-36, 11.3-37, 11.3-38, and 11.3-39 show the results of the focused PRA analyses and the RTNSS PRA analyses for shutdown internal events, fire, floods, and high winds. The focused internal events shutdown sensitivity analysis generated a CDF of 1.69×10^{-6}/yr, and the RTNSS generated a CDF of 4.41×10^{-7}/yr. Based on the CDF results for the shutdown focus sensitivities, the NRC goal of 1×10^{-4}/yr CDF is met for both the shutdown focus and RTNSS sensitivities. Since all shutdown CDF sequences are assumed to be direct LRF contributors, the LRF goal of 1×10^{-6}/yr is applicable as well. The RTNSS LRF meets the threshold, but the shutdown focus exceeds the LRF threshold. The difference in CDF showed a decrease of about a factor of 4. A review of risk-significant events from the RTNSS shutdown results highlights the importance of the FPS/FAPCS injection pathway.

The focused shutdown fire study generated a CDF of 2.87×10^{-6}/yr. The RTNSS study generated a CDF of 3.91×10^{-7}/yr. Based on the CDF results for the shutdown fire focused sensitivity analysis, the NRC goal of 1×10^{-4}/yr CDF is met for both the baseline fire and RTNSS scenarios. Since all shutdown CDF sequences are assumed to be direct LRF contributors, the LRF goal of 1×10^{-6}/yr is met for the RTNSS case, but exceeded in the case of the focused shutdown fire. The RTNSS results show a risk reduction of 88 percent as compared to the results of the focused study. Similar to the shutdown internal events RTNSS results, the focused fire study shows that the FPS/FAPCS injection pathway is risk significant.

The focused shutdown flood study generated a CDF of 6.35×10^{-7}/yr and the RTNSS study generated a CDF of 2.81×10^{-7}/yr. Based on the shutdown flood focused sensitivity study, the NRC goals of 1×10^{-4} CDF and 1×10^{-6}/yr LRF are met. The RTNSS results show a risk reduction of approximately 56 percent as compared to the focused results. Similar to the shutdown internal events RTNSS results, the focused flood study shows that the FPS/FAPCS injection pathway is risk significant.

The focused shutdown high-winds study generated a CDF of 1.20×10^{-6}/yr for tornados and hurricanes, and the RTNSS study generated a CDF of 1.71×10^{-7}/yr for tornados and hurricanes. The results for the focused high-winds sensitivity showed significant impact to CDF, with the failure of non-safety systems in both the RTNSS and focused cases. The RTNSS results indicate a CDF reduction of approximately 86 percent as compared to the focused case. Similar to the shutdown internal events RTNSS results, the focused high-winds study shows that the FPS/FAPCS injection pathway is risk significant.

19.1.6.1.3.4.2 Loss-of-Coolant Accident Frequency Sensitivity. 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 basis for the reduction. This sensitivity case shows the following CDF/LRF results with no reduction factor applied.

- baseline results = 1.63×10^{-8}/yr
- sensitivity results = 9.42×10^{-8}/yr

The CDF/LRF for the sensitivity increases by a factor of 8 as compared to the baseline results since LOCAs constitute 50 percent of the baseline results. Thus, the shutdown PRA results depend on the LOCA frequencies and how they are determined. However, without the reduction factor, the ESBWR shutdown CDF results are still below the NRC safety goals.

19.1.6.1.3.4.3 Lower Drywell Hatch Sensitivity. RWCU/SDC drainline breaks below TAF and instrument line breaks below TAF that may occur during all four POSs comprise about 50 percent of the ESBWR internal event shutdown CDF/LRF. 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 6 hours 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 6 hours 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 ran a sensitivity case applying a 50-percent failure rate for both hatch closure events. The resulting CDF/LRF is 3.48×10^{-7}/yr. The resulting ESBWR shutdown CDF and LRF increased by almost a factor of 20, indicating that the operator's ability to reliably close the drywell hatches is risk significant.

GEH also 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 are approximately 26 percent of the baseline value of 1.25×10^{-8} /yr.

19.1.6.1.3.4.4 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 two operator action sensitivity cases discussed below.

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 hatches following a LOCA was also set to TRUE (failed).

The resulting CDF/LRF for Case 1 is $5.76\times^{-6}$/yr. Case 1 results show an increase of more than two 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 one order of magnitude lower than most modeled human actions. It shows how the CDF could be affected if credit is taken for very effective operator response to transients.

The resulting CDF for Case 2 is 1.14×10^{-9}/yr. Case 2 results in a decrease in CDF of approximately one 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 remain 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.

19.1.6.1.3.4.5 <u>Reactor Pressure Vessel Draindown Initiating Event Frequency Sensitivity</u>. The initiating event frequency for RPV draindown events used in the ESBWR shutdown PRA analysis is lower than the initiating event value developed in EPRI 1003113. The use of the lower frequency is based on design improvement of the ESBWR RWCU system compared to current BWR RHR systems, as well as a review of the RWCU piping and instrumentation drawings. This sensitivity case shows the following results with the initiating event frequency developed by EPRI (i.e., 2.80×10^{-5} per hour) applied to the shutdown PRA sequences:

Baseline Results = 1.63×10^{-8}/yr (with a truncation limit of 1×10^{-14}/yr)
Sensitivity Results = 2.58×10^{-8}/yr

The shutdown CDF increases when using the initiating event frequency developed by EPRI for RPV draindown events. The CDF increases by nearly 58 percent over the baseline shutdown case. RPV leaks account for about 30 percent of the CDF in the base case. These leaks represent by far the largest contribution after LOCA events. In this sensitivity case, CDF contribution from draindown events increases to 56 percent. Although there is a notable increase in CDF with the EPRI value, the results are still within the NRC stated goals for CDF and LRF.

GEH stated that the basis for the reduced frequency of a draindown event is reasonable. The RWCU system has very few possible leak or diversion paths (non-LOCA) and almost all are small lines under 50.8 millimeters (2 inches). A review of the incidents used to develop the EPRI value showed that very few of the events are relevant to the ESBWR design.

GEH also mentioned that the risk associated with draindown events (especially in Mode 6) may be overestimated. The cases with the largest contribution are generally Mode 6 (Unflooded) cases in which the RWCU system is isolated, but no alternate DHR source is successful. These cases account for about 21 percent of the overall baseline CDF. In these cases, no credit is taken for water in the pools above the vessel or fire pump hoses in the RB. In these cases, the

vessel head and the containment head are removed, and the RPV is open to the refueling floor of the RB. Makeup to the reactor well/RPV would be available from one or several of the pools in the RB. This would require operators opening a valve to cross-connect the pools to the reactor well. FPS pumps pumping to these pools, or a FPS pump truck connected to the building, could also provide additional water to the vessel in these scenarios. The analysis took no credit for these potential inventory sources.

19.1.6.1.4 Conclusion

Based on the discussions above, the staff concludes that the methodology and approach of the internal events low-power and shutdown risk analysis is technically adequate to identify risk insights to support DC. However, the applicant modified the RWCU/SDC BOC event trees to include an additional top event involving the actuation of four DPVs before GDCS actuation, This modification makes the RWCU/SDC BOC trees consistent with the RPV leak and diversion event trees, which the applicant modified in Revision 4 of the PRA. GEH submitted a proposed revision to Section 22 of the PRA report evaluating the modified trees by quantifying the new sequences that were generated. The new logic does not bring in any new system model changes. The new sequences generated an additional CDF/LRF contribution of 0.012 percent of the baseline internal events shutdown CDF. Since the shutdown external events initiators do not follow the RWCU BOC sequences, the new sequences added have no impact on the shutdown external events models. The staff finds the resolution of this issue to be acceptable for the purposes of identifying risk insights to support DC.

19.1.6.2 Results and Insights from External Events Low-Power and Shutdown Operations Probabilistic Risk Assessment

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 Results and Insights from the Low-Power and Shutdown Fire Risk Assessment

19.1.6.2.1.1 Summary of Technical Information.

19.1.6.2.1.1.1 Methodology and Approach. Based on the Level 1 internal events shutdown PRA, GEH performed a quantitative fire risk assessment.

The applicant performed the ESBWR full-power fire assessment according to the guidance in NUREG/CR-6850. The guidance in NUREG/CR-6850 is not applicable to qualitative screening for shutdown conditions. Therefore, the GEH performed the screening for the shutdown fire model assuming that the postulated fire has to result in one of the initiating events defined in the shutdown model. The critical safety functions essential to the shutdown model are DHR and inventory control. The applicant assumed that reactivity control and SFP cooling would 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. Similar to the

internal events shutdown PRA, 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.

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 also help to restore the fire barrier in time to prevent fire propagation.

19.1.6.2.1.1.2 Shutdown Fire Risk Significant Core Damage Scenarios and Dominant Contributors. This section describes the top two sequences contributing over 90 percent of the shutdown fire risk, as reported by GEH. Shutdown Fire Scenario 1, contributing approximately 51 percent, is initiated by a postulated fire in the TB general area (F4197 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. Other systems failed by a postulated fire in F4197 include the condensate and feedwater system, TCCWS, service air system, and UPS buses in the TB and other places. These failures make fire area F4197 a significant risk contributor to the shutdown fire risk.

The cabling for the RCCWS and PSWS is assumed not to be failed by a fire in F4197 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 42 percent of the CDF, is initiated by a postulated fire in the switchyard (fire area F7300). A fire in the switchyard is conservatively assumed to result in loss of DHR. The transfer from the offsite power to diesel generators is assumed not to be fast enough to prevent the failure of the RWCU system. The analysis assumes no recovery of offsite power.

19.1.6.2.1.1.3 Risk-Significant Function/Design Feature, Phenomena/Challenges and Human Actions for the 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, Revision 4.

19.1.6.2.1.2 Acceptance Criteria.

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used applicable guidance from SRP Section 19.0, Revision 2, in its review.

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

19.1.6.2.1.3 Staff Evaluation. GEH assumed the probability of a fire barrier failure to be 7.4×10^{-3}. 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 applicant will choose between roving and continuous fire watches for barriers of increased risk significance. The staff tracked RAI 19.1-126 as an open item in the SER with open items.

In Revision 2 of the PRA, GEH performed sensitivity studies to evaluate the risk impact of fire barrier failure associated with a fire watch. GEH analyzed the following two cases of fire barriers for the shutdown fire PRA:

1) Only one fire barrier exists on the fire propagation path where the fire barrier is a fire door.

2) Only one fire barrier exists on the fire propagation path where the fire barriers are walls or sealed penetrations. Multiple barriers in series exist on the fire propagation path.

In response to RAI 19.1-126, Supplement 1, GEH updated Revision 3 of the PRA report and provided the following risk insights in Table 19.2-3 of DCD, Tier 2, Revision 5:

> During shutdown conditions, a continuous fire watch is required for the following scenarios with breached fire barriers for maintenance activities:
>
> - The breaching of the fire door between fire areas F1152 and F1162 (the RB fire areas that house RWCU pumps).
> - The simultaneous breach of the multiple fire barriers that can open fire areas F3301 and F3302 (the N-DCIS room fire areas) to the fire area F3100 (the corridor fire area) at the same time.

The risk insights added to Table 19.2-3 of the DCD regarding risk signficiant fire barriers was acceptable to resolve the staff's concern. Therefore, RAI 19.1-126 is resolved.

In Revision 4 of the PRA report, with the changes in fire area designations, GEH updated the shutdown fire barrier sensitivity studies as reported in Tables 11.3-49 and 11.3-50 of the PRA. GEH then updated Table 19.2-3 in DCD, Tier 2, Revision 6, to include continuous fire watches for additional fire areas as follows:

> During shutdown conditions, a continuous fire watch is required for the following scenarios with breached fire barriers for maintenance activities:

- The breaching of the fire doors between fire areas F1152 and F1162 (the RB fire areas that house RWCU pumps) and between fire areas F4250 and F4260 (the TB fire areas that house the RCCW pumps).

- The simultaneously breaching of the multiple fire barriers that can open fire areas F3301 and F3302 (the N-DCIS room fire areas) to fire area F3100 (the corridor fire area) at the same time.

- The simultaneously breaching of the multiple fire barriers that can open fire areas F5350 and F5360 (the PIP electric equipment room fire areas) to fire area F5100 (the corridor fire area) at the same time.

Based on the GEH updates to the PRA and to Chapter 19 of DCD, Tier 2, Revision 6, the staff's concerns associated with the identification of risk significant fire barriers to be resolved.

Since NUREG/CR-6850 excludes low power/shutdown operations, the applicant calculated the shutdown fire ignition frequencies using a different method. The estimation of fire ignition frequencies for shutdown conditions is performed using the information provided in the RES/OERAB/S02-01, "Fire Events—Update of U.S. Operating Experience, 1986–1999." This document expands and updates the information of AEOD/S97-03, "Special Study: Fire Events—Feedback of U.S. Operating Experience," issued June 1997. RES/OERAB/S02-01 summarizes information on fire events that occurred during power operation and during shutdown conditions and estimates fire frequencies in both power and shutdown operation for different types of buildings and locations.

To compare the shutdown fire risk with the full-power fire risk, the shutdown fire initiating event frequencies are converted from shutdown year to calendar year. Table 12.7-7 in the PRA Report, Revision 4, calculates the conversion factors for each mode by assuming a 2-year refueling cycle and an outage duration of 548 hours. Therefore, the shutdown fire initiating event frequency calculations assume one-half shutdown per year (274 hours). The total CDF for all shutdown fire scenarios is 9.56×10^{-9}/yr.

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 AC 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. The staff tracked RAI 19.1-129 as an open item in the SER with open items. In response to RAI 19.1-129, GEH stated that the ESBWR MCR controls are connected to the back panel rooms via fiber optic cables, which are unaffected by a postulated 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. GEH also added that it had evaluated fires in the back panel rooms separately and considered their impact on the operability of automatic systems. In addition, Table 19.2-3 of DCD, Tier 2, Revision 6, states that the communication links between the MCR and the Q-DCIS and N-DCIS rooms do not include any copper or other wire conductors that could potentially cause fire-induced spurious actuations that could adversely affect safe shutdown. Based on these updates to the DCD, the staff 's concerns regarding a postulated MCR fire and its impact on safety and non-safety-related equipment are resolved. Therefore, RAI 19.1-129 is resolved.

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 the RWCU/SDC system. The RWCU system inboard containment isolation valves are located in the LDW, which could, according to GEH, be well separated spatially. 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 the RWCU/SDC system 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 19.1.0-4, Supplement 1, Part F, the staff requested that GEH address this issue.

In response to RAI 19.1.0-4, Supplement 1, Part F, GEH stated that drywell/containment fires that could result in loss of the RWCU/SDC and the RWCU inboard containment isolation valves were screened as not significant based on spatial separation of these valves. GEH also stated that this is a level of detail that is consistent with many design features that, although important, are not expected to change and are not considered to be significant assumptions. This response did not address the staff's concern. In RAI 19.1.0-4 Supplement 2, Part F, the staff requested GEH to document as a key risk insight that drywell/containment fires that could result in loss of the RWCU/SDC and the RWCU inboard containment isolation valves were screened based on spatial separation of the RWCU, or to assess and quantify drywell/containment fires that could result in loss of the RWCU/SDC and the RWCU inboard containment isolation valves in the ESBWR fire PRA.

In response to RAI 19.1.0-4, Supplement 2, Part F, GEH added the screening of LDW fires (ones that can impact the RWCU/SDC system and the RWCU isolation valves) to DCD, Tier 2, Revision 6, Table 19.2-3, as a key risk insight based on physical separation of the components, the limited number of ignition sources in the area, and the limited combustible material in the area. Therefore, RAI 19.1.0-4, Supplements 1 and 2, Part F is resolved.

Based on the staff's review of the risk achievement results, the shutdown fire PRA results are not as sensitive to operator errors as CCFs in the following systems: GDCS, ADS, Q-DCIS, the UPS, and the dc power supply system (DCS). The CCFs in these systems have the highest RAW values. For example, the Q-DCIS system, UPS, and DCS have RAW values exceeding 1,000. In contrast, failure of the operator to recognize the need for low pressure makeup after depressurization has a RAW value of approximately 26. Failure of the operator to open two DPVs manually has a RAW value of approximately 8. Failure of the operator to actuate the FPS in LPCI mode has a RAW value of approximately 7.

19.1.6.2.1.4 Conclusion. The staff has reviewed the GEH shutdown fire risk assessment and found to it be technically adequate to support DC and the identification of risk insights.

19.1.6.2.2 Results and Insights from the Low-Power and Shutdown Internal Flooding Risk Assessment

19.1.6.2.2.1 Summary of Technical Information. As in the full-power assessment, the applicant performed the shutdown internal flooding analysis using equipment locations based on existing plant layout drawings. Also similar to the full-power assessment, the applicant divided buildings into flood zones based on separation for flooding. GEH screened those flood zones that do not contain flood sources or PRA equipment 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.

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 that contribute approximately 56 percent of the shutdown flooding CDF of 5.2×10^{-9}/yr.

Flooding Sequence 1, contributing about 24 percent, is initiated by a break in the makeup water system in RB elevation 17,500 millimeters (Flood Zone RB3-P30-L-M5, M5O, and M6U).

Flooding Sequence 2, contributing about 16 percent, is initiated by a flood caused by a service water line break in the service water building (Flood Zones SF-P41A_S_SD and SF-P41B_S_SD), which impacts the PSWS.

Flood Sequence 3, contributing about 10 percent, is initiated by a break in the FPS in the TB elevation 4,650 millimeters (Flood Zone TB-U43-L-M5, M5O, and M6U).

Flood Sequence 4, contributing about 6 percent, is initiated by a flood in the TB main condenser (Flood Zone TBC-B21A-S-M5, M5O, and M6U), which impacts the RWCU/SDC system.

19.1.6.2.2.2 Acceptance Criteria.

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used applicable guidance from SRP Section 19.0, Revision 2, in its review.

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

19.1.6.2.2.3 Staff Evaluation. 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. The staff considers this to be an acceptable approach. As in the full-power assessment, the applicant estimated the initiating event frequency for each flood zone 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 Assessment of Oconee, Unit 3," issued July 1984, for the expansion joint failure data. The staff considers these to be appropriate data sources.

GEH estimated the shutdown CDF for each flood damage state by quantifying the loss of RWCU/SDC for three POSs: 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 staff agrees that Mode 6 (flooded) need not be considered because of the abundant cooling capability when the vessel is flooded.

Based on the staff review of the risk achievement results, the shutdown flooding PRA results are not as sensitive to operator errors as CCFs in the GDCS, ADS, and Q-DCIS equipment. These CCFs have the highest RAWs, which are greater than 100. In contrast, two operator actions, failure of the operator to actuate the FPS in LPCI mode and failure of the operator to recognize the need for low-pressure makeup after depressurization, have RAW values of approximately 11 and 18, respectively.

19.1.6.2.2.4 Conclusion. The staff has reviewed GEH's shutdown flooding risk assessment and found to it be technically adequate to support DC and the identification of risk insights.

19.1.6.2.3 Results and Insights from the Low-Power and Shutdown Internal High-Winds Risk Assessment

19.1.6.2.3.1 Summary of Technical Information. As in the full-power assessment, GEH performed the following major steps to complete the high-winds 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 applicant calculated the tornado strike initiating event frequency using the methodology provided in NUREG/CR-4461. To ensure a bounding analysis, tornado strike initiating frequencies that encompass most sites are generated using data from the central region of the United States where the tornado intensities and frequencies of occurrence are highest. In addition, the analysis assumed an ESBWR characteristic length of 400 feet, which represents a value double the assumed characteristic length used in NUREG/CR-4461. This results in the doubling of the strike probabilities for finite structures. GEH then estimated the shutdown tornado frequencies 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.

GEH assumed that the risk associated with LOPP due to a tornado strike when the reactor well is flooded (Mode 6 (Flooded)) is negligible because of a large quantity of water that is passively available to provide cooling for a time period in excess of 24 hours. This time period allows for an adequate path from an external water source to the reactor well to be established. Equipment and systems, such as CRD pumps, FAPCS pumps, RWCU/SDC pumps, and firewater pumps, are housed in seismic Category I structures and would be available to provide an adequate cooling pathway when powered from onsite power. For this reason, the ESBWR high-winds analysis did not consider the shutdown analysis for Mode 6 (Flooded) operations. The staff accepts this approach.

As in the full-power analysis, GEH obtained the LOPP data used to determine the strike frequency associated with hurricane events from NUREG/CR-6890. The applicant collected a subset of the coastal plant data for plants located on shorelines and in areas with high return rates for hurricanes. These data were limited to plants located in Florida, Louisiana, and North Carolina. To calculate a bounding hurricane strike frequency, a total operating duration for shoreline plants is 58.51 rcy and 5.49 reactor shutdown-years for a total of 64 rcy. The resulting

hurricane initiating event frequency is roughly 5 times the frequency for all coastal plants. The ESBWR high-winds risk analysis used a hurricane strike frequency of 7.60×10^{-2} events/rcy, which represents an increase by a factor of 5 over the hurricane initiating frequency for coastal data.

GEH developed hurricane strike frequencies for Mode 5, Mode 5 (Open), and Mode 6 (Unflooded) based on the number of hours per calendar year that the plant is expected to be in each shutdown POS. The analysis assumed one outage every 2 calendar years. The shutdown hurricane Mode 6 (Unflooded) high-winds strike frequency was added to the shutdown hurricane Mode 5 (Open) high-winds strike frequency so that both modes were evaluated assuming Mode 5 (Open) conditions. GEH believes that that combining these two modes is acceptable, assuming that coastal plants have several days of warning to prepare for a hurricane strike and should be able to transition to a plant configuration such as Mode 5 (Open) before the hurricane strike.

GEH used the shutdown PRA accident sequence structures, system fault trees, and success criteria to calculate shutdown high-winds CDF and releases. GEH estimated the mean high-winds shutdown CDF to be 4.0×10^{-8}/yr. Since GEH assumed the containment 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 99 percent of the shutdown high-winds CDF.

19.1.6.2.3.2 Acceptance Criteria

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used applicable guidance from SRP Section 19.0, Revision 2, in its review.

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

19.1.6.2.3.3 <u>Staff Evaluation</u>

19.1.6.2.3.3.1 <u>Shutdown High-Winds, Risk-Significant Core Damage Scenarios and Dominant Contributors</u>. The staff has reviewed the applicant's explanation of dominant contributors to risk and finds it acceptable. In addition, and for the reasons discussed below, the staff finds that combining the high-winds strike frequencies for Mode 5 (Open) and Mode 6 (Unflooded) is an acceptable approach for treating these conditions.

GEH did not assess the high-winds risk during Mode 6 (Unflooded) operation. In Mode 6 (Unflooded), the containment is open, the reactor vessel is open, and the water above the core will not keep the core cool for an extended period of time without additional mitigating systems. GEH assumed that there would be sufficient time before a hurricane strike for the plant to transition to another mode so that long-term cooling water would be more reliable. The model reflects this assumption by adding the shutdown hurricane Mode 6 (Unflooded) high-winds strike frequency to that of the shutdown hurricane Mode 5 (Open). GEH documented this assumption in Table 19.2-3 of DCD, Tier 2, Revision 6, as a key risk insight from the analysis. Table 19.2-3 now contains an entry that states the following:

> The plant should not be in a Mode 6 Unflooded condition when a hurricane strike occurs. This is because in Mode 6 Unflooded the containment is open, the

reactor vessel is open and the water above the core will not keep the core cool for an extended period of time.

The staff finds this treatment of high-winds risk during Mode 6 (Unflooded) operation to be acceptable since this key risk insight will be available to all COL applicants that reference the ESBWR DC.

The high-winds risk assessment presented in Revision 4 of the PRA report does not explicitly quantify scenarios that could occur during Mode 4 because of the short period assumed for transition from Mode 3 to Mode 5. The staff recognized that the PRA report, Revision 4, Table 18-1, and DCD, Tier 2, Table 19.2-3, did not capture certain implicit insights. In RAI 19.1-186, the staff asked the applicant to address two important implicit assumptions in the high-winds risk assessment: (1) the plant will go to Mode 4 and will not deinert in Mode 4 when the plant shuts down in anticipation of a hurricane strike and (2) when a hurricane is approaching the site, the plant will not voluntarily take any equipment out of service that is credited in the high-winds PRA. In its response, the applicant has added these insights in Table 19.2-3 in Revision 7 of the DCD and Table 18-1 of the PRA report, Revision 5. This is acceptable to the staff. Therefore, RAI 19.1-186 is resolved.

19.1.6.2.3.3.2 Results and Insights from the Shutdown High-Winds Importance and Sensitivity Studies. The applicant has performed studies of RAW using the high-winds PRA model. The results of these studies show that the shutdown high-winds PRA results are not as sensitive to operator errors as CCFs in the following systems: GDCS, ADS, Q-DCIS, UPS, and DCS, which have the highest RAWs. For example, CCFs in the Q-DCIS, UPS, and DCS have RAW values exceeding 1,000. In contrast, failure of the operator to recognize the need for low-pressure makeup after depressurization has a RAW value of approximately 30. Failure of the operator to open two DPVs manually has a RAW value of approximately 9.

The staff has reviewed the applicant's sensitivity studies and finds that they are acceptable for gathering important insights regarding the risk contribution from high winds during shutdown operation.

19.1.6.2.3.4 Conclusion. The staff reviewed the GEH shutdown high-winds risk assessment and finds it be technically adequate to support DC and the identification of risk insights.

19.1.6.2.4 Results and Insights from the Low-Power and Shutdown Internal Seismic Assessment

19.1.6.2.4.1 Summary of Technical Information. 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.

19.1.6.2.4.2 Acceptance Criteria

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used applicable guidance from SRP Section 19.0, Revision 2, in its review.

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

19.1.6.2.4.3 Staff Evaluation. The HCLPF nodal fault trees used for the shutdown seismic analysis are the same as those used in the full at-power seismic analysis, with the exception of the structural failure node. The structural failure nodal fault tree (SIS) for the shutdown seismic event tree is developed to include the structural failures included in the at-power SI nodal fault tree, as well as the structural elements related to reactivity control. This approach is acceptable to the staff.

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. GEH developed shutdown seismic event trees for Mode 5, Mode 5 (Open), Mode 6 (Unflooded), and Mode 6 (Flooded). No shutdown accident sequence has an HCLPF lower than 0.84g because of the assumption made for component-level HCLPF. The PRA-based shutdown SMA shows that the ESBWR design can meet the 0.84g HCLPF value, if the seismic capacities of safety system components are qualified to be above the specified acceptable design value of 0.84g. In Section 19.2.6 of DCD, Tier 2, Revision 6, the applicant stated the following:

> The COL applicant referencing the ESBWR certified design shall compare the as-built SSC HCLPFs to those assumed in the ESBWR SMA shown in Table 19.2-4 [of the DCD, Tier 2, Revision 6]. 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.

The staff finds this COL information item (i.e., COL Information Item 19.2.6-1-A) to be acceptable.

19.1.6.2.4.4 Conclusion. The staff has reviewed GEH's shutdown seismic assessment and finds it technically adequate to support DC and the identification of risk insights.

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

19.1.7.1 Summary of Technical Information

The applicant used the PRA insights and assumptions to develop a list of DC requirements. DCD, Tier 2, Revision 6, incorporates these requirements in Table 19.2-3, 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.

19.1.7.2 Acceptance Criteria

No specific regulatory requirements govern the safety insights used to support DC. However, the staff used applicable guidance from SRP Section 19.0, Revision 2, in its review.

The staff evaluated the PRA input to the DC process against the Commission's objectives for new reactor designs, as 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 TS, as well as COL and interface requirements.

19.1.7.3 Staff Evaluation

19.1.7.3.1 Probabilistic Risk Assessment Input to the Design Certification Process

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 through 19.1.6 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 RAI 19.1.0.-4, Supplement 1, in order to understand why certain assumptions and insights were not translated into DC requirements. The staff tracked RAI 19.1.0.-4 Supplement 1, Parts A through F as an open items in the SER with open items. In response, the applicant reviewed the assumptions in the PRA and, using its process for identifying and documenting key assumptions and risk insights, GEH included additional assumptions related to design and operation in Table 19.2-3 of DCD, Tier 2, Revision 6. The applicant also provided additional explanation of its process for ensuring that key assumptions and risk insights are identified and documented for use by COL applicants. The staff has reviewed the revisions in DCD, Tier 2, Revision 6, and finds them acceptable. Therefore, RAI 19.1.0-4, Supplement 1 is resolved.

In light of the revisions made to Table 19.2-3 in DCD, Tier 2, Revision 6, the staff finds that the applicant has achieved the Commission's objective of identifying important safety insights related to design features and assumptions made in the PRA to support certification requirements.

19.1.7.3.2 Probabilistic Risk Assessment Input to the Maintenance Rule Implementation

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 DCD, Tier 2, Revision 6. Section 17.4 of this report documents the staff's evaluation of the information provided in Section 17.4.6 of DCD, Tier 2, Revision 6.

19.1.7.3.3 Probabilistic Risk Assessment Input to the Reliability Assurance Program

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 ensuring equipment safety and limiting risk to the public. GEH used the importance measures derived from the PRA to develop a list of risk-significant SSCs

for the ESBWR DC, as discussed in Section 17.4.6 of DCD, Tier 2, Revision 6. 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.

19.1.7.3.4 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 postaccident 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. The RTNSS process requires an assessment of safety functions that are relied upon during at-power and 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. Section 22 of this report documents the staff's evaluation of the focused PRA studies used to support the RTNSS process.

19.1.8 Conclusion

The NRC staff evaluated the ESBWR PRA and its use in the design and certification processes and identified a number of issues that the applicant did not adequately address. GEH has now addressed all of these issues adequately through its responses to the staff's RAIs. The staff has described each open issue and the basis for resolution of the issue in the appropriate section of this report. Based on its review, the staff finds that the applicant has adequately addressed the Commission's objectives, which are described in Section 19.1.1, regarding the preparation and use of a PRA 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 10 CFR 52.47(a)(23), "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 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 10 CFR 52.47(a)(8).

The staff used applicable guidance from SRP Section 19.0 Revision 2 in its review.

19.2.2 Severe Accident Prevention

19.2.2.1 Severe Accident Prevention Features

Section 19.1.3.1 of this report summarizes important severe accident prevention 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, Tier 2, Revision 6, 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, Tier 2, Revision 6, 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/PCCS pools and SFP 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 B) described in DCD, Tier 2, Revision 6, Section 19A.8.3.

19.2.2.1.5 Intersystem Loss-of-Coolant Accident

As stated earlier in Section 19.1.3.1 of this report, 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 of this report describes this issue.

19.2.2.1.8 Isolation Condenser

Section 19.1.3.1 of this report describes this issue.

19.2.2.2 Conclusion

The applicant has provided a number of important design features that contribute to the prevention of severe accidents. The staff has evaluated the impact of these features on risk and found that in many cases these features can substantially reduce the risk associated with severe accidents. The staff concludes that, in accordance with the Commission's objectives for new reactor designs, the applicant has reduced the significant risk contributors of existing operating plants by introducing appropriate and effective design features that contribute to the prevention of severe accidents.

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.

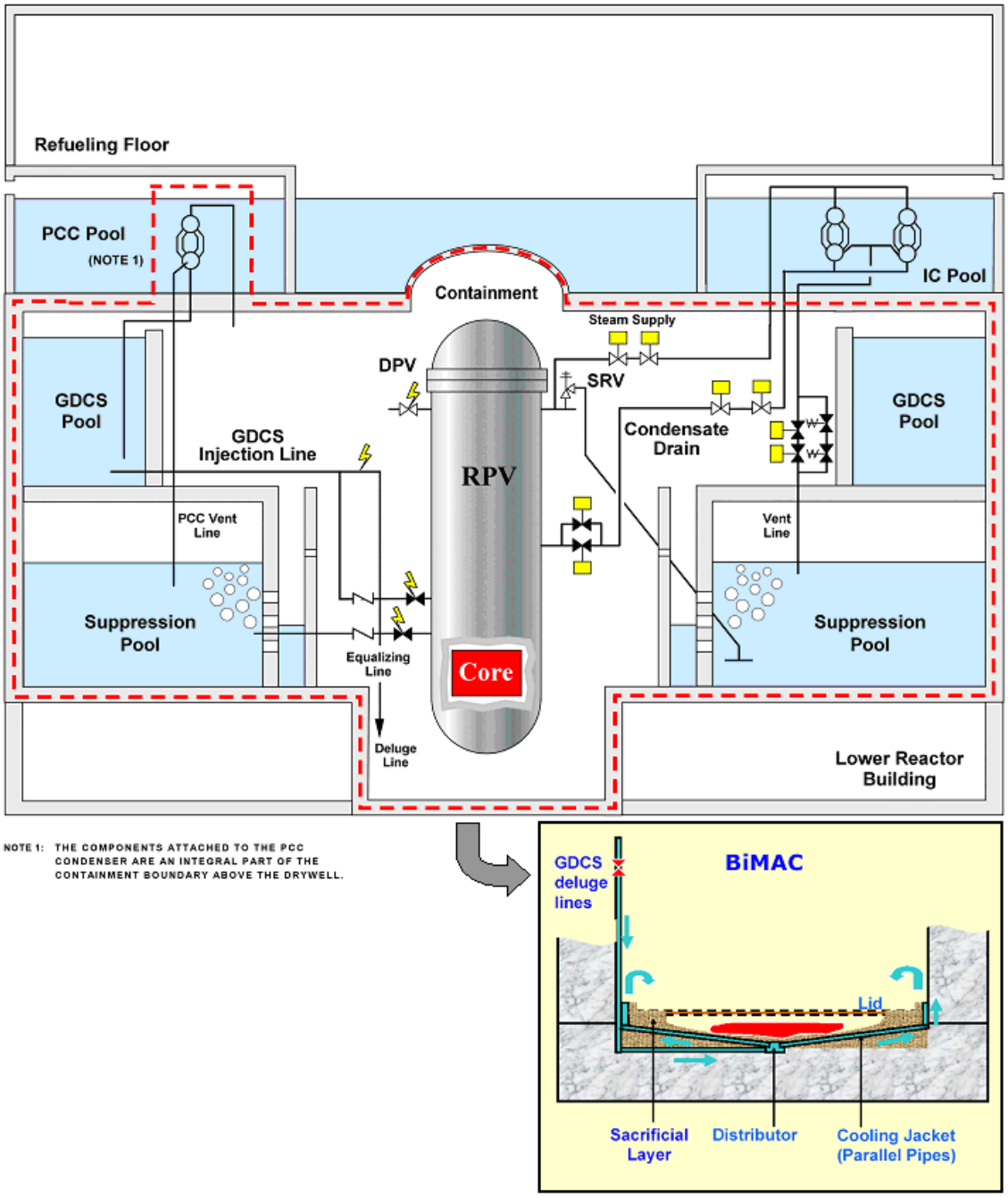

Figure 19.2-1 ESBWR design features for severe accident conditions

19.2.3.2 Severe Accident Progression

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 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 an ADS to ensure 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. 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) 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) 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 evaluates 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 Hydrogen Generation and Control

19.2.3.3.1.1 Staff Evaluation. The analysis of the radiolytic oxygen concentration in containment, as discussed in Section 8.1 of the PRA report, Revision 4, is based on the methodology of Appendix A to SRP Section 6.2.5, Revision 2 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).

19.2.3.3.1.1.1 Preventive and 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.

19.2.3.3.1.1.2 Risk Caused by Deinerted Operation. The PRA analysis assumes a 24-hour/yr period of noninerted containment atmosphere. This adds an additional BYP frequency of 4.6×10^{-11}/yr.

19.2.3.3.1.1.3 Basis for Acceptability. The specific requirements in 10 CFR 50.44(c)(2) establish the following for future water-cooled reactor applicants and licensees:

> [a]ll 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, in accordance with 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-DBA 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. Implementation of the severe accident management guidelines (SAMGs) will manage the accumulation of combustible gases that may develop in the period after about 24 hours. 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.

19.2.3.3.2 Conclusion

The present review confirms 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.

19.2.3.3.2 Core Debris Coolability

19.2.3.3.2.1 Staff Evaluation. 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 operating LWRs, the severe accident management case is based on the 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 the 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 associated with these interactions, inconsequential.

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 used for the passive PWR designs in the United States. 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.

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 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 if 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 $0<\theta<20$ degrees, 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 near the upper end of this range.

Protective concrete layer. A protective layer of concrete 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.

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 protective concrete 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 protective concrete layer to eliminate any challenges resulting from impingement of the superheated, metallic melt jets on the BiMAC cooling pipes.

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 go only 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 melt attack. Two sumps are 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, while 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 1×10^{-3}. The detection and actuation of the system are 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 interval between 0 and 20 degrees.

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 requested documentation of the test results in RAIs 19.2-23, Supplement 2, and 19.2-25, Supplement 2. The staff tracked RAI 19.2-23, Supplement 2, and 19.2-25 as open items in the SER with open items. By letter dated March 28, 2008, the applicant provided the test results as a topical report (NEDO-33392, Revision 2, "The MAC Experiments Fine Tuninng of the BIMAC Design," dated March 28, 2008) in its response to RAIs 19.2-23, Supplement 2, and 19.2-25, Supplement 2. Review of the report engendered additional RAI questions 19.2-93 through 19.2-119 and supplemental RAIs. GEH responded to these RAIs and GEH also decided to modify the design to change the material on the LDW floor from zirconia to a layer of sacrificial concrete. In response to RAI 19.2-127, GEH submitted to the NRC staff an analysis of the effects of erosion of this concrete. The GEH responses satisfactorily show that the BiMAC

would be adequately protected and would function as designed and RAIs 19.2-93 through 19.2-119 and RAI 19.2-127 were resolved. Therefore, RAIs 19.2-23 and 19.2-25, including their supplements, and the associated open items are also resolved.

19.2.3.3.2.2 Conclusion. The PRA report, Revision 4, describes the detailed probabilistic framework, quantification of BMP loads, quantification of fragility to BMP, and prediction of failure probability caused by BMP. The results of the BMP device analysis described in the PRA report, Revision 4, 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 BMP issue becomes moot, as is containment overpressurization generated by the concrete decomposition gases.

19.2.3.3.3 High-Pressure Melt Ejection

19.2.3.3.3.1 Staff Evaluation. 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 vent 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, the events could, without further analysis, be left in the category of residual risks. Still, because of the potentially severe consequences, the applicant chose to further examine the likelihood of energetic CF from DCH and concluded, by analysis, that such a failure is physically unreasonable.

The key factor in reaching this conclusion is that the approximately 14 square meters (m^2) of vent area, connecting to the condensation potential of the suppression pool, make 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 physically unreasonable, 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 are 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 as yet to be determined protective layer of concrete 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 meters and 0.3 meters for 100 and 300 tons of melt involved in the expulsion process, respectively. These results establish 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 physically unreasonable 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 throughout the 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 physically unreasonable 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. As a result of these analyses, GEH found 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 -1,000 K, which is considerably below the near-melting temperatures (over 1,650 K) that could cause failure.

However, the GEH calculations also show short periods of potentially very high temperatures in the LDW atmosphere (up to 4,000 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 noted in the high-pressure CET. The branch is used only in a Level 3 sensitivity study.

19.2.3.3.3.2 Conclusion. Based on its review of the applicant's analyses, the staff accepts that the exclusion of DCH-induced catastrophic containment failure is reasonable. Furthermore, based on its confirmatory assessment, the staff also agrees that a high probability of localized liner failures in the LDW exists.

19.2.3.3.4 Fuel Coolant Interactions

The containment function may be challenged by a rapid energy release during an 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.

19.2.3.3.4.1 Staff Evaluation

19.2.3.3.4.1.1 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 considered in detail only the issue of in-vessel steam explosions for PWRs. 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 also applies to the ESBWR design.

19.2.3.3.4.1.2 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 kilobar 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 leaktightness 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, BMP, and containment pressurization due to 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, incipient liner failure and noticeable concrete damage occur. 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 physically unreasonable 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, "Analysis of Ex-Vessel Fuel Coolant Interactions for ESBWR," issued July 2006). Fragilities were not recalculated. Calculations for 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 physically unreasonable for low-pressure core melt drops in pools less than 1.5 meters deep.

19.2.3.3.4.1.3 Minimization of Ex-Vessel Steam Explosion Effects in the ESBWR. The principal element of the GEH ESBWR severe accident management approach to 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 spillover lines with squib valves, similar 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 is designed to be functional immediately upon opening of the deluge lines. Thus, preflooding of the LDW is unnecessary, and the detailed design of the deluge line valve activation system is based on detecting melt arrival onto the LDW floor. This activation system is accessible both automatically and 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 appear only as options in the SAMGs. Section 19.2.3.3.8 of this report further discusses spray usage.

Section 21.4 of the PRA report, Revision 4, 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 physically unreasonable.

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., depth 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 exclude ranges of conditions that GEH does not believe could be reasonably captured by current capabilities and experience.

19.2.3.3.4.2 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} per reactor-year. 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 physically unreasonable. The staff previously performed separate analyses for the ABWR design to justify a similar conclusion for that design. (See ERI/NRC-93-203).

19.2.3.3.5 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.

19.2.3.3.5.1 Staff Evaluation

19.2.3.3.5.1.1 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.

19.2.3.3.5.1.2 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 TB. 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 exists. 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 RWCU 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 SLCS check valves.

Release categories BOC, BYP, and OPVB include scenarios that bypass the suppression pool. Their combined frequency contributes about 10 percent of the CFs, and their risk contribution is about 9 percent of the 72-hour societal risk within 10 miles. This is about 10 percent of the total

offsite risk from internal event sequences and therefore does not present an undue offsite risk. Nevertheless, in RAIs 19.2-6, 19.2-10, and 19.2-11, the staff requested further information on vacuum breaker performance pertaining to vacuum breaker design. The staff tracked RAIs 19.2-6, 19.2-10, and 19.2-11 as open items in the SER with open items.

Subsequently, GEH modified the design to include upstream isolation valves to prevent bypass leakage in the event that the vacuum breakers do not completely close. Redundant proximity sensors and temperature sensors are also provided to detect the closed position of the vacuum breakers. The documentation of Table 19.1-1 in DCD, Tier 2, Revision 6, explicitly references these changes. The issue of potential containment bypass resulting from vacuum breaker leakage is resolved. Therefore, RAIs 19.2-6, 19.2-10, and 19.2-11 and the associated open items are resolved.

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. 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 still be under the ultimate pressure capability expected of the containment.

19.2.3.3.5.2 Conclusion. The staff concludes that GEH performed a complete analysis to facilitate an understanding of the capability of the ESBWR containment to accommodate a range of bypass conditions.

19.2.3.3.6 Containment Vent Design

19.2.3.3.6.1 Staff Evaluation. The system designated 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 4.0.6 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 of the PRA report, Revision 4. 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 stated that this arrangement for venting is satisfactory because the line sizes are adequate, and the system has the requisite monitoring and control capabilities. The staff agrees that the line sizes are adequate and that the requisite monitoring can be put into place.

19.2.3.3.6.2 Conclusion. The applicant has included in the ESBWR design the capability to vent the containment. The staff has reviewed the design of the venting capability and concludes that it can be an effective feature for mitigating containment pressurization events that may challenge containment integrity.

19.2.3.3.7 Equipment Survivability

SECY-90-016 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, Revision 4, 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 Staff Evaluation.

19.2.3.3.7.1.1 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. From this perspective, 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 system 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

19.2.3.3.7.1.2 Severe Accident Environmental Conditions. The applicant performed MAAP 4.0.6 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.

19.2.3.3.7.2 Conclusion. The applicant carried out a systematic evaluation of 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

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 DCD, Tier 2, Revision 6, imply that there will be interlocks that must be overridden before the sprays can be used. This is acceptable because the PRA does not credit any benefit of the containment spray system on fission product.

19.2.3.4 Conclusion

The applicant has provided several important design features that contribute to the mitigation of severe accidents. The staff has evaluated the impact of these features on risk and finds that these features can be effective in reducing the risk associated with severe accidents. The staff concludes that, in accordance with the Commission's objectives for new reactor designs, the applicant has reduced the significant risk contributors of existing operating plants by introducing appropriate and effective design features that contribute to the mitigation of severe accidents.

19.2.4 Containment Performance Capability

This section describes the staff's assessment of the ESBWR containment structural performance to resist loads induced by postulated beyond-design-basis severe accidents. The ESBWR containment design and structural characteristics are described in DCD, Tier 2, Revision 6, Sections 3.8.1 and 3.8.2. DCD, Tier 2, Revision 6, Chapter 19, and the PRA report, Revision 4, describe the severe accident assessments, including the containment performance under postulated beyond-design–basis accident scenarios. The staff reviewed the applicable sections of DCD, Tier 2, Revision 6, relating to PRA-based SMA for the containment and the containment performance against overpressurization induced by beyond-design-basis severe accident loads.

The staff used the review criteria as described in Section 19.2.4.1 below to review and evaluate DCD, Tier 2, Revision 6, and the supporting PRA report, Revision 4, and to determine the adequacy of the applicant's assessment of the containment structural performance. This section describes the GEH containment structural performance assessment and the staff's evaluation of that assessment.

19.2.4.1 Regulatory Criteria

The staff used the following relevant regulations and regulatory guidance documents to perform this review:

- General Design Criterion (GDC) 16, “Containment Design,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” 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 relates to the quality assurance criteria for nuclear power plants.

- 10 CFR 52.47 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 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 the performance of an analysis using an analytical technique acceptable to the staff 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 Reports 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)(27) requires that the applicant provide a description of a 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 expectation in SECY-93-087 with respect to the deterministic containment performance assessment is 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.

The staff has used applicable guidance from SRP Section 19.0, Revision 2.

19.2.4.2 Summary of Technical Information

In DCD, Tier 2, Revision 6, Section 3.8, GEH described the physical characteristics of the concrete containment for the ESBWR plant. The containment is a reinforced concrete structure with a 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. Severe accident loads may also affect 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.

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. The ESBWR primary containment is designed to withstand a maximum pressure of 0.310 MPa (45 psig) and a design temperature of 340 degrees F.

19.2.4.2.1 10 CFR 50.44 Requirement

The regulation in 10 CFR 50.44(c)(5) requires that an analysis be performed to demonstrate the containment structural integrity against loads generated by an accident that releases hydrogen from 100-percent fuel clad-coolant reaction accompanied by hydrogen burning. Section 8 of the GEH PRA report, Revision 4, provides an evaluation of the ability of the ESBWR containment to

withstand system-related containment challenges associated with potential combustible gas deflagration, overpressurization, and bypass. The ESBWR design employs an inerted containment, and 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. GEH concluded that the CF caused by combustible gas deflagration is unrealistic. GEH also estimated the containment pressure induced by the hydrogen buildup to be 0.987 MPa gauge (143 psig) and performed a detailed containment structural analysis as described below.

DCD, Tier 2, Revision 6, Section 19B provides a detailed finite element analysis to estimate the containment structural capacity using the guidance of RG 1.70, Revision 3. GEH assessed the containment performance to withstand the pressure and temperature loads resulting from the containment hydrogen buildup, assuming 100-percent fuel clad-coolant reaction, and estimated the containment pressure capability in terms of ASME Service Level C or factored load limits to be 1.011 MPa gauge (146.5 psig), which is adequate to withstand the pressure load of 0.987 MPa gauge (143 psig) resulting from 100-percent fuel clad-coolant reaction. GEH also identified the limiting component as the RCCV liner strain at the connection of the UDW wall and the top slab.

19.2.4.2.2 SECY-93-087 Deterministic Containment Performance Expectation

DCD, Tier 2, Revision 6, Section 19B also addresses the SECY-93-087, Section I.J, expectation regarding the deterministic containment performance assessment against the pressure and temperature loads generated for the more likely accident scenarios, which GEH defined as sequences accounting for an aggregated 97 percent of CDF. The pressures and temperatures for these sequences were determined to be enveloped by the sequence T_nDP_nIN_TSL, a Class III sequence with peak pressure of 0.62 MPa gauge (90 psig). GEH assessed the pressure capability of the containment structure following the guidance of SECY-93-087 and determined that the containment Level C (or factored load) limit is much higher than 0.62 MPa gauge (90 psig) pressure load taking into account the temperature effect on the material strength.

19.2.4.2.3 Probabilistic Containment Performance Assessment

GEH developed the containment pressure fragility in DCD, Tier 2, Revision 6, Section 19C. The containment fragility is used in the Level 2 PRA and severe accident assessment of the containment phenomena. The containment pressure fragility was established with the aid of a detailed ABAQUS/ANACAP-U three-dimensional finite element containment model. The analysis also quantified the uncertainty associated with material properties and defined the failure criteria or limit states for estimating containment failure pressure capacity. Median capacity was calculated by setting all parameters to their median values.

The uncertainty in material properties and failure criteria was assessed by computing the 95-percent confidence value of a specific parameter, assuming a lognormal distribution, while keeping all other parameters at the median values. The containment failure pressure was also assumed as a lognormal distribution. Thus, the uncertainty in the failure pressure caused by the uncertainty of a parameter (either a material property or a failure criterion) can simply be determined using the relation, beta = $Ln(P_{95}/P_m)/(-1.645)$. The uncertainty can then be aggregated for all parameters using the square root of the sum of the squares (SRSS) method.

The modeling uncertainty (e.g., mesh fidelity, element formulations, robustness of the constitutive models) was assessed 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.

Section 19.3 of DCD, Tier 2, Revision 6, describes the containment phenomenological challenges such as DCH, EVE, and BMP and the containment response assessment. Chapter 21 of the PRA report, Revision 4, provides the detailed treatment of the containment phenomenological challenges and the corresponding containment responses, based on the ROAAM methodology.

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 of such events occurring is very small (i.e., 2.8×10^{-9}). The Level 1 PRA also indicates that high-pressure accidents contribute only about 1 percent of the CDF. The containment pressure load induced by a DCH event was estimated to be 0.7 MPa (100 pounds per square inch absolute), which intercepts the containment pressure fragility at very low probability (much less than 10^{-5}). GEH concluded that a DCH event in the ESBWR is physically unreasonable and categorized the DCH events discussed in Section 21.3 of the PRA report, Revision 1, as remote and speculative.

EVE events are energetic FCIs, which are triggered by melt released from the 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 4, GEH described the containment and BiMAC performance against an EVE. The relevant structures subject 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 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 meters), 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 physically unreasonable.

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 melts and preventing BMP of the molten core debris for a minimum of 24 hours and hence preventing 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 RB 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," issued August 1979. 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 as the failure mode.

GEH reevaluated the Level C capacity of the drywell head in DCD, Tier 2, Revision 1, Appendix 19B 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 1.033 MPa 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 Equation (B.8-1) (from Shield and Drucker, "Design of Thin-Walled Torispherical and Toriconical Pressure-Vessel Heads," issued June 1961) for plastic yielding failure mode and the Galletly Equation (B.8-3). GEH determined that the Shield and Drucker Equation (B.8-1) governs the pressure capacity of the head shell 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 of about 533 K (500 degrees 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 degrees 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 and Radhamohan, "Elastic-Plastic Buckling of Internally-Pressurized Thin Torispherical Shells," issued August 1979, and Galletly and Blachnut, "Torispherical Shells Under Internal Pressure—Failure Due to Asymmetric Plastic Buckling or Axisymmetric 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 is1.623 MPa at 500 degrees 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 (P_d) of 0.31 MPa (45 psig), governed by the plastic yielding of the drywell head shell.

The applicant later revised the fragility analysis in DCD, Tier 2, Revision 4, Section 19C 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 533 degrees K (500 degrees F), which is 4.65 times P_d.

The failure pressure was further revised in DCD, Tier 2, Revision 5, Section 19C as 1.374 MPa (4.43 P_d) at 533 degrees K (500 degrees F) with a 95-percent confidence level. The corresponding failure mode was identified by the tensile yielding of the flange anchor bolts.

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., IC/PCCS 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 degrees 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 RB 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 degrees 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 0.987 MPa 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 5, Section 19B identified it as a local peak strain due to the membrane and bending effect for which the Level C strain limit is 1 percent.

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, provides only the temperature time histories for the LDW, which show the steady-state temperature to be nearly 450 K (350 degrees F).

Another source of 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 1,000 K (1,340 degrees 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 degrees 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 physically unreasonable.

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 and replaced with a brief description in DCD, Tier 2, Revision 1, Section 19.2.3.5, which also includes a table of the qualitative structural HCLPF capacities. Section 15 of the PRA report, Revision 1, describes in detail 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, "A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," issued January 1983, 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 the 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 PGA, and the North Anna early site permit (ESP) site-specific SSE spectrum. In accordance with the soil-structure interaction analysis described in Appendix 3A to DCD, Tier 2, Revision 6, generic sites with 0.3g input (per RG 1.60) 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, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," issued May 1978, 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, Revision 6, 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. <u>Containment Liner—Failure of Pressure Containment Function during Severe Accident Loadings</u>. 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 values 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 be examined only 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 degrees F. At this temperature, GEH concluded that the ESBWR liner would not fail, given a containment internal pressure of 1.204 MPa.

In DCD, Tier 2, Revision 4, Section 19B 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 DCD, Tier 2, Revision 4, Section 19C the failure criteria for liner strain at a 95-percent confidence level was established at 2.04 percent at 260 degrees C (500 degrees F), and the corresponding 95-percent failure pressure for RCCV was calculated to be 1.317 MPa gauge (4.25 P_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 1,000 K (1,340 degrees F); however, for very short periods (less than 1 second), GEH estimated that the LDW could experience very high temperatures of up to 4,000 K (6,740 degrees F). A DYNA-3D analysis shows that a liner with concrete backing can sustain high temperatures up to 1,650 K (2,510 degrees F), and the calculated thermal strains are about 8 percent.

19.2.4.2.5.4 <u>Penetrations—Failure of Pressure Containment Function during Severe Accident Loadings</u>. 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 internally 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. EVE events result in energetic pressure pulses, with magnitudes in the kilobar 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.

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 failures of the reactor pedestal and the steel liner induced by steam explosions are physically unreasonable.

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 meters). 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 meters), 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 physically unreasonable.

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 comprises thick-walled steel pipes protected by a sacrificial refractory layer of 0.2-meter-thick ceramic zirconia, with properties to resist very high heat. 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 BMP by the molten core debris for a minimum of 24 hours, thus 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 use of the 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 physically unreasonable.

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 precluding 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 meters 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 described 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, Revision 6, and the PRA report, Revision 4, 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, revisions, and revisions of the PRA report, including the information in DCD, Tier 2, Chapters 6 and 19, and relevant sections of the PRA report regarding the structural containment performance in the event of severe accidents and (2) the GEH responses to the staff's RAIs.

19.2.4.3.1 10 CFR 50.44 Requirements

GEH addressed the requirements of 10 CFR 50.44 as they relate to hydrogen combustion in DCD, Tier 2, Revision 6, Sections 6.2.5.4 and 6.2.5.5. Since the ESBWR containment is inerted, the staff finds that the burning of hydrogen 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, Tier 2, Revision 6, 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 GEH estimated the containment pressure load resulting from the 100-percent fuel clad-coolant reaction 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 DCD, Tier 2, Revision 4, Section 19B. The applicant revised other sections of DCD, Tier 2, Revision 4, that are related to Level C containment pressure capacity by reference to Section 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 and nonsymmetric loads caused by GDCS pools and pool structures above the top slab, which an axisymmetric finite element model may be unable 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 degrees C (110 degrees F) specified for the drywell head while the rest of the UDW airspace is kept at 260 degrees C (500 degrees F) in steady state. Since the drywell head airspace is separated from the drywell airspace only by the bellow, which is made of a steel plate, the staff questioned whether the head shell can be kept at 43.3 degrees C (110 degrees F) while the drywell airspace is assumed to be at 260 degrees C (500 degrees F) steady state. Because the refueling pool is located directly above the drywell head, which is kept from being submerged during a postulated beyond-DBA, overheating of the drywell head shell is prevented. 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 asked GEH to address this issue. The staff tracked RAI 19.2-41 as an open item in the SER with open items.

In response to RAI 19.2-41, Supplement 2, GEH agreed with the staff that the temperature boundary condition was incorrectly specified for the drywell head in the containment pressure capacity analyses provided in DCD, Tier, Revision 4, Appendices 19B and 19C. Based on venting channels between drywell and drywell head airspaces and more detailed MAAP 4.0.6 analyses, GEH modified the temperature under the drywell head from 43.3 degrees C (110 degrees F) to 260 degrees C (500 degrees F) and the temperature of water in the pools above the drywell head from 43.3 degrees C (110 degrees F) to 100 degrees C (212 degrees F). GEH also updated Level C and pressure fragility analyses, which are provided in DCD, Tier 2, Revision 5. The Level C pressure capacity was determined to be 1.011 MPa, controlled by the RCCV liner tensile strain (DCD, Tier 2, Revision 4, identified the containment pressure capacity as being controlled by the drywell head). GEH has addressed the staff's concern regarding the temperature boundary condition for the drywell head. Based on the above discussion, the staff considers that the GEH response is adequate and acceptable, and RAI 19.2-41, Supplement 2, is closed. Therefore, RAI 19.2-41 and the associated open item are resolved.

The new ABAQUS/ANACAP-U analysis result shows that, at an internal pressure of 0.987 MPa gauge , or 3.18 P_d, the strain in the liner of the UDW wall at the connection with the top slab reached 0.72 percent, which exceeds 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 should consider pressure plus dead load alone. Based on the information 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 that GEH clarify how it calculated the strain. The staff tracked RAI 19.2-86 as an open item in the SER with open items.

In response to RAI 19.2-86, GEH addressed the staff's concern by performing the pressure capacity analysis with the appropriate temperature boundary conditions and identified the liner strains at locations where prominent geometric discontinuities are present and other locations away from any geometric discontinuity. GEH determined the Level C capacity of the liner in accordance with the criteria provided in ASME Code, Section III, Division 2, Subarticle CC-3720, for both membrane and membrane plus bending strain allowables. The staff has determined that the GEH response is adequate and the GEH analysis is acceptable, and RAI 19.2-86 is considered closed. Therefore, RAI 19.2-86 and the associated open item are resolved.

Based on the above discussion, the staff concludes that DCD, Tier 2, Revision 5, Section 19B adequately addressed the containment Level C pressure capacity to withstand the pressure loads induced by considering a 100-percent fuel clad-coolant reaction, and therefore, meets the requirements of 10 CFR 50.44(c)(5).

19.2.4.3.2 SECY-93-087 Deterministic Containment Performance Expectation

The staff reviewed the GEH approach to addressing the expectation stated in SECY-93-087 for containment performance (i.e., by referencing a containment Level C pressure capacity analysis described in RG 1.7, Revision 3). During its review, the staff identified several issues that should be considered in addressing 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, because it envelops 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 degrees F).

GEH calculated 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, Revision 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 that make up 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 degrees C (110 degrees F) specified for the drywell head.

The Level C analysis results, as revised by GEH and described in DCD, Tier 2, Revision 5, Section 19B show that, at an internal pressure of 0.62 MPa gauge , or 2.0 P_d, peak strains in the liners of the RCCV were kept well below the factored load limit (0.3-percent tensile membrane strain, ASME Code, Section III, Division 2, Subarticle CC-3720). Furthermore, the GEH analysis in Table 19B-6 of DCD, Tier 2, Revision 5, indicated that at 2.0 P_d, the induced stresses in RCCV rebar and concrete are significantly less than the ASME Code allowable stresses. The staff concludes that the GEH deterministic containment performance analysis meets the expectation of SECY-93-087.

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 the 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, combining the 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 a shear failure of the suppression pool slab near the RCCV wall governs the RCCV pressure capacity.

The staff also found that the set of empirical equations that GEH used 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 ratio of diameter to thickness of shell 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 issue of 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 5. 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 of the containment pressure boundary. During the staff's February 5–7, 2007, onsite audit, GEH presented the results of the ABAQUS analysis of containment performance, which the staff found to be appropriate. 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 degrees C (500 degrees F)), and (3) transient thermal conditions for a temperature spike

representative of DCH conditions (peak temperature at 538 degrees C (1,000 degrees 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.

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; 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 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, in RAI 19.2-40 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. For this reason, the staff questioned GEH's decision to base the pressure capability estimate for the drywell head shell on the empirical equations discussed above

The GEH response to RAI 19.2-40 explains 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 of this report and explained in Appendices 19B and 19C to DCD, Tier 2, Revision 5, GEH gave details of the three-dimensional ABAQUS/ANACAP-U analysis that replaced the analysis based on empirical equations. 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 the ASME Code, Section III. Table 19B-9 of DCD, Tier 2, Revision 5, provides the results. The drywell head Level C capacity for the steady-state temperature condition of 260 degrees C (500 degrees F) is 1.033 MPa gauge (3.2 P_d), which is controlled by the capacity of the inside flange plate of the 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 degrees C (500 degrees F) for the drywell head can be estimated using information provided in DCD, Tier2, Revision 6, Table 19C-10. Given the median capacity at 1.426 MPa gauge and the composite uncertainty of 0.1535, the staff estimates that the HCLPF (99-percent confidence value) is 0.99 MPa gauge (3.2 P_d). The staff believes that with the yielding of the bolts for the bolted flanges being the likely failure mode for the drywell head, an uncontrolled large release through the head would not be possible.

The staff concludes that the reevaluation of the pressure capacity of the drywell head is acceptable. Therefore, RAI 19.2-40 is resolved.

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 IC/PCCS 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 as provided in DCD, Tier 2, Revision 4, Section 19B 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 degrees C (110 degrees F). The staff questioned the use of 43.3 degrees C (110 degrees F) as the temperature of the head shell and issued RAI 19.2-41, Supplement 2. In response, GEH modified the temperature under the drywell head from 43.3 degrees C (110 degrees F) to 260 degrees C (500 degrees F) and the temperature of water in the pools above the drywell head from 43.3 degrees C (110 degrees F) to 100 degrees C (212 degrees F). GEH also updated the Level C and pressure fragility analyses provided in DCD, Tier 2, Revision 5. The staff considers the GEH assessment adequate and the revision included in DCD, Tier 2, Revision 5, Section 19B acceptable. Therefore, RAI 19.2-41 is resolved.

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, Tier 2, Revision 6, 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 in DCD, Tier 2, Revision 6 compare the maximum stresses in critical areas of the RCCV to Level C allowable limits. Figure 19B-5 in DCD, Tier 2, Revision 6 identifies the critical strain locations for the RCCV liner, where location A near the top of the 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 3.17 P_d. The liner strain at location A, however, at 2.5 P_d, would just exceed the 0.3-percent Level C limit.

The GEH fragility analysis, as provided DCD, Tier 2, Revision 4, Section 19C 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 degrees C (500 degrees F) steady-state

temperature, the median RCCV pressure capacity and lognormal uncertainty for liner tearing are 1.708 MPa gauge (5.51 P_d) and 0.1512, respectively. Therefore, the HCLPF pressure capacity for the RCCV is calculated to be 1.2 MPa gauge (3.877 P_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.

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 degrees F for the most likely severe accident scenarios for the ESBWR. The staff requested that GEH clarify the use of the value of 500 degrees F 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 degrees 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 degrees 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.

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, 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 onsite audit from February 5–7, 2007, 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 envelops 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.

Consistent with SECY-93-087, 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, Revision 6, 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 asked GEH to address two issues in the seismic margin assessment: (1) compatibility of the shape of the review-level earthquake spectrum with the design-basis spectrum and (2) selection of the review-level earthquake PGA to be about 1.67 times the design-basis PGA.

The staff finds that the seismic margin assessment based on PRA seismic sequences, as described in Section 15 of the PRA report, Revision 2, is in accord with SECY-93-087. However, the applicant estimated HCLPFs for only five structural components by analysis, while assuming that the remaining SSCs had HCLPFs equal to 0.84g PGA. Since Chapter 19 of DCD, Tier 2, Revision 4, did not state that all SSCs identified on the seismic sequences will be qualified for HCLPF capacity equal to or greater than 1.67 times the ESBWR certified seismic design response spectrum (CSDRS), the staff believes that the COL applicants or licensees can qualify SSC HCLPFs with respect to the site-specific ground motion response spectrum, which is generally less than the CSDRS. RAI 19.2-92 discusses this issue further.

To address the issues raised by the staff, in DCD, Tier 2, Revision 4, Chapter 19, and in Section 15 of the PRA report, Revision 2, GEH developed a performance-based seismic response spectrum (PBRS), which is the same as the 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. Based on its review of the PBRS, the staff questioned why GEH did not use the CSDRS for the margin assessment. The staff believes that, consistent with SECY-93-087, CSDRS should be used for the margin assessment. The staff tracked RAI 19.2-92 as an open item in the SER with open items. In the response to RAI 19.2-92, the applicant agreed to use the ESBWR CSDRS in the seismic margin assessment and to change Table 19.2-4 of DCD, Tier 2, Revision 4, accordingly. On this basis, the staff finds that the seismic margin method used for the ESBWR certified design is acceptable. The staff also agrees that the SMA has demonstrated the sequence-level HCLPF value of 1.67*CSDRS for the ESBWR standard design, provided that the associated COL Information Item 19.2.6-1-A will be successfully confirmed. Therefore, RAI 19.2-92 and the associated open item are resolved.

The fire water service complex, which is designed and analyzed using the CSRDS, consists of two waste 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 the Fire Protection Water System (FPWS) should have all three components in OR-gates. In RAI 19.2-91, the staff requested that the applicant correct the FPWS fault tree and provide a revised HCLPF calculation. The staff tracked RAI 19.2-91 as an open item in the SER with open items. In response to RAI 19.2-91, GEH agreed with the staff's request. GEH modified the fault tree for the FPWS to include all three components in OR-gates. GEH also assessed the HCLPF capacity for these components to exceed 1.67*SSE. The applicant revised Table 19.2-4 of DCD, Tier 2, Revision 5, Chapter 19, and Section 15 of the PRA report, Revision 4, to reflect the changes made to the FPWS. The staff finds that GEH has addressed the concern adequately, and the issue is closed. Therefore, RAI 19.2-91 and the associated open item are resolved.

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 pressure-induced liner failure should be assessed. The greatest concern is at major penetrations, where stress concentrations and constraints to thermal growth are expected.

The applicant used 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 subject to a 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 1,100 degrees F to be about 23 percent, based on the available test data for SA-533 and A36 steel. The staff found that these tests were performed using specimens typically 2 inches or less, and the tests 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 has determined that the applicant's approach is acceptable, and the issues related to the ultimate fracture strain and maimum linear strain discussed above are resolved.

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 4.0.6 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 millimeters squared (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 assess the leakage potential of major penetrations. GEH also stated that bolts for equipment hatches and the drywell head will be preloaded to ensure that there are 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 in DCD, Tier 2, Revision 6 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 DCD, Tier 2, Revision 6, 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 degrees C (500 degrees 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 degrees C (500 degrees F).

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 internally 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. EVE events result in energetic pressure pulses, with magnitudes in the kilobar 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. The 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-meters thick) of sacrificial refractory material (ceramic zirconia), which 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 the EVE pressure loads. GEH concluded that the failure of cavity structures from EVE events is physically unreasonable. 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 thermocouples 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 2,950–3,120 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 physically unreasonable.

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 physically unreasonable. 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 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 this failure in the Level 2 PRA accident progression analysis. The staff finds the GEH approach acceptable.

19.2.4.4 Conclusion

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 of 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.

On the basis of its review and assessment, the staff concludes that the applicant's containment performance evaluation meets 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 pressure fragility analysis.

19.2.5 Accident Management

19.2.5.1 Summary of Technical Information

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 nonstandard 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 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 to 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 to 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 reduce the need for the emergency response organization to make rapid decisions and will permit a greater reliance on support from outside sources. For longer times (several hours to several days) after an accident, 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, decisionmaking responsibilities have been delineated, and utility self-evaluation methodologies have been developed and utilized.

19.2.5.2 Staff Evaluation

In RAI 19.2.4-1 and its supplements, the staff requested additional information on the technical basis for severe accident management in the ESBWR and on the process that GEH will use to develop the severe accident guidelines (SAGs). In response to RAI 19.2.4-1, Supplement 2, GEH provided additional details regarding development of the SAGs and referred to NEDO-33274, Revision 2, "ESBWR Human Factors Engineering Procedures Development Implementation Plan," issued March 2007, which presents the processes and methodologies to be used in the development of procedures including ESBWR SAGs and the ESBWR SAMGs derived from them. The staff tracked RAI 19.2.4-1 as an open item in the SER with open items. GEH provided additional details in response to supplement 3 of RAI 19..2.4-1. The staff reviewed the response, which includes suggested changes to the containment flooding severe accident guideline currently in place to be followed by BWR Owners Group (BWROG)

members, and is satisfied that it will enhance the technical basis supporting the existing BWROG accident management procedures. Therefore, RAI 19.2.4-1 and the associated open item are resolved.

19.2.5.3 Conclusion

The staff finds the process that the applicant proposes to use for the development of SAGs to be consistent with the process used by currently operating BWRs and, therefore, adequate.

19.2.6 Consideration of Potential Design Improvements under 10 CFR 50.34(f)

19.2.6.1 Regulatory Criteria

In 10 CFR 52.47(b)(2), the NRC requires applicants for standard DC to include an environmental report required by 10 CFR 51.55, "Environmental report—standard design certification." Regulation 10 CFR 51.55(a) requires the DC applicant to "address the cost and benefit of severe accident mitigation design alternatives (SAMDA), and the bases for not incorporating SAMDAs in the design to be certified."

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 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 (RAI 19.4-1, Supplement 3). 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.

The staff has followed applicable guidance from SRP Section 19.0, Revision 2 in performing its analysis.

19.2.6.2 Summary of Technical Information

19.2.6.2.1 Estimate of Risk for the ESBWR

As stated in Section 19.1.4.3 of this report, 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, Revision 6. Table 19.1-10 of this report 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 four 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. The PRA report, Revision 4 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 two 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 are well protected 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
- PCCS
- 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 considered to be similar to those already included in the ESBWR design, 29 items marked as procedural or administrative as opposed to design features (and that had benefits considered unlikely to exceed those of alternatives evaluated relative to their potentially high costs). Another 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 January 1997, 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 $4,630. 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 of this report).

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 19.2.6.3 of this report further discusses this assessment.

19.2.6.2.4 Cost Impacts of Candidate Design Improvements

NEDO-33306, Revision 1, "ESBWR Severe Accident Mitigation Design Alternatives," issued August 2007, does 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 that of current operating reactors; therefore, any additional design modifications would be costly 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 results provided 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 total 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^{-7}/yr; total CDF from Revision 4 of the PRA report). The revised CDF is driven by high CDFs from internal and high-wind events during shutdown.

The applicant provided estimates using a 7-percent discount rate. The NRC 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. The revised version 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 almost double 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]	232,680
	Occupational	38	76	230
Property	Offsite	157	1,570 [d]	55,610 [e]
	Onsite	NA [f]	NA [f]	NA [f]
Cleanup and Decontamination	Onsite	1,167	1,591	4,840
Replacement Power		2,900	34,486	89,000
Total		**4,628**	**41,383**	**382,360**

[a] "Best estimate" is based on mean release frequency (from Revision 1 of the PRA report) and "best estimate" parameter values.

[b] Maximum estimate is based on mean release frequency (from Revision 1 of the PRA report) and high estimate parameter values.

[c] Reviewer maximum is based on parameter values used in b, release frequency (from Revision 4 of the PRA report), 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] This value was not analyzed.

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 have an excessive impact 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 severe accident mitigation alternatives (SAMA) reports and from the ABWR design. The ESBWR design already incorporates several design enhancements related to severe accident mitigation. These design improvements have resulted in a CDF that is about one 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), as CDF sequences that contributed more than 40 percent of CDF in the ABWR design are eliminated.

The staff considers the applicant's review of the potential SAMDAs 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 report and a 3-percent discount rate indicate a maximum value of about $382,360. The staff notes that the estimated averted public exposure is a major contributor. This arises from high release frequencies for internal and high-wind events during shutdown which conservatively assume core damage scenarios to result in a large release, because the containment is open during most of the shutdown period. The staff also notes that the estimated averted cost could be higher, if one were to apply correction factors because of uncertainties in release frequencies and CDFs, and the adjustment for annual replacement power cost for future energy cost increases.

19.2.6.4 Conclusion

GEH indicated that any design modifications, including even a change of a manufacturer to reduce CCF, would incur costs of 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 concludes 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.

Based on the applicant's response, RAI 19.4-1 is resolved.

19.2.7 Design Features for Protection against a Large, Commercial Aircraft Impact

This section describes the staff's evaluation of the description of design features and functional capabilities credited by the applicant to show that the facility can withstand the effects of a large, commercial aircraft impact. These design features and functional capabilities are described in DCD, Tier 2, Revision 6, Section 19D.

The impact of a large, commercial aircraft is a beyond-design-basis event. Under 10 CFR 50.150, applicants for new nuclear power reactors[12] are required to perform an assessment of the effects on the designed facility of the impact of a large, commercial aircraft. Applicants are required to submit a description of the design features and functional capabilities identified as a result of the assessment (key design features) in their DCD together with a description of how the identified design features and functional capabilities show that the acceptance criteria in 10 CFR 50.150(a)(1) are met. Applicants subject to 10 CFR 50.150 must make the complete aircraft impact assessment available for NRC inspection, at the applicants' offices or their contractors' offices, upon NRC request in accordance with 10 CFR 50.70, 10 CFR 50.71, and Section 161.c of the Atomic Energy Act of 1954, as amended.

19.2.7.1 Regulatory Criteria

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," March 2007 was not used to perform this review because it does not address large, commercial aircraft impact analysis requirements. The staff used the following relevant regulations and guidance to perform this review.

19.2.7.1.1 Applicable Regulations

- 10 CFR 50.150(a)(1) requires that applicants perform a design specific assessment of the effects on the facility of the impact of a large, commercial aircraft. Using realistic analyses, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions: (i) The reactor core remains cooled, or the containment remains intact; and (ii) spent fuel cooling or SFP integrity is maintained.

- 10 CFR 50.150(b) requires that the final safety analysis report include a description of: (1) the design features and functional capabilities which the applicant has identified for

[12] "Applicants for new nuclear power reactors" is defined in the Statement of Considerations for the Aircraft Impact Rule [74 FR 28112, June 12, 2009].

inclusion in the design to show that the facility can withstand the effects of a large, commercial aircraft impact in accordance with 10 CFR 50.150(a)(1); and (2) how those design features and functional capabilities meet the assessment requirements of 10 CFR 50.150(a)(1).

19.2.7.1.2 Review Guidance

- Draft Guide (DG) 1176 "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts," issued July 2009, provides guidance for meeting the requirements in 10 CFR 50.150(a), and specifically, documents NRC endorsement of the methodologies described in the industry guidance document, Nuclear Energy Institute (NEI) 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Revision 7, issued May 2009.

- Statements of Consideration for the aircraft impact assessment rule [74 FR 28112, June 12, 2009] which indicate, among other things, that for the NRC to conclude that the rule has been met, it must find that the applicant has performed an aircraft impact assessment reasonably formulated to identify design features and functional capabilities to show, with reduced use of operator action, that the acceptance criteria in 10 CFR 50.150(a)(1) are met.

- The following NRC staff interim review guidelines:

  (a) Reasonably Formulated Assessment Guideline

  The NRC considers an aircraft impact assessment performed by qualified personnel using a method that conforms to the guidance in NEI 07-13, Revision 7 to be a method which is reasonably formulated. The NRC considers qualified personnel to be: (1) an applicant who is the designer of the facility for which the aircraft impact assessment applies; and (2) an applicant's primary contractor for the aircraft impact assessment who has designed a nuclear power reactor facility either already licensed or certified by the NRC or currently under review by the NRC.

  (b) Reactor Core and Spent Fuel Pool Cooling Design Features Guideline

  The "reactor core cooling" criterion or "spent fuel pool cooling" criterion in 10 CFR 50.150(a)(1) is satisfied if design features have been included in the design of the plant to specifically perform that cooling function with reduced use of operator action.

  (c) Intact Containment Guideline

  The "intact containment" criterion in 10 CFR 50.150(a)(1) is satisfied if the containment: (1) will not be perforated by the impact of a large, commercial aircraft; and (2) will maintain ultimate pressure capability, given a core damage event until effective mitigation strategies can be implemented. Effective mitigation strategies are those that provide, for an indefinite period of time, sufficient cooling to the damaged core or containment to limit temperature and pressure challenges below the ultimate pressure capability of the containment as defined in Section 19 of the DCD, Tier 2, Revision 6.

  (d) Spent Fuel Pool Integrity Guideline

The “spent fuel pool integrity” criterion in 10 CFR 50.150(a)(1) is satisfied if the impact of a large, commercial aircraft on the SFP wall or support structures would not result in leakage through the SFP liner below the required minimum water level of the pool.

(e) Reduced Operator Action Guideline

The NRC considers use of operator action to be reduced when: (1) all necessary actions to control the nuclear facility can be performed in the control room or at an alternate station containing equipment specifically designed for control purposes; and (2) a reduced amount of active operator intervention, if any, is required to meet the acceptance criteria in 10 CFR 50.150(a)(1). Reduction in the use of operator action is measured relative to the actions required to address aircraft impact without the aircraft impact assessment rule in place (e.g., similar actions contained in operational programs in place at current operating reactor sites).

19.2.7.2 Summary of Technical Information

In DCD, Tier 2, Revision 6, Section 19D, the applicant states that they performed an aircraft impact assessment in accordance with the requirements in 10 CFR 50.150(a)(1) using the methodology described in NEI 07-13, “Methodology for Performing Aircraft Impact Assessments for New Plant Designs,” Revision 7, as endorsed by the NRC in DG-1176. Based on the results of the assessment, the applicant has identified a set of key design features to show that the acceptance criteria in 10 CFR 50.150(a)(1) are satisfied. These key design features are reported in DCD, Tier 2, Revision 6, Section 19D, along with references to other sections of the DCD that provide additional detail. DCD, Tier 2, Revision 6, Section 19D also contains descriptions of how the key design features show that the acceptance criteria in 10 CFR 50.150(a)(1) are met.

19.2.7.2.1 Description of Key Design Features

The credited design features, their function(s), and references to sections containing the detailed descriptions are summarized below:

- The ICS, as described in DCD, Tier 2, Revision 6, Section 5.4.6 provides core cooling.
- The ECCS, as described in DCD, Tier 2, Revision 6, Section 6.3 provide core cooling.
- The Main Steam Isolation System (MSIS), as described in DCD, Tier 2, Revision 6, Section 5.4.5 maintains high pressure for core cooling with the ICS.
- The CRDS, as described in DCD, Tier 2, Revision 6, Section 4.6 inserts control rods to shutdown the reactor. This enables core cooling with the systems described above.
- The Safety Related Instrumentation and Control System (Q-DCIS), as described in DCD, Tier 2, Revision 6, Section 7.1 actuates the CRDS to shutdown the reactor and enable core cooling and initiates ADS and Gravity Driven Cooling System (GDCS) for core cooling at low pressure.

- The Reinforced Concrete Containment Vessel (RCCV), as described in DCD, Tier 2, Revision 6, Sections 3.8 and 6.2 protects key design features located inside the RCCV from structural and fire damage.

- The location and design of the Reactor Building (RB) structure, including exterior walls, interior walls, intervening structures inside the building and barriers on large openings in the exterior walls, as described in DCD, Tier 2, Revision 6, Section 3.8 protects the RCCV from impact by a large, commercial aircraft.

- The location and design of the TB structure, as described in DCD, Tier 2, Revision 6, Section 3.8 protects the adjacent wall of the RB from impact by a large commercial, aircraft.

- The location and design of the FB structure, as described in DCD, Tier 2, Revision 6, Section 3.8 protects the adjacent wall of the RB from impact by a large, commercial aircraft.

- The location and design of fire barriers inside the RB, as described in DCD, Tier 2, Revision 6, Sections 9.5.1, 9A, and 19D protects credited core cooling equipment from fire damage.

- The location (below grade) and design of SFP structure, as described in DCD, Tier 2, Revision 6, Section 1.2, Figures 1.2-1 – 1.2-20 protects the SFP from impact by a large, commercial aircraft.

19.2.7.2.2 Description of How Regulatory Acceptance Criteria are Met

The acceptance criteria in 10 CFR 50.150(a)(1) are: (1) the reactor core will remain cooled or the containment will remain intact; and (2) SFP cooling or SFP integrity is maintained. The applicant has met 10 CFR 50.150(a)(1) by including features in the ESBWR design that maintain core cooling and SFP integrity following the impact of a large, commercial aircraft.

As indicated in DCD, Tier 2, Revision 6, Section 19D, the applicant proposes to maintain core cooling using the safety-related systems described in DCD, Tier 2, Revision 6, Section 19D which have been designed specifically to ensure that the reactor can be shutdown and decay heat can be removed adequately from the reactor core. Some of this equipment is located inside the RCCV and some is located inside the RB. Locations inside the RCCV are protected from structural, shock and fire damage by the design of the RCCV structure as well as the RB structure which limits the penetration of a large, commercial aircraft such that the RCCV is not perforated. Equipment inside the RB is protected by structural design features of the RB itself and by structures adjacent to the RB, including the TB and the FB. In addition, fire barriers have been designed and located in the RB to contain the spread of fire inside the building such that at least one train of safety-related equipment for core cooling is protected for each RB impact scenario.

The ESBWR satisfies the SFP integrity acceptance criterion in 10 CFR 50.150(a)(1) due to the location of the SFP. The SFP structure is located below ground which protects the structure from impact by a large, commercial aircraft.

19.2.7.3 Staff Evaluation

The staff has reviewed the description of key design features provided by the applicant and the description of how the key design features show that the acceptance criteria in 10 CFR 50.150(a)(1) are met. The staff's evaluation is provided below.

19.2.7.3.1 Reasonably Formulated Assessment

The applicant states in DCD, Tier 2, Revision 6, Section 19D that their aircraft impact assessment is based on the guidance of NEI 07-13, Revision 7. Based on the applicant's use of NRC endorsed guidance document NEI 07-13, Revision 7, the staff finds that the applicant has performed a reasonably formulated assessment.

19.2.7.3.2 Key Design Features for Core Cooling

The key design features listed in DCD, Tier 2, Revision 6, Section 19D that perform a core cooling related function are all safety-related design features that have been designed specifically to perform the core cooling functions during normal power operation and following design-basis events initiated during power operation. The staff has considered the descriptions of the features, as well as staff reviews documented in other sections of this report of the ability of these features to perform their design basis safety functions, in order to confirm that they are suitable for maintaining core cooling following impact of a large, commercial aircraft. During its review, the staff confirmed that all of these design features can be initiated and operated from the control room and require little, if any, further operator intervention to maintain the core cooling function.

In its initial review of the descriptions provided by the applicant, the staff noted that the applicant did not include a description of design features nor functional capabilities relied upon to ensure that the acceptance criteria in 10 CFR 50.150(a)(1) are met while the plant is shutdown and the reactor core is being cooled via the SDC system. In RAI 19.5-15 the staff requested that the applicant: (1) describe those design features and/or functional capabilities relied upon to ensure that the acceptance criteria in 10 CFR 50.150(a)(1) are met while the plant is shutdown and the reactor core is being cooled via the SDC system; (2) describe how these design features and/or functional capabilities meet the acceptance criteria in 10 CFR 50.150(a)(1); and (3) modify DCD, Tier 2, Revision 6, Section 19D, to include these descriptions. In their response the applicant proposed a modification to DCD ,Tier 2, Revision 6, Section 19D, which states that when normal cooling systems are not available following impact of a large, commercial aircraft, the ICS serves as a key design feature for core cooling when the plant is shutdown and the reactor is in Mode 5 and the Gravity Driven Cooling System (GDCS) serves that function when the reactor is in Mode 6. The staff has evaluated use of the ICS and GDCS to provide core cooling in these modes in Section 19.1.6 of this report and found it to be acceptable. Based on the applicant's response, RAI 19.5-15 was resolved. The staff accepts the applicant's proposed revision of DCD, Tier 2, Revision 6, Section 19D. Therefore, RAI 19.5-15 is resolved.

During its initial review of the ICS description, the staff noted that the pools of water used for cooling the IC condensers are not identified as part of the ICS. In RAI 19.5-20 the staff requested that the applicant state whether or not the isolation condenser/passive containment cooling system (IC/PCCS) water pools were considered key design features, since these pools are needed for the ICS to successfully remove decay heat from the core. In their response the applicant stated that these pools were considered key design features and proposed a modification to DCD, Tier 2, Revision 6, Section 19D, which includes a statement that the IC/PCCS pools are key design features. The staff accepts the applicant's proposed

modification to DCD, Tier 2, Revision 6, Section 19D. Based on the applicant's response, RAI 19.5-20 was resolved. Therefore, RAI 19.5-20 is resolved.

The staff noted during its initial review of the ICS description that the ICS in combination with the IC/PCCS pools is designed to remove decay heat for a period of 72 hours following a design basis event without operator intervention to refill the pools. In RAI 19.5-21, the staff asked the applicant to clarify whether or not the IC and its heat sink were capable of ensuring core cooling following a beyond-design-basis large, commercial aircraft impact event until measures for long term cooling could be established. In their response the applicant stated that the inventory of water available in the IC/PCCS pools following a beyond-design-basis aircraft impact event had been calculated for beyond-design-basis event conditions and is sufficient to allow measures for long term cooling to be established. The staff finds the applicant's response acceptable. Based on the applicant's response, RAI 19.5-21 was resolved. Therefore, RAI 19.5-21 is resolved.

19.2.7.3.3 Key Design Features that Protect Core Cooling Design Features

19.2.7.3.3.1 Fire Protection

The fire protection key design features that protect core cooling key design features include the fire-rated barriers located within the RB as described in DCD, Tier 2, Revision 6, Sections 9.5.1, 9A, and 19D and Figures 9A.2-1 through 9A.2-11. The applicant states the design and locations of the credited fire barriers confine the spread of fire damage resulting from a large, commercial aircraft impact. Specifically, the applicant states the fire damage is confined to either safety divisions one and three (east side) or safety division two and four (west side) and at least one division of safety-related controls remain available for core cooling.

The staff noted during its initial review of the key design features descriptions provided by the applicant that a clarification was required concerning locations of the credited fire protection features. In RAI 19.5-18 the staff requested that the applicant clarify if there were any fire protection-related key design features in the FB and CB. In their response and proposed modification to DCD, Tier 2, Revision 6, Section 19D, the applicant states there are no fire protection related key design features within the FB, CB or any other site building. The staff finds this response acceptable. Based on the applicant's response, RAI 19.5-18 was resolved. Therefore, RAI 19.5-18 is resolved.

The staff also noted during its initial review of the key design features descriptions provided by the applicant, that DCD, Tier 2, Revision 6, Section 19D did not contain adequate identifications or descriptions of the fire barriers within the RB. In RAI 19.5-19 the staff requested that the applicant provide the overpressure capabilities for each fire door (i.e., 5 psid rated or regular fire door) that is a key design feature. The applicant responded to the staff's RAI. However, the applicant's response was inadequate as it did not provide the overpressure capability of each fire door, individually, nor include all the credited fire protection key design features. The applicant's revised response clarifies that all fire barriers between the east side (safety divisions 1 and 2) and the west side (safety divisions 3 and 4) are to be credited key design features. The response included an addition to DCD, Tier 2, Revision 6, Section 19D of Table 19D-1 that lists each credited fire door required to be at least 5 psid. The applicant also clarified that all elements of the fire barriers, including walls, doors, penetration seals, that are used to protect the core cooling equipment can withstand a differential pressure of 5 psid where necessary. The applicant also identified additional key design features such as the fire doors throughout the RB stairways, the refueling floor (elevation 34000), and credits the location of the RB HVAC system in a proposed modification to DCD,Tier 2, Revision 6, Section 19D. The applicant states

that the stairway doors and refueling floor (elevation 34000) protect the required core cooling equipment, located below the refueling floor, from the spread of a fire caused by a large, commercial aircraft impacting the refueling floor. In addition, the applicant states RB HVAC trains do not penetrate the walls separating the east and west sides of the RB thus eliminating any penetrations and flow paths for the fire damage to spread to either the east side or west side of the RB. The staff accepts the final response and the proposed changes to DCD, Tier 2, Revision 6, Section 19D. The staff finds the applicant's description of the fire protection key design features for maintaining core cooling to be adequate. Based on the applicant's response, RAI 19.5-19 was resolved. Therefore, RAI 19.5-19 is resolved.

19.2.7.3.3.2 Reinforced Concrete Containment Vessel Structure

In DCD, Tier 2, Revision 6, Section 19D, the applicant states that the RCCV is a key design feature that would provide physical protection to the safety systems located inside the RCCV. The staff reviewed DCD, Tier 2, Revision 6, general arrangement drawings (Figures 1.1-1, 1.2-1 through 1.2-20) and Section 3.8 information. The applicant states that the RCCV is entirely surrounded by the RB structure and, therefore, a direct impact on the RCCV of a large, commercial aircraft is not possible. Based on its review, the staff finds the applicant's description of the RCCV as a key design feature for protecting safety systems inside the RCCV to maintain core cooling to be adequate.

19.2.7.3.3.3 Reactor Building Structure

In DCD, Tier 2, Revision 6, Section 19D, the applicant states that the location and design of the RB structure are key design features that protect the RCCV from the impact of a large, commercial aircraft. The staff reviewed DCD, Tier 2, Revision 6, general arrangement drawings (Figures 1.1-1, 1.2-1 through 1.2-20) and Section 3.8 information. During review of this information, the staff noted that there were openings on the RB refueling floor that could be subjected to secondary impacts (e.g., debris) from a large, commercial aircraft impact. To address this concern, the staff issued RAI 19.5-17 in which the applicant was requested to state whether secondary impacts were considered in the assessment of structural damage to the refueling floor. The applicant responded to the staff's request by stating the analysis of aircraft impacts to the refueling floor considers openings that may be subjected to secondary impacts. Further, the applicant stated that acceptance criteria listed in DCD, Tier 2, Revision 6, Section 19D (from 10 CFR 50.150(a)(1)), are met and that the DCD, Tier 2, Revision 6, Section 19D would be revised to provide additional information relative to secondary impacts. Based on the applicant's RAI response and proposed DCD revision, the staff considers RAI 19.5-17 resolved. Therefore, RAI 19.5-17 is resolved.

The staff noted during its initial review of the ICS description that the IC/PCCS pools, equipment storage pool and ICS heat exchangers are located outside of the RCCV, but inside the RB structure. In addition, the staff found that the description provided did not contain sufficient detail to confirm that three of the four ICS heat exchangers (minimum required for successful heat removal) and the inner and outer expansion pools that provide heat exchanger cooling are adequately protected. In RAI 19.5-20 the staff asked the applicant to describe the structures that protect the ICS heat exchangers and water pools and how such protection ensures that three of the four ICS heat exchangers and sufficient water is available to remove decay heat. In their response the applicant described inner and outer expansion pools located on opposite sides of the RB and indicated that both sides could not be damaged simultaneously by a large, commercial aircraft impact. They also stated that cross-connect valves between the equipment storage pool and the inner expansion pools are located in wells that protect them from damage.

In addition, the applicant stated that check valves prevent loss of water from the inner expansion pool to its adjacent outer expansion pool. In addition the applicant stated that the aircraft impact assessment considers the potential for loss or diversion of pool water due to damage caused by a large, commercial aircraft impact and found that sufficient water is retained to provide adequate core cooling. The staff finds that the additional description of the design of the heat sink for the ICS heat exchangers, including features that prevent the loss or diversion of water from the inner expansion pools that directly support the heat exchangers, is adequate. Based on the applicant's response, RAI 19.5-20 was resolved. Therefore, RAI 19.5-20 is resolved.

The staff finds the applicant's description of the RB as a key design feature for providing physical protection for maintaining core cooling to be adequate.

19.2.7.3.3.4 Turbine Building and Fuel Building Structures

In DCD, Tier 2, Revision 6. Section 19D, the applicant states that the location and design of the TB and FB structures, as shown in DCD, Tier 2, Revision 6 general arrangement drawings (Figures 1.1-1, 1.2-1 through 1.2-20), are key design features that protect the RB from the impact of a large, commercial aircraft. The staff finds the applicant's description of the key design features for providing physical protection to the RB for maintaining core cooling to be adequate.

19.2.7.3.4 Integrity of the Spent Fuel Pool

The key design feature credited to maintain the integrity of the SFP is the location of the SFP structure as described in DCD, Tier 2, Revision 6, Figures 1.2-1 though 1.2-4. The applicant states that the SFP structure is located entirely below grade and therefore, the SFP is protected from the impact of a large, commercial aircraft. The staff finds that the description of the key design feature for ensuring SFP integrity is adequate.

19.2.7.4 Conclusions

The staff finds that the applicant has performed an aircraft impact assessment that is reasonably formulated to identify design features and functional capabilities to show, with reduced use of operator action, that the acceptance criteria in 10 CFR 50.150(a)(1) are met. The staff finds that the applicant adequately describes the key design features credited to meet 10 CFR 50.150, including descriptions of how the key design features show that the acceptance criteria in 10 CFR 50.150(a)(1) are met. Therefore, the staff finds that the applicant meets the applicable requirements of 10 CFR 50.150(b).

19.2.8 Resolution of Generic Safety Issues

19.2.8.1 Generic Letter (GL) 89-16 Installation of Hardened Wetwell Vent

GL 89-16 describes the safety benefits of installing a fixed vent pipe in the wetwell of boiling water reactors with a MARK I containment design and requests each licensee operating a BWR with a MARK I plant provide notification of its plans to install a hardened wetwell vent.

The ESBWR design does not include a MARK I containment design. However, a wetwell vent is part of the ESBWR design. The staff evaluated its effectiveness in Section 19.1.4.2.3 of this

report and found that the wetwell vent can be an effective means of averting containment failure, should it be needed.

Inclusion of a hardened wetwell vent in the ESBWR design adequately resolves the issues addressed in GL 89-16.

19.2.8.2 TMI Action Plan Item II.B.8: Rulemaking Proceedings on Degraded Core Accidents

Item II.B.8 discusses the need to establish policy, goals, and requirements to address accidents resulting in core damage greater than the existing design basis. The Commission expects that new designs will achieve a higher standard of severe accident safety performance than previous designs. In an effort to provide this additional level of safety in the design of advanced nuclear power plants, the NRC developed guidance and goals for designers to strive for in accommodating events that are beyond what was previously known as the design-basis of the plant.

For advanced passive nuclear power plants, like the ESBWR, the staff concludes that vendors should address severe accidents during the design stage to take full advantage of the insights gained from probabilistic safety assessments, operating experience, severe accident research and accident analysis by designing features to reduce the likelihood that severe accidents will occur and, in the unlikely occurrence of a severe accident, to mitigate the consequences of such an accident. Incorporating insights and design features during the design phase is much more cost effective than modifying existing plants. The NRC issued guidance for addressing severe accidents in the following documents:

- NRC Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," issued August 8, 1985;
- NRC Policy Statement, "Safety Goals for the Operation of Nuclear Power Plants," issued August 4, 1986;
- NRC Policy Statement, "Nuclear Power Plant Standardization," issued September 15,1987;
- 10 CFR Part 52, " Licenses, Certifications, and Approvals for Nuclear Power Plants";
- SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, and the corresponding SRM dated June 26, 1990;
- SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, and the corresponding SRM dated July 21, 1993.

The NRC policy statements provide guidance as to the appropriate course for addressing severe accidents, 10 CFR Part 52 contains general requirements for addressing severe accidents, and the SRMs relating to SECY-90-016 and SECY-93-087 offer Commission approved positions for implementing features in new designs for preventing severe accidents and mitigating their effects.

SECY-93-087 and 10 CFR Part 52 serve as the basis for resolving severe accident issues associated with the ESBWR. 10 CFR Part 52 requires (1) compliance with the TMI requirements in 10 CFR 50.34(f), (2) resolution of USIs and GSIs,and (3) completion of a design-specific PRA. The staff evaluates these criteria in Sections 20 and 19 of this report, respectively.

The Commission-approved positions on the issues discussed in SECY-93-087 form the basis of the staff's deterministic evaluation of severe accident performance for the ESBWR design. The staff evaluates the ESBWR design relative to these criteria in Section 19.2 of this SER. Issue II.B.8 is resolved for the ESBWR design on the basis of the staff's evaluation of the probabilistic and deterministic analyses in the ESBWR PRA, as documented above.

19.2.8.3 Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities

The NRC issued GL 88-20 in November 1988, requesting that all reactor licensees perform an IPE to identify plant-specific vulnerabilities to severe accidents and report the results to the Commission. Each licensee developed a plant-specific PRA and used it to perform the requested IPE.

GEH has developed a PRA for the ESBWR and used it to identify vulnerabilities to severe accidents and evaluate alternative ways to eliminate such vulnerabilities. The results are documented in DCD, Tier 2, Revision 6, Section 19. The NRC staff has reviewed GEH's application of its PRA in the identification and elimination of severe accident vulnerabilities and found it acceptable. The staff's evaluation is documented in Section 19 of this SER. This resolves the issues addressed by GL 88-20 for the ESBWR.

19.2.8.4 Generic issue 157: Containment Performance

The results of NRC-sponsored research which culminated in the assessment of risk at five U.S. nuclear reactors in the late 1980s indicated that, for the Peach Bottom boiling water reactor, the core-melt probability was relatively low. However, it also indicated that the containment could be severely challenged if a large core-melt occurred. The Peach Bottom design includes the MARK I containment design. Consequently, the NRC decided to examine MARK I plants for potential plant and containment modifications to improve containment performance. Subsequently, this examination was expanded to include all other types of containment utilized at nuclear power plants regulated by the NRC. These studies were conducted under the Containment Performance Improvement (CPI) program. In some cases, these studies revealed highly beneficial design improvements (see discussion of the hardened wet well vent above in Section 19.2.8.1 of this SER.)

GEH has performed probabilistic and deterministic assessments of ESBWR containment performance and documented them in DCD, Tier 2, Revision 6, Section 19B and 19C. The staff reviewed these assessments and documented its results in Section 19.2.4 of this SER. They found that the applicant's containment performance evaluation meets 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 pressure fragility analysis. This resolves the issues addressed by Generic Issue 157 for the ESBWR.

19.2.9 Conclusion

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 reflect the applicant's efforts to systematically minimize the effect of initiators and sequences that have been important contributors to CDF in previous BWR PRAs. The applicant achieved this minimization largely through the incorporation 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 many plant features aimed at reducing CDF and risk, the benefits and risk reduction potential of additional plant improvements are significantly reduced. This reduction applies to 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 limits of probabilistic techniques.

The NRC staff evaluated the applicant's severe accident evaluation and identified several issues that were not adequately addressed. The applicant has now addressed all of these issues adequately through its responses to the staff's RAIs and the follow-up activities identified in the audit report. The staff has described each open issue and the basis for resolution of the issue in the appropriate section of this report. Based on its review, the staff finds that the applicant has adequately addressed the Commission's objectives, described above in Section 19.1.1, regarding the prevention and mitigation of severe accidents.

19.3 References

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code

— — — — —, Section II, “Materials,” New York, NY, 1995.

— — — — —, “Cases of ASME Boiler and Pressure Vessel Code,” CASE N-284-1, New York, NY, 1995.

— — — — —, Section III, “Rules for Construction of Nuclear Power Plant Components,” Divisions 1 and 2, New York, NY, 2004.

— — — — —, ASME-RA-Sb-2005, “Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,” New York, NY, 2005.

Electric Power Research Institute (EPRI)

— — — — —, EPRI, Nuclear Safety Analysis Center, NSAC-60, “Oconee PRA: A Probabilistic Risk Assessment of Oconee, Unit 3,” July 1984.

— — — — —, “EPRI—ALWR Utility Requirements Document,” Revision 4, April 1992.

GE-Hitachi Nuclear Energy Americas LLC (GEH)

— — — — —, “General Electric Company Application for Final Design Approval and Design Certification of ESBWR Standard Plant Design,” August 2005, (ADAMS Accession No. ML052490334).

— — — — —, NEDO-33201, Revision 1, “ESBWR Probabilistic Risk Assessment,” October 2006, (ADAMS Accession No. ML063310371).

— — — — —, NEDO-33306, Revision 1, “ESBWR Severe Accident Mitigation Design Alternatives,” August 2007, (ADAMS Accession ML072390051).

— — — — —, NEDE-33386, Revision 0, “ESBWR Plant Flood Zone Definition Drawings and Other PRA Support Information,” September 2007, (ADAMS Accession No. ML072820483).

— — — — —, “ESBWR Standard Plant – Design – Revision 4 to Design Control Document – Tier 1 and Tier 2,” September 2007, (ADAMS Accession No. ML072900510.

— — — — —, NEDO-33201, Revision 2, “ESBWR Probabilistic Risk Assessment,” September 2007, (ADAMS Accession No. ML081210271).

— — — — —, NEDO-33392, Revision 2, “The MAC Experiments Fine Tuninng of the BIMAC Design,” March 2008.

— — — — —, NEDO-33201, Revision 4, “ESBWR Design Certification Probabilistic Risk Assessment,” September 2009, (ADAMS Accession No. ML092030211).

Institute of Electrical and Electronic Engineers (IEEE)

— — — — —, IEEE, Standard 384-92, “Standard Criteria for Independence of Class 1E Equipment and Circuits,” Piscataway, NJ, 1992.

Institution of Mechanical Engineers

— — — — —, “Asymmetric Plastic Buckling or Axi-symmetric Yielding,” Proc. of Institution of Mech. Engineers, Vol. 199, No. C3, 1985.

Nuclear Energy Institute

— — — — —, NEI 07-13, Revision 7, “Methodology for Performing Aircraft Impact Assessments for New Plant Designs,” May 2009, (ADAMS Accession No. ML09140054).

Nuclear Management and Resources Council

— — — — —, NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management,” December 1991.

U.S. Code of Federal Regulations

— — — — —, *Title 10, Energy*, Part 50, “Domestic Licensing of Production and Utilization Facilities.”

— — — — —, *Title 10, Energy*, Part 52, “Energy Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

U.S. Nuclear Regulatory Commission

Commission Papers

— — — — —, SECY-89-013, “Design Requirements Related to the Evolutionary Advanced Light Water Reactors,” January 19, 1989, (ADAMS Accession No. ML003707947)..

— — — — —, SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” January 12, 1990, (ADAMS Accession No. ML003707849).

— — — — —, SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” April 2, 1993, (ADAMS Accession No. ML0037080201).

— — — — —, SECY-97-168, “Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation,” July 30, 1997.

Federal Register notice

— — — — —, “Rules and Regulation” Federal Register, Vol. 74 No. 112, June 11, 2009, pp. 28111-28147

Generic Letter

— — — — —, Generic Letter 88-20, “Individual Plant Evaluation for Severe Accident Vulnerabilities – 10CFR§50.54(f)”, November 23, 1988.

NUREG-Series Report

— — — — —, NUREG/BR-0058, Revision 4, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission,” September 2004, (ADAMS Accession No. ML042820192).

— — — — —, NUREG/BR-0184, “Regulatory Analysis Technical Evaluation Handbook,” January 1997, (ADAMS Accession No. ML050190193),

— — — — —, NUREG-1560, “Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance,” October 1997.

— — — — —, NUREG/CR-2239, “Technical Guidance for Siting Criteria Development,” December 1982, (ADAMS Accession No. ML072320420).

— — — — —, NUREG/CR-3862, “Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments,” May 1985, (ADAMS Accession No. ML070570060).

— — — — —, NUREG/CR-4461, Revision1, “Tornado Climatology of the Contiguous United States,” April 2005, (ADAMS Accession No. ML051080477).

— — — — —, NUREG/CR-4780, Volume 1, “Procedures for Treating Common Cause Failures in Safety and Reliability Studies,” January 1989, (ADAMS Accession No. ML070570105).

— — — — —, NUREG/CR-5497, “Common-Cause Failure Parameter Estimations,” October 1998, (ADAMS Accession No. ML070580356).

— — — — —, NUREG/CR-5750, “Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995,” February 1999, (ADAMS Accession No. ML022200457)..

— — — — —, NUREG/CR-5801, “Procedures for Analysis of Common Cause Failure in Safety Analysis,” April 1993, (ADAMS Accession No. ML072410498).

— — — — —, NUREG/CR-6595, “An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events,” January 1999, (ADAMS Accession No. ML072710052).

— — — — —, NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” February 2007, (ADAMS Accession No. ML070650650).

— — — — —, NUREG/CR-6850, Volume 1, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” September 2005, (ADAMS Accession No. ML052580075).

— — — — —, NUREG/CR-6850, Volume 2, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” September 2005, (ADAMS Accession No. ML052580118).

— — — — —, NUREG/CR-6890, Volume 1, “Reevaluation of Station Blackout Risk at Nuclear Power Plants,” “Analysis of Loss of Offsite Power Events: 1986–2004,” December 2005, (ADAMS Accession No. ML060200477).

— — — — —, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” March 2007, (ADAMS Accession No. ML070660036).

— — — — —, NUREG-0800, Draft Revision 3, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 3.8.1, “Concrete Containment,” and Section 3.8.2, “Steel Components of the Reinforced Containment,” April 1996.

Regulatory Guide

— — — — —, Regulatory Guide 1.70, Revision 3, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” November 1978, (ADAMS Accession No. ML011340122).

— — — — —, Regulatory Guide 1.75, Revision 3, “Physical Independence of Electric Systems,” February 2005, (ADAMS Accession No. ML082120622).

— — — — —, Regulatory Guide 1.174, Revision 1, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” November 2002, (ADAMS Accession No. ML081690565).

— — — — —, Draft Regulatory Guide 1176, “Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts,” July 2009, (ADAMS Accession No. ML073170252).

— — — — —, Regulatory Guide 1.200, Revision 1, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” January 2007, (ADAMS Accession No. ML063170035).

Other NRC Documents

— — — — —, AEOD/S97-03“Special Study, Fire Events – Feedback of U.S. Operating Experience”, June 1997.

— — — — —, “Comment and Responses Fire Events – Update of Operating Experience, 1986 – 1999 (Draft)”, January 2002, (ADAMS Accession ML020450047).

— — — — —, Memorandum from Cruz, Jeffrey, NRC to Hamazehee, Hossein, NRC "Audit Report Regarding Economic Simplified Boiling Water Reactor Probabilistic Risk Assessment Models Used to Support Economic Simplified Boiling Water Reactor Design Certification," June 23, 2009, (ADAMS Accession No. ML0917500092).

Individual Authors

— — — — —, Shield, R.T., and D.C. Drucker, "Design of Thin-Walled Torispherical and Toriconical Pressure-Vessel Heads," Journal of Applied Mechanics, Transactions of ASME, American Society of Mechanical Engineers, June 1961.

— — — — —, Galletly, G.D., and S.K. Radhamohan, "Elastic-Plastic Buckling of Internally-Pressurized Thin Torispherical Shells," Journal of Pressure Vessel Technology, American Society of Mechanical Engineers, Vol. 101, August 1979.

— — — — —, Galletly, G.D., and J. Blachnut, "Torispherical Shells Under Internal Pressure—Failure Due to Asymmetric Plastic Buckling or Axisymmetric Yielding," Proceedings of the Institution of Mechanical Engineers, Vol 199, No C3, pp 225-238, 1985.

— — — — —, Galletly, G.D., "A Simple Design Equation for Preventing Buckling in Fabricated Torispherical Shells under Internal Pressure," Journal of Pressure Vessel Technology, American Society of Mechanical Engineers, Vol. 108, November 1986.

— — — — —, Energy Research, Inc., ERI/NRC-93-203, "An Assessment of Ex-Vessel Fuel-Coolant-Interaction Energetics for General Electric Advanced Boiling Water Reactor," February 1993.

— — — — —, Energy Research, Inc., ERI/NRC-06-202, "Analysis of Ex-Vessel Fuel Coolant Interactions for ESBWR," July 2006.