15. TRANSIENT AND ACCIDENT ANALYSIS

15.1 Introduction

In the design control document (DCD) Tier 2, Revision 3, Chapter 15, "Safety Analyses," the applicant discussed the analysis of various design-basis anticipated operational occurrences (AOOs) and accidents. The staff reviewed the economic simplified boiling-water reactor (ESBWR) transient and accident analyses in accordance with Chapter 15, "Accident Analysis," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP), Draft Revision 3, issued April 1996. The staff used the later version of SRP Section 15.0, "Introduction—Transient and Accident Analyses," Revision 3, issued March 2007, only for defining different event categories.

Licensing Topical Reports (LTRs) NEDE-33338P, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis" and NEDC-33337P, "ESBWR Initial Core Transient Analysis" were recently submitted by the applicant and are currently being reviewed by the staff. These analysis will be addressed in the final SER and may result in additional open items.

15.1.1 Event Categorization

The standard review plan (SRP) divides events into AOOs and postulated accidents. The requirements of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the Code of Federal Regulations (10 CFR Part 50) define AOOs as conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.

SRP Section 15.0, Revision 3, defines postulated accidents as "Unanticipated conditions of operation (i.e., not expected to occur during the life of the nuclear power unit)."

General Electric Hitachi Nuclear America, LLC (GEH), the applicant, proposed a new subcategory of events - infrequent events (IEs) - under the broad category of accidents. GEH proposed this recategorization of events because of the unique passive cooling design of the ESBWR, the anticipated frequency of events occurrence, and the unique design features, such as the four divisions of safety systems and three channels of control systems which are redundant and fault tolerant. These design features could reduce the frequency of design-basis events (DBEs). As part of the proposal, GEH submitted topical report NEDO-33175, "Classification of ESBWR Abnormal Events and Determination of Their Safety Analysis Acceptance Criteria," Revision 1, issued February 2005. In this report, GEH reviewed the regulatory criteria for event classification for the ESBWR passive plant design to determine the appropriate abnormal event classifications and their associated safety analysis acceptance criteria. GEH provided additional information related to classification of events in its responses to U.S. Nuclear Regulatory Commission (NRC) staff requests for additional information (RAIs).

There may be new initiating events that require consideration within the scope of accidents and transients that result from the new and unique design features of the ESBWR. For example, the DCD does not include events such as inadvertent actuation of the control rod drive system (CRDS) in the injection mode to the reactor pressure vessel (RPV) or the gravity-driven cooling system (GDCS) inadvertent injection into the reactor vessel. The staff submitted RAI 15.0-1 to

identify all possible transients and accidents that may result from the unique design features of the ESBWR. In its response, GEH presented the results of a study confirming that all equipment in the ESBWR was reviewed to determine whether credible failures in the system or operator errors could initiate a DBE. GEH performed the study, which covered all the ESBWR systems and addressed possible new events resulting from the unique design features of the ESBWR. The staff found the evaluation performed by GEH to be acceptable. Therefore, RAI 15.0-1 is resolved. The evaluation covered the following event categories:

- increase in heat removal by the secondary system
- decrease in heat removal by the secondary system
- decrease in reactor flow rate
- reactor reactivity and power distribution anomalies
- increase in reactor coolant inventory
- decrease in reactor coolant inventory
- radioactive release from a subsystem or component

The following sections evaluate four groups of DBEs - AOOs, accidents - IEs, design-basis accidents (DBAs), and special events.

15.1.1.1 Anticipated Operational Occurrences

AOOs are expected during the life of the plant and require analysis to ensure that they will not cause damage to either the fuel or the reactor coolant pressure boundary (RCPB) or lead to a worse plant condition. The designed lifetime of the ESBWR plant is 60 years. In its evaluation, GEH conservatively assumed the plant to operate for 100 years. The conservative definition of AOOs for the ESBWR includes events with a frequency less than or equal to 1.0×10^{-2} per reactor year (pry). The acceptance criteria for the AOOs, as given in the SRP, are the following:

- General Design Criterion (GDC) 10, “Reactor Design,” as it relates to the reactor coolant system (RCS) design having appropriate margin to ensure that the plant does not exceed specified acceptable fuel design limits (SAFDLs) during AOOs,

- GDC 13, “Instrumentation and Control,” which requires the availability of instrumentation to monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions to ensure adequate safety, as well as appropriate controls to maintain these variables and systems within prescribed operating ranges,

- GDC 15, “Reactor Coolant System Design,” as it relates to the RCS design having appropriate margin to ensure against breach of the pressure boundary during AOOs

- GDC 17, “Electric Power Systems,” as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components important to safety will function during normal operation, including AOOs, and to ensuring sufficient capacity and capability to prevent the reactor from exceeding SAFDLs and design conditions of the RCPB during an AOO,

- GDC 20, “Protection System Functions,” as it relates to the reactor protection system

(RPS) being designed to initiate automatic operation of reactivity control systems to ensure that the reactor does not exceed SAFDLs during AOOs,

- GDC 25, “Protection System Requirements for Reactivity Control Malfunctions,” which requires that the RPS design will ensure the reactor does not exceed SAFDLs in the event of a single malfunction of the reactivity control system,

- GDC 26, “Reactivity Control System Redundancy and Capability,” as it relates to the system providing reliable control of reactivity changes by accounting for the appropriate margin for malfunctions, such as stuck rods, to ensure that the reactor does not exceed SAFDLs during AOOs.

The specific criteria necessary to meet the requirements of the GDC include the following:

- The plant design should maintain fuel cladding integrity by ensuring that the minimum critical power ratio (MCPR) remains above the safety limit minimum critical power ratio (SLMCPR),

- The plant design should maintain pressure in the reactor coolant and main steam systems below 110 percent of the design value,

- AOOs should not lead into a worse situation without another failure or operator error.

It is the staff’s position that the Safety Limit Minimum Critical Power Ratio (SLMCPR) numerical value should be kept as a safety limit in the TS as in the BWR Standard Technical Specification (TS). The staff reviewed the applicant’s response to RAI 15.0-16, Supplement 1, and found it unacceptable.

The staff bases its position on the following:

- Allowing the removal of the SLMCPR eliminates regulatory control of core analysis issues and eliminates a mechanism for the staff to apply conditions that might be needed in some situations to ensure safety. The NRC previously considered and rejected the same request (i.e., removal of the SLMCPR from the TS) from the Boiling Water Reactor Owners Group and Exelon (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML043140475 and ML030520480),

- Use of TRACG for calculating the operating limit minimum critical power ratio (OLMCPR) is not an appropriate basis for removing the SLMCPR from the TS. In its response, GEH referred to the ESBWR TRACG methodology used for the ESBWR OLMCPR calculation. GEH states that this process allows for the direct calculation of the number of rods subject to boiling transition for a transient. GEH maintains that because the SLMCPR is not used to calculate the OLMCPR, it is appropriate not to include the SLMCPR in the TS as assurance that the ESBWR meets the SAFDLs. The staff does not find use of the TRACG methodology to calculate the OLMCPR to be an appropriate basis for excluding the SLMCPR from the TS.

The NRC has approved the TRACG methodology for calculating the OLMCPR in the past for BWR/2-6s, and licensees that currently use the TRACG methodology for calculating the OLMCPR must still have an SLMCPR TS:

- 10 CFR 50.36(d)(1)(i)A specifically states, "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity." The staff has interpreted this section as requiring that the values of the safety limits must remain in a licensee's TSs. The revised TS Section 2.1.1.2 (Revision 3) proposes to replace the MCPR safety limit values with a description of what the safety limit protects against (i.e., "greater than 99.9 percent of the fuel rods in the core would be expected to avoid boiling transition"). The proposed description is a fuel condition and is not an acceptance criterion. The staff does not believe that the proposed change is consistent with the staff's interpretation of 10 CFR 50.36(d)(1)(i)A since it is not a safety limit, but a criterion.

GEH responded that although using the ESBWR TRACG Fuel Cladding Integrity Safety Limit (FCISL) Reactor Core Safety Limit terminology ensures protection of the fuel cladding for AOOs, it is recognized that a separate lower bound on the steady-state MCPR (i.e., SLMCPR) protects the fuel cladding when the MCPR is not within its Limiting Condition for Operation (LCO) specification. A potential violation of the Reactor Core Safety Limit would only occur if the newly defined ESBWR SLMCPR is violated during steady-state operations, or if an AOO occurs when the MCPR is not within its LCO specification. For both of these situations, the process variable MCPR could be used. GEH provided a revised response updating the original RAI 15.0-16 response. The staff is reviewing GEH's revised response to RAI 15.0-16.
RAI 15.0-16 is being tracked as an open item.

The substantive requirements summarized above apply to every AOO analyzed in section 15.2. Evaluation of each AOO considers how the requirements are met.

15.1.1.2 Accidents

Infrequent Events

SRP Section 15.0, Revision 3, defines IEs as events that may occur during the lifetime of the plant. IEs are DBEs with a frequency of less than 1.0×10^{-2} pry and a radiological consequence less than that of most DBAs.

GEH submitted DCD Tier 2, Appendix 15A, "Event Probability Analyses," providing the determination of the event frequency of the IEs. The staff's evaluation of the event frequency determination is provided in Section 15.A of this report.

The applicant proposed including 16 events in this new category. These events include reactivity, power and pressure anomalies such as control rod withdrawal error, mislocation and misorientation of fuel bundles, and generator load rejection with total bypass failure.

Some of these events are traditionally designated as AOOs for current boiling-water reactors

(BWRs), and some of them are new events. SRP Chapter 15 does not identify the IE category as such, but the IE category approximates the accident category in the SRP. Since the acceptance criteria include radiological consequences, IEs are considered as accidents. Acceptance criteria for IEs are bounded by the same criteria that govern accidents. The acceptance criteria for IEs are the following:

- The plant maintains the reactor water level above the top of active fuel (TAF),
- The RCPB pressure is less than 1500 pounds per square inch gauge (psig) (American Society of Mechanical Engineers (ASME) Code Pressure Service Level C, 120 percent of the RCS design pressures),
- Radiological consequence is less than 2.5 rem total effective dose equivalent (TEDE), 10 percent of the radioactive material release allowed in 10 CFR 50.34(a)(1). The DCD states that 1000 fuel rods is a bounding number for the fuel damage that meets the 2.5 rem criteria,
- The plant maintains containment and suppression pool pressures and temperatures below their design values,
- Control room (CR) personnel do not receive a radiation exposure in excess of 5 rem TEDE for the duration of the event.

The relaxation of the acceptance criteria for less probable events follows the rationale that events assessed as having a high frequency of occurrence shall have a small consequence (protection of the SLMCPR) and events assessed as having a lower frequency can have a more severe consequence (i.e., fuel damage may occur, but radiological dose must fall within the limits set forth in 10 CFR 50.34, “Contents of Construction Permit and Operating License Applications; Technical Information”).

GEH proposed ASME Code Service Level C (120 percent of the design pressure) as the criteria for RCPB pressure. DCD Tier 2, Section 15.0.1.2(4) defines an accident as a postulated DBE not expected to occur during the lifetime of the plant and with radiological releases not to exceed the calculated exposure in 10 CFR 50.34(a). The DCD also states that an accident equates to ASME Code Service Level C or D acceptance criteria. The staff is not aware of such equivalency, except for anticipated transients without scram (ATWS), as GEH stated in its response to RAI 15.0-17. ASME Code service levels require justification on a case-by-case basis in a manner similar to ATWS, and GEH did not provide this justification in its response to RAI 15.0-17.

During an August 21, 2007 GE-NRC conference call, the NRC staff stated that the response to RAI 15.0-17 S01 was acceptable; however, the DCD should include a commitment to perform post-pressurization event inspections-testing, if any event caused an ESBWR reactor coolant system to exceed its ASME Code Service Level B pressure limit. GEH replied that the Tier 2 safety analyses demonstrates that no design bases event can cause an ESBWR reactor coolant system to exceed its ASME Code Service Level B pressure limit. However, ASME Code Section XI does require adequate inspections/testing to confirm the operability of the safety-related components potentially affected by the hypothetical pressurization event.

Therefore, the ASME Code is used as the basis for the requested post-event inspections. DCD Tier 2 Subsection 3.9.3.1.2 will be updated in DCD Tier 2, Revision 5 in response to this request. Staff accepts this response. RAI 15.0-17 is resolved. **RAI 15.0-17 is being tracked as a confirmatory item.**

In Revision 3 of DCD Tier 2, Section 15.0.1.2(3), the applicant states, "An infrequent event is defined as a DBE (with or without assuming a single active component failure or single operator error) with probability of occurrence of 1.0×10^{-2} pry and a radiological consequence less than an accident." In RAI 15.0-26, the staff indicated that the IE category is a subset of the accident category and that the radiological consequence is less than that of a DBA. On response, the applicant revised the above text to read "radiological consequence less than a design-basis accident." This is acceptable. Therefore, RAI 15.0-26 is resolved.

The staff is currently performing independent confirmatory calculations for limiting AOOs with the TRACE/PARCS computer code. The staff's evaluation of the applicant's analyses is provided in Section 21.6 of this report.

The substantive requirements summarized above apply to every IE analyzed in section 15.3. Evaluation of each IE considers how the requirements are met.

15.1.1.3 Design-Basis Accidents

SRP Section 15.0, Revision 3, defines DBAs as postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components. The acceptance criteria for accidents include radiological consequences less than 2.5 rem TEDE, 6.3 rem TEDE, or 25 rem TEDE radioactive material release allowed in 10 CFR 50.34(a)(1), depending on the accident-specific acceptance criterion in Chapter 15 of the SRP.

This category includes loss-of-coolant accidents (LOCAs). The acceptance criteria for the emergency core cooling system (ECCS) in the case of LOCAs, specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," are as follows:

- The peak cladding temperature must remain below 1204.4 °C (2200 °F),
- Maximum cladding oxidation - The calculated total oxidation of the cladding must nowhere exceed 17 percent of the total cladding thickness before oxidation,
- Total hydrogen generation must not exceed 1.0 percent of the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,
- The system must maintain the core in a coolable geometry,
- The system must maintain calculated core temperatures after successful initial operation of the ECCS at acceptably low levels and remove decay heat for the extended period of time required by the long-lived radioactivity remaining in the core.

Staff evaluation of compliance with 10 CFR 50.46 is included in Section 6.3 of this report.

The substantive requirements summarized above apply to every DBA analyzed in section 15.4. Evaluation of each DBA considers how the requirements are met.

15.1.1.4 Special Events

The special events category may include a common-mode equipment failure or additional failure beyond single-failure criteria. ATWS, station blackouts (SBOs), and safe-shutdown fires fall under the special events designation. The acceptance criterion for each of these special events derives from a specific event basis, coping, mitigation, and the acceptance criteria specified in the NRC regulations and the SRP.

15.1.1.5 Analytical Methods

TRACG4

GEH transient analyses used the TRACG evaluation model, described in licensing topical report (LTR) NEDC-33083P-A, "TRACG Application for ESBWR," issued March 2005, to analyze most of the AOOs and IEs in DCD Tier 2, Chapter 15. This LTR demonstrates calculations for ESBWR AOOs. However, this LTR contains no discussion of IEs analyses. GEH should confirm whether it is planning to submit additional information to qualify TRACG for analyzing IEs. **RAI 15.0-27 is being tracked as an open item**.

TRACG is a multidimensional, two-fluid reactor thermal-hydraulics analysis code with three-dimensional neutron kinetics capability. The code is designed to perform transient analyses in a realistic manner with conservatism added, as appropriate, via the input specifications. Section 21.6 of this report provides the staff's evaluation of the assumptions with respect to initial power, scram reactivity, reactivity coefficients, power profiles, and other parameters used in the analyses.

15.1.1.6 Non-Safety-Related Systems Assumed in the Analysis

In response to RAI 15.0-2, GEH submitted a table listing the non-safety-related systems and equipment used for mitigating transients and accidents described in DCD Tier 2, Chapter 15. In accordance with 10 CFR 50.36(d)(1), the technical specifications (TSs) must include the non-safety-grade equipment credited in the transient and accident analyses.

In its response to RAI 15.0-2, the applicant described the function of the CRDS as follows:

> Control Rod Drive System (CRDS): The high pressure makeup water function of this system is credited in several event scenarios as backup level control to feedwater. This function of CRDS is non-safety-related. If credit is not taken for the high pressure makeup water function of the CRDS, then the Isolation Condenser System and Gravity-Driven Cooling System would ensure acceptable inventory control.

In its response to RAI 16.2-33, GEH stated as follows:

Both the RAI 15.0-2 and the RAI 16.0-1 responses indicated that this function is not in the primary success path for mitigating transients and accidents because the safety-related isolation condenser and gravity-driven cooling system will ensure water inventory is maintained within the acceptance criteria for the applicable event even if the non-safety-related CRD system makeup water function failed.

The staff requested that GEH revise the DCD to include this information and to include the results of analysis that support this conclusion. The staff also requested the applicant to add a table in Section 15.0 of DCD Tier 2 listing the following non-safety-related equipment that is credited in the AOO, IE, and/or accident analyses:

- CRDS - makeup water CRDS (not included in the TSs),
- selected control rod run-in (SCRRI) (included in the TSs),
- fuel and auxiliary pool cooling system feedwater control system RC&IS (rod worth minimizer and automated thermal limit monitor (ATLM) are included in the TSs),
- steam bypass and control system (included in the TSs).

RAIs 15.0-2 and 16.0-1 are being tracked as open items.

15.1.1.7 Chapter 15 Loss of Offsite Power Assumptions

GEH addressed GDC 17 compliance with regard to DCD Tier 2, Chapter 15 analyses in its response to RAI 15.0-4 (MFN 06-173, June 16, 2006). For 72 hours, no safety-related function requires either offsite alternating current (ac) power or onsite emergency diesel generator ac power. After 72 hours, the analyses take credit for the non-safety-related direct current (dc) and ac power. No ESBWR accident analyses assume the availability of offsite power. The ESBWR AOO events do include the loss of offsite power event. Since the ESBWR has no reactor recirculation pumps that normally get power supply from off site, the loss of offsite power event is not a significant event for the ESBWR. Chapter 8 of this report includes a detailed evaluation of GDC 17.

15.2 Analysis of Anticipated Operational Occurrences

In the design control document (DCD) Tier 2, Revision 3, Chapter 15, “Safety Analyses,” the applicant discussed the analysis of various design-basis anticipated operational occurrences (AOOs).

GEH analyzed the following categories of AOOs:

- decrease in core coolant temperature,
- increase in reactor pressure,
- increase in reactor coolant inventory,
- decrease in reactor coolant inventory.

15.2.1 Decrease in Core Coolant Temperature

15.2.1.1 Loss of Feedwater Heating

15.2.1.1.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

The ESBWR can lose feedwater (FW) heating in at least two ways, including the closing of the heater to the steam extraction line and the FW bypassing around the heater. The ESBWR design is such that no single failure or operator error will cause loss of FW heating greater than 37.8 °C (100 °F). Loss of FW heating will increase moderation, which in turn will increase power. The feedwater control system (FWCS) monitors and compares FW temperature to a reference temperature and, when it detects increased subcooling, it alerts the operator and signals the reactor control and information system (RC&IS) to activate the selected control rod run-in (SCRRI) to automatically reduce reactor power and avert a scram. If the core power distribution is top peaked, the SCRRI is not able to totally suppress the power increase until the blades reach the upper part of the core. For cases in which the temperature difference does not reach the required set difference, the system does not activate the SCRRI and the transient is milder than described here. DCD Tier 2, Table 15.2-4 and DCD Tier 2, Figure 15.2-1 demonstrate the results of the analysis for this transient.

15.2.1.1.2 Technical Evaluation

Loss of FW heating increases core inlet subcooling, resulting in a core power elevation from increased moderation. The ESBWR design includes FWCS logic that monitors the temperature difference to a standard value and adjusts power by control rod insertion to reduce power and avoid reactor scram. For a single FW failure, the maximum ΔT is about 21.1 °C (70 °F) while the analysis conservatively assumes ΔT to be 37.8 °C (100 °F). The results indicate that the maximum change in critical power ratio (ΔCPR) is about 0.10, the MCPR remains well above the designated acceptance limit, the transient does not induce a more serious condition, and the instrument spans and setpoints do not affect the results.

In RAI 5.2-5, the staff asked the applicant to explain how the ESBWR, in the event of a partial failure of the SCRRI, would avoid violating local thermal limits or creating a core instability without shutting down the core. This RAI was based on DCD Tier 2, Revision 1, Figure 15.2-1e, which demonstrates the importance of the SCRRI insertion for mitigation of this transient. GEH has not responded to this RAI. **RAI 15.2-5 is being tracked as an open item.**

15.2.1.1.3 Conclusion

Due to the open item discussed above, the staff cannot finalize its conclusions regarding the acceptability of the results of the Loss of Feedwater Heating events.

15.2.2 Increase in Reactor Pressure

15.2.2.1 Closure of One Turbine Control Valve

15.2.2.1.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

The steam bypass and pressure control (SB&PC) system controls turbine valves. The DCD states that the SB&PC uses a triplicate digital control system and is not subject to a credible single failure. For the purposes of this analysis, the applicant assumed that one turbine control valve (TCV) inadvertently remains closed (i.e., fails) at full power. The SB&PC system will sense the pressure increase and will open the remaining TCVs to maintain pressure. However, this may not be sufficient, and pressure and power will increase until the reactor reaches high flux or high-pressure scram. Depending on the turbine steam admission design (i.e., full or partial arc), the flow through the remaining three TCVs is 95 percent for full arc and 85 percent for partial arc. Therefore, the partial arc case is more conservative, and the applicant analyzed it here. In addition, the analysis can assume fast or slow closure. In this analysis, GEH assumed full stroke time (either fast at 0.08 seconds or slow at 2.5 seconds) for closure at rated steamflow. DCD Tier 2, Tables 15.2-6 and 15.2-7 and DCD Tier 2, Figures 15.2-2 and 15.2-3 present the analytical results for the fast and slow closures, respectively. Both transients are in the pressure increase category.

In the fast closure case, there is a 120 percent power peak at around 1 second, followed by a 110 percent FW flow peak at about 6 seconds and achievement of steady state at about 10 seconds. At the steady state, the total flow divides, with approximately 85 percent flowing through the turbine and 15 percent flowing through the TCV. Core reactivity mirrors the power peak from void collapse. The MCPR remains above 1.34.

In the slow closure case, the transients are similar except that the power peak is about 106 percent at 3 seconds and 110 percent FW peak at about 6 seconds. Steady state occurs at about the same time and to the same levels of power and TCV flow. The MCPR stays well above 1.34, and the pressure shows little change in either the fast or slow closure case.

15.2.2.1.2 Technical Evaluation

Inadvertent closing of one TCV at full power creates a power spike of short duration; however, the associated pressure and MCPR changes are very small. The results of this transient meet the acceptance criteria because pressure and MCPR are well within the acceptance limits and the transient will not cause any other adverse consequence.

15.2.2.1.3 Conclusions

SRP Section 15.0 states that transient and accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the SAFDLs and RCPB pressure limits as a result of these events. Based on this finding, the NRC staff concludes that the ESBWR will meet the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.2.2.2 <u>Generator Load Rejection with Turbine Bypass System</u>

15.2.2.2.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO. Grid electrical disturbances could cause loss of load to the generator. To avoid damage to the turbine generator from overspeed, the TCVs must be able to close very quickly. TCV closure would increase vessel pressure, but the opening of the steam bypass valves prevents this. If there is no other failure, the steam bypass system will discharge the entire steamflow at full power. The SCRRI and select rod insert (SRI) will then insert control rods to shut down the reactor. DCD Tier 2, Table 15.2-8 lists the sequence of events for this transient, and DCD Tier 2, Figure 15.2-4 shows the calculated results. In the interim of the fast closing of the TCVs and opening of the TBVs, there is a sharp pressure pulse accompanied by a power generation pulse that lasts slightly over 1 second. At the same time, the SCRRI and SRI activate and the first SRI inserts, followed by the second, third, and fourth at 10, 20, and 30 second intervals. SCRRI insertion is complete at 111 seconds. At 140 seconds, the power level reaches 60 percent of rated power, and at 300 seconds the system establishes a new power level at 60 percent power and 45 percent FW flow.

15.2.2.2.2 Technical Evaluation

The most important requirement for this transient is to ensure that the system isolates the TG as fast as possible to avoid damage from overspeed. The SB&PC system generates signals for fast closure of the TCVs, with simultaneous opening of the steam bypass valves and activation of the SCRRI system to shut down the reactor. Assuming no other equipment failure, total power will peak at about 110 seconds, the FW flow will peak at about 100 seconds at about 115 percent, and, if there is partial rod insertion, core power will stabilize at about 75 percent with 55 percent FW and steamflow. If the turbine bypass valves (TBVs) operate as designed, no vessel pressurization will occur and RPV pressure will actually decrease. Rod reactivity insertion will counterbalance the void reactivity increase from void collapse and a small reactivity increase resulting from fuel temperature. The MCPR stays above 1.35. The reactor stabilizes.

The results provided in DCD Tier 2, Figure 15.2-4(a) shows a high and narrow power peak of less than a 1-second duration. The applicant has not calculated energy deposition to ensure acceptable fuel cladding interaction. In RAI 15.2-2, supplement 1, the staff requested the applicant to explain why it did not consider fuel energy deposition. In addition, the transient calculation terminated before the transient stabilized. The applicant committed to continue the analysis of this transient. GEH has not responded to this request. **RAI 15.2-2 is being tracked as an open item.**

15.2.2.2.3 Conclusion

Due to the open item discussed above, the staff cannot finalize its conclusions regarding the acceptability of the results of the Generator Load Rejection with Turbine Bypass System event.

15.2.2.3 <u>Generator Load Rejection with a Single Failure in the Turbine Bypass System</u>

15.2.2.3.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

Initiation of this transient is the same as that for generator load rejection with turbine bypass; that is, when it senses generator load rejection, the SB&PC system signals closure of the TCVs and opening of the steam bypass valves. In this case, the analysis assumes a single failure in the turbine bypass system. For conservatism, the analysis assumes the bypass capacity to be at 50 percent. DCD Tier 2, Figure 15.2-5 shows core parameter variation as a function of time, and DCD Tier 2, Table 15.2-9 lists the sequence of events.

The calculated results indicate that the bypass valves initiate opening at 0.02 seconds into the transient and the TCVs will be closed at 0.08 seconds. At 0.22 seconds, the system will sense inadequate bypass, and the control rods will initiate insertion.

For a short time (less than 1 second), steamflow decreases because of limited bypass, which increases pressure, moderation, and power. The power spike lasts less than a second, and steamflow stabilizes in under 5 seconds to less than 60 percent. Finally, rod insertion shuts down the reactor. The pressure reaches a peak value at about 2 seconds into the transient at 1130 pounds per square inch absolute (psia), which is below the safety/relief valve (SRV) lift setting. The MCPR remains well above 1.30, the operating safety limit, and the reactor reaches a lower power stable state.

15.2.2.3.2 Technical Evaluation

As discussed earlier, TBV failure is highly unlikely because the SB&PC system uses a triplicate digital controller. After the system detects inadequate turbine bypass, control rod insertion begins at about 0.40 seconds after transient initiation. The resulting pressure and thermal power pulses are less than a second in duration. Should high pressure compress the water to the L2 level for 10 seconds or more, the control rod drive (CRD) high-pressure makeup injection will activate. Should the low-level signal remain for 30 seconds, the main steam isolation valve (MSIV) and isolation condenser (IC) will activate. The vessel pressure remains within acceptable limits, the OLMCPR remains above 1.30, the operating safety limit, and the reactor reaches a lower power stable state; therefore, no fuel rods will be in boiling transition and the regulatory criteria will be met.

The results provided in DCD Figure 15.2-5(a) show a high and narrow power peak of less than a second duration. The applicant has not calculated energy deposition to ensure acceptable fuel cladding interaction. In RAI 15.2-2, supplement 1, the NRC staff requested the applicant why it did not consider fuel energy deposition. In addition, the transient calculation terminated before the transient stabilized. The applicant committed to continue the analysis of this transient. GEH has not responded to this request. **RAI 15.2-2 is being tracked as an open item.**

15.2.2.3.3 Conclusion

Due to the open item discussed above, the staff cannot finalize its conclusions regarding the acceptability of the results of the Generator Load Rejection with a Single Failure in the Turbine Bypass System event.

15.2.2.4 Turbine Trip with Turbine Bypass System

15.2.2.4.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

A variety of causes, such as vibrations, low condenser vacuum, and loss of turbine control fluid pressure, can initiate turbine trip. Upon turbine trip, the SB&PC system will open the bypass valves. At about 0.20 seconds into the transient, the SCRRI and SRI initiate rod insertion. At 1.0 second, SCRRI initiates insertion and the first SRI group inserts. Second, third, and fourth SCRRI groups initiate insertion at 10, 20, and 30 seconds, respectively. Control rod insertion is completed at 101 seconds. The reactor attains a steady state at about 60 percent of normal power with about 45 percent of normal FW flow. Peak FW flow occurs at 100 seconds at 140 percent of normal. Vessel pressure shows very small perturbation and falls to about 94 percent of the normal operating value. At the new steady state, about 45 percent of the steamflow removes about 60 percent of total power. The value of the OLMCPR remains well above the 1.30 operating limit. DCD Tier 2,Table 15.2-10 and DCD Tier 2, Figure 15.2-6 summarize the results of the calculation.

15.2.2.4.2 Technical Evaluation

After turbine trip, a fast rise in core pressure causes void collapse, increased moderation, and increased power in the form of a power pulse. Void collapse also causes a momentary increase in FW flow. The cold water slug entering the core also contributes to the power pulse. The calculated results show that control rod insertion compensates for the increased reactivity from increased moderation. The reactor reaches a steady state at a power level of about 60 percent and a corresponding FW flow of about 45 percent. The MCPR remains well above its designated safety limit.

15.2.2.4.3 Conclusion

Because of the fast opening of the bypass valves, the calculated results indicate that only a minor power disturbance occurs, no pressure surge takes place, the MCPR remains above the SLMCPR, and the reactor assumes a lower power stable state. Therefore, the results of this transient meet the acceptance criteria and regulatory requirements, and the transient is acceptable.

SRP Section 15.0 states that transient and accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the SAFDLs and RCPB pressure limits as a result of these events. Based on this finding, the NRC staff concludes that the ESBWR will meet the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.2.2.5 Turbine Trip with a Single Failure in the Turbine Bypass System

15.2.2.5.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

A variety of causes, such as vibrations, low condenser vacuum, and loss of control fluid pressure, can initiate turbine trip. Upon turbine trip, the SB&PC system will open the bypass valves and initiate TSV closure. In this transient analysis, the applicant assumed that the single failure would result in the loss of 50 percent of the bypass capacity. Since it would require more than a single bypass valve failure to lose 50 percent of the bypass capacity, this is a conservative assumption. The shortfall in bypass capacity creates a pressure pulse. There is a rapid increase in neutron power from pressure increase, void collapse, and increased moderation. The power pulse is about a third of a second at half-maximum and peaks at over 140 percent of rated power. There is a small vessel pressure peak at about 2.0 seconds into the transient which remains well below the SRV setpoint. The TSV closure initiates a reactor scram, and the opening of the bypass valves ameliorates the pressure pulse. The reactor is essentially shut down in less than 2.0 seconds. The transient calculation terminates at 10 seconds, when many parameters have not yet stabilized. Because the reactor is shut down, the staff concludes that the reactor is in a stable shutdown condition. The MCPR remains well above the SLMCPR. DCD Tier 2, Figure 15.2-7 and DCD Tier 2, Table 15.2-11 show the calculated results. The wide range (WR) water level remains above the L8 level, but the TSV closure also activates CRD injection to recover vessel water level. At 10 seconds (the end of the transient calculation), the water level is still falling. The expectation is that the RPV level will stabilize at a level above the TAF.

15.2.2.5.2 Technical Evaluation

Following turbine trip and inadequate bypass flow, the RCS pressure peaks, causing increased moderation and the creation of a power peak. The RPS initiates control rod insertion and the reactor (neutron) power falls to about 5 percent in less than 3 seconds. TSVs close. The vessel pressure remains well below the safety valve lift setpoint, and the MCPR stays well above the SLMCPR. The plant enters shutdown, with activation of CRD injection for vessel water level control. The reactor enters a safe-shutdown state, the pressure remains well below the safety limits, and the MCPR remains well above the designated safety limit value. Therefore, the results of this transient meet the acceptance criteria, and the transient is acceptable.

The results provided in DCD Tier 2, Figure 15.2-7(a) shows a high and narrow power peak of about a third of a second in duration. The applicant has not calculated energy deposition to ensure acceptable fuel cladding interaction. In RAI 15.2-2, the staff requested GEH to explain why it did not consider fuel energy deposition. In addition, the transient calculation terminated before the transient stabilized. The applicant committed to continue the analysis of this transient. GEH has not responded to this request. **RAI 15.2-2 is being tracked as an open item.**

15.2.2.5.3 Conclusion

Due to the open item discussed above, the staff cannot finalize its conclusions regarding the acceptability of the results of the Turbine Trip with a Single Failure in the Bypass System event.

15.2.2.6 Closure of One Main Steam Isolation Valve

15.2.2.6.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

One MSIV could close under testing conditions (i.e., below certain power levels) without reactor scram. However, at full power, one MSIV's inadvertent closure would cause closure of all MSIVs, leading to a reactor scram. In this case, the applicant assumed that one MSIV closure does not lead to reactor shutdown. DCD Tier 2, Figure 15.2-8 and DCD Tier 2, Table 15.2-12 show the calculated results.

The MSIV closure occurs within 3.0 seconds. However, at the initiation of the closing process, reactor pressure rises, which increases moderation and power. Neutronic power peaks at 2 seconds, but total power assumes a new steady-state level at about 101 percent of normal power with an FW flow of about 4 percent higher than normal. There is very little pressure vessel variation, and the MCPR remains well above the SLMCPR.

15.2.2.6.2 Technical Evaluation

Under full power conditions, the MSIV takes 3 seconds to close. During closure, power increases from the void collapse and increased moderation, but fuel temperature reactivity feedback will offset the increase and total reactivity will settle back to just critical. The calculated results show that the transient has little if any effect on vessel pressure and the MCPR will remain above 1.35; therefore, the regulatory requirements are met. In addition, this transient is bounded by the all-MSIV-closure transient, discussed in Section 15.2.2.7 of this report.

15.2.2.6.3 Conclusion

Closure of one MSIV is a minor perturbation in reactor operation without a serious challenge from either overpressure or OLMCPR. This conclusion assumes that the RPS will work as intended. The pressure stays very close to the operating range, the MCPR is well above the operating safety limit, and the transient assumes a stable steady state with the core fully covered and does not lead to another transient. Therefore, the results of this transient meet the acceptance criteria and the transient is acceptable.

SRP Section 15.0 states that transient and accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the SAFDLs and RCPB pressure limits as a result of these events. Based on this finding, the NRC staff concludes that the ESBWR will meet the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.2.2.7 Closure of All Main Steam Isolation Valves

15.2.2.7.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

As stated in Section 15.2.2.6 of this report, inadvertent closure of one MSIV at power levels above the testing power level will cause all of the MSIVs to close. In addition, low steamline pressure, high steamline flow, high steamline radiation, low water level, or manual action will activate MSIV closure. Total time for completion of MSIV closure is 3.0 seconds. DCD Tier 2, Figure 15.2-9 and DCD Tier 2, Table 15.2-13 show the calculated results for the evolution of the transient.

MSIV closure initiates a reactor scram on high flux. The same signal also initiates IC operation, which prevents lifting of the SRVs by lowering the RCS temperature and pressure. The applicant conservatively assumed MSIV closure to be achieved in 3 seconds, which is the shortest time in the MSIV closure range and thus would cause the highest pressure pulse. Vessel pressure reaches a maximum in 4.8 seconds at 7.9 megapascales (MPa) (1146 psig) while the lowest SRV setting is 8.62 MPa. Control rod insertion is completed within 4 seconds, and the MCPR reaches the lowest value at 1.25 seconds, which is well above 1.30 the operating limit minimum critical power ratio (OLMCPR). The FW flow decreases to about 75 percent of normal at about 4 seconds because of increased RCS pressure, while core flow increases temporarily to about 130 percent of normal. Because of the higher RCS pressure, the WR water level falls to about 6 feet above TAF. CRD operation initiates simultaneously with the scram signal to recover water level. Liquid flow to the ICs initiates at about 17 seconds and is at full flow at about 30 seconds.

15.2.2.7.2 Technical Evaluation

A variety of circumstances will result in the MSIV closure signal, which also activates rod insertion, IC initiation, and CRD injection. Assuming that the RPS operates as designed, rod insertion will dominate core reactivity. Within 1 second, the reactor becomes subcritical; therefore, void collapse and increased moderation have no effect on power level. Because the ICs and CRD injection initiate simultaneously with the reactor scram signal, there is no need for operator intervention. The OLMCPR remains well above 1.30, and the SRVs are not challenged; therefore, the regulatory requirements are met.

15.2.2.7.3 Conclusion

MSIV closure is a fast-evolving transient where rod insertion, IC initiation, CRD injection, and reactor water cleanup (RWCU) and shutdown cooling (SDC) initiate simultaneously. Rod insertion dominates the transient reactivity by suppressing the power elevation caused by increased moderation from the pressure pulse. The pressure increase is within the acceptable range, the OLMCPR is well above 1.30, and the reactor is shut down in a stable condition with the core covered. Thus, the results of this transient meet the acceptance criteria.

SRP Section 15.0 states that transient and accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the

analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the SAFDLs and RCPB pressure limits as a result of these events. Based on this finding, the NRC staff concludes that the ESBWR will meet the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.2.2.8 Loss of Condenser Vacuum

15.2.2.8.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

Failure to isolate the steam jet air injectors and loss of one or more circulating water pumps are common causes for loss of condenser vacuum. Sensing condenser vacuum, the RPS will initiate turbine trip and reactor scram. Turbine bypass will activate to regulate pressure, close the turbine stop valves, and close the MSIVs, and, when the system reaches the L2 water level, the high-pressure CRD injection initiates to control and restore water level. DCD Tier 2, Figure 15.2-10 and DCD Tier 2, Tables 15.2-14 and 15.2-15 present the calculated results.

As described in Section 15.2.2.7 of this report, control rod insertion dominates reactivity response; thus, pressure increase has no effect on power level via increased moderation. For the first 10 seconds, steamflow increases but so does FW flow. Sensed WR water level reaches a minimum in 20 seconds at about 6 feet above TAF. Because of the fast-acting instrumentation and the TBVs, the vessel pressure trends lower from the operating level. Similarly, as a result of the prompt control rod insertion, the OLMCPR remains at operating or higher level values.

15.2.2.8.2 Technical Evaluation

Failure to isolate the steam jet air injectors or loss of one or more condenser circulating water pumps will result in loss of condenser vacuum. Almost simultaneously, the TBVs begin to open to regulate RCS pressure. Reactor scram initiates, and the main TBVs open and initiate MSIV closure. MSIV closure elevates vessel pressure and lowers water level. The high-pressure CRD injection activates to restore water level. The MCPR remains well above the SLMCPR, and the high RCS pressure does not challenge the SRVs.

15.2.2.8.3 Conclusion

Loss of condenser vacuum leads to a series of fast actions by the RPS to scram the reactor, shut down the turbine, accommodate the existing steamflow, and ensure RPV water level. Assuming that the instrumentation and the appropriate valves respond according to their design, the vessel pressure remains below operating levels, the OLMCPR remains well above the SLMCPR, and the reactor is shut down with the core stable and covered. Therefore, the results of this transient meet the acceptance criteria, and the transient response is acceptable.

SRP Section 15.0 states that transient and accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the

analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the SAFDLs and RCPB pressure limits as a result of these events. Based on this finding, the NRC staff concludes that the ESBWR will meet the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.2.2.9 Loss of Shutdown Cooling Function of the Reactor Water Cleanup and Shutdown Cooling System

15.2.2.9.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

In the ESBWR the reactor water cleanup/shutdown cooling (RWCU/SDC) system is not a safety system. Nevertheless, it can provide for high- and low-pressure core cooling. The system consists of two trains with the necessary piping, heat exchangers, instrumentation, and power supply. In addition to the water cleanup function, the RWCU/SDC provides shutdown cooling where each train takes suction from the RPV and returns cooler water to the water supply line. Each train is supplied from offsite power, but if power is lost, each train has its own independent diesel power supply. In this manner, the system is single-failure proof.

15.2.2.9.2 Technical Evaluation

For the shutdown cooling function, each train has its own suction line from the RPV (unlike in the current reactors) and returns to the FW line. Thus, each of the RWCU/SDC trains is completely independent of the others. If the single active failure criterion is applied to the RWCU/SDC system, one of the RWCU/SDC trains would be inoperable. However, the operable RWCU/SDC train could achieve cold shutdown (less than 210 °F) within 36 hours after reactor shutdown.

15.2.2.9.3 Conclusion

Because the loss of shutdown cooling function of the RWCU/SDC is a very mild event and the reactor is not challenged by any transients, the applicant did not perform TRACG analyses. The acceptance criteria are satisfied.

15.2.3 Reactivity and Power Distribution Anomalies

15.2.3.1 Summary of Technical Information

DCD Tier 2, Revision 3, Section 15.2.3, states that there are no reactivity and power distribution anomaly AOOs identified for the ESBWR.

15.2.3.2 Technical Evaluation

In RAI 15.2-10, the staff questioned the applicant's conclusion that there are no reactivity and power distribution anomaly AOOs identified for the ESBWR. In response to RAI 15.2-10 S01,

the applicant states that the rod control and information system (RC&IS) controls fine motion CRDs (FMCRDs) employed in the ESBWR and rods are moved electrically. Mechanical failure of a single relay will not cause an inadvertent rod withdrawal error (RWE). Additionally, failure of the contact of a single switch will not cause RWE. GEH extends the argument to the electronic equipment being also redundant which at most will result in the inability to move the associated FMCRD by normal motor movement. The response describes several additional improvements in the FMCRDs to support the argument that RWEs are not likely and can only result from multiple failures. The response references DCD Tier 2, Sections 15A and 7.7.2. Staff accepts this response. RAI 15.2-10 is resolved.

15.2.3.3 Conclusions

The applicant's response to RAI 15.2-10 that there are no reactivity and power distribution anomaly AOOs identified for the ESBWR was accepted by the staff. The staff agrees that DCD Tier 2, Revision 3, Section 15.2.3, supports that there are no reactivity and power distribution anomaly AOOs identified for the ESBWR.

15.2.4 Increase in Reactor Coolant Inventory

15.2.4.1 Inadvertent Isolation-Condenser (IC) Initiation

15.2.4.1.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

In its analysis of inadvertent IC initiation, the applicant assumed that all IC units are inadvertently activated, therefore, that this is a bounding case. The system establishes full IC flow in 31 seconds, and power stabilizes in about 150 seconds. The analysis assumes that the system operates without additional failures. DCD Tier 2, Figure 15.2-11 shows the results, and DCD Tier 2, Table 15.2-17 depicts the sequence of events.

As the coolant flow through the core increases and coolant temperature decreases, the core power increases due to increased moderation. Control reactivity also increases and power level reaches a maximum at about 50 seconds into the transient. FW flow decreases accordingly to keep the vessel water at the same level. With power at about 115 percent and FW at about 88 percent of normal, the OLMCPR falls below the designated value of 1.30, to be recovered at about 155 seconds into the transient with power at 105 percent and FW flow almost normal. Control reactivity, total reactivity, and power, as expected, change at the same rate. Vessel pressure stays at the normal operating level. The MCPR value reaches 1.28 at about 130 seconds into the transient. The MCPR is above the SLMCPR.

15.2.4.1.2 Technical Evaluation

In this bounding case, the applicant assumed that all four ICs start injecting colder water into the vessel, increasing core water inventory and core moderation. The only reasonable assumption for the simultaneous initiation of all four ICs is inadvertent manual operator action. The transient proceeds relatively slowly with gradual increase in the thermal power and decrease in the FW flow. The calculated results indicate that in about 300 seconds the reactor

attains equilibrium operation at normal power with FW flow at about 90 percent of normal. The OLMCPR is recovered at a value above 1.32. Vessel pressure stabilizes at a slightly lower level than normal. The core remains fully covered and stable.

15.2.4.1.3 Conclusions

Inadvertent activation of all four IC units causes a bounding cold-water injection transient. From the above discussion, it is apparent that the pressure remains well within the acceptable limits, the MCPR stays above the safety limit (but slightly below the operating limit), the core remains fully covered, and the reactor returns to a stable state. Therefore, the results of this transient meet the acceptance criteria, and the transient is acceptable.

SRP Section 15.0 states that transient and accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the SAFDLs and RCPB pressure limits as a result of these events. Based on this finding, the NRC staff concludes that the ESBWR will meet the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.2.4.2 Runout of One Feedwater Pump

15.2.4.2.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

Three FW pumps are running during normal operation in the ESBWR. FW pumps are motor driven with variable speed motors. A runout transient consists of one pump increasing speed (and FW flow) to its maximum capacity. The FWCS uses a triplicate digital control system including a fault tolerant controller. The controller contains three parallel processing channels each with microprocessor-based hardware and associated software necessary to perform the control calculations. The operator interface provides system status and required control functions. The processor is capable of identifying faults and isolating faulty channels. There are two credible single failures that could lead to loss of control of one actuator for one FW pump with increasing flow. This is the case considered here. DCD Tier 2, Figure 15.2-13 and DCD Tier 2, Tables 15.2-18 and 15.2-19 show the calculated results.

When the system senses the increased flow, the FW controller will lower FW flow to the other two pumps so the total flow stays at the predetermined value with a minimal disturbance to the system. This occurs in about 21 seconds. The vessel pressure does not change perceptively. Fuel temperature and void reactivities change in opposite directions, resulting in small changes in total reactivity without control rod movement. FW flow changes equalize at about 40 seconds, and reactivity variations stabilize at about 100 seconds into the transient.

15.2.4.2.2 Technical Evaluation

This transient results in increased FW flow caused by a single FW pump runout. FW controller

action to reduce FW flow promptly compensates for increased FW, and the system achieves normal water level at 21 seconds into the transient according to the submitted analytical results. The increased water injection causes an increase in power which returns to normal when normal water level is achieved. The transient does not cause a scram, the OLMCPR remains well above the SLMCPR, and there is no perceptible variation in pressure.

15.2.4.2.3 Conclusion

Single FW pump runout creates a minor disturbance to reactivity, power, FW flow, vessel pressure, and reactivity components. The calculated results indicate that vessel pressure remains at normal operation level, the OLMCPR stays well above the SLMCPR, and the core returns to a fully covered and stable position. Therefore, the results of this transient satisfy the acceptance criteria, and the transient analysis is acceptable.

SRP Section 15.0 states that transient and accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the SAFDLs and RCPB pressure limits as a result of these events. Based on this finding, the NRC staff concludes that the ESBWR will meet the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.2.5 Decrease in Reactor Coolant Inventory

15.2.5.1 Opening of One Turbine Control or Bypass Valve

15.2.5.1.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

Instrumentation failure, such as actuator or voter failure, could cause inadvertent opening of a turbine bypass or a TCV. Such failure is highly unlikely. Because the SB&PC system has a triplicate control system (see Section 15.2.4.2 of this report), no credible single failure could result in TCV or TBV failure. Inadvertent operator action could also cause a TCV or a TBV to open. DCD Tier 2, Figure 15.2-14 shows the calculated results, and DCD Tier 2, Table 15.2-20 depicts the sequence of events.

The calculated results indicate that at the initiation of the transient there is a very short elevation in steamflow which increases the void fraction, causing a corresponding dip in power and dome pressure. However, the lower pressure increases the FW flow, which promptly increases moderation, and power recovers at about 30 seconds into the transient. At this time, the turbine steamflow reduces to about 82 percent of normal and the TCV flow remains at 15 percent. Regarding reactivity changes, there is a sharp dip in void reactivity, but the simultaneous increase in FW flow collapses some voids and total reactivity returns to normal at about 30 seconds into the transient. The vessel pressure remains almost unchanged from the normal operating value, the OLMCPR stays well above 1.30, and the reactor assumes a stable

condition while the fuel remains covered.

15.2.5.1.2 Technical Evaluation

In the event of an inadvertent or faulty opening of a TCV or an SBV, assuming the control and instrumentation will operate as designed, the SB&PC system will promptly compensate for the bypass steam, arrest the evolution of the transient, and return the reactor to a stable state.

DCD Tier 2, Table 15.2-20 and DCD Tier 2, Figure 15.2-14 demonstrate the evolution of the event.

There is a small increase in steamflow after transient initiation with a corresponding oscillation in FW flow and steady steamflow from the TCV or bypass flow. At about 30 seconds into the transient, the system establishes a new steady state. The vessel pressure varies by a small amount from the normal operating value, the OLMCPR remains well above the SLMCPR, and the reactor returns to a stable condition.

15.2.5.1.3 Conclusion

Opening of a TCV or bypass valve creates a minor disturbance, mainly as a result of the automated action of the control system to adjust bypass or turbine control flow. The vessel pressure is subjected to a very small change, the OLMCPR stays well above the operating safety limit, and the reactor achieves a stable and covered core steady state. Therefore, the results of this transient satisfy the acceptance criteria, and the transient analysis is acceptable.

SRP Section 15.0 states that transient and accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the SAFDLs and RCPB pressure limits as a result of these events. Based on this finding, the NRC staff concludes that the ESBWR will meet the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.2.5.2 <u>Loss of Nonemergency Alternating Current Power to Station Auxiliaries</u>

15.2.5.2.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

Loss of power to the station auxiliaries can result from lightning, storms, wind, load instabilities, loss of load, load rejection, or similar causes that could lead to failure of the unit auxiliary transformer. In this analysis, the applicant assumed that concurrent with load rejection, there is simultaneous loss of power on the four power generation buses, which will cause the FW and circulation pumps to be lost. Loss of the circulating water pumps results in loss of condenser vacuum, which in turn causes turbine trip. The bypass valves will be initially available, but loss of the power buses will produce a scram signal and initiation signal for the ICs and high-pressure CRD injection. With loss of the station transformer, the station emergency diesel

generators will activate to provide power to the CRD pumps. In summary, loss of the station auxiliary power will lead to reactor shutdown. DCD Tier 2, Table 15.2-21 shows the sequence of events, and DCD Tier 2, Figure 15.2-15 depicts the time-dependent variation of reactor parameters.

There is a delay time of 50 seconds for the main condenser loss of vacuum scram signal. Upon loss of load, there is a short decrease in FW flow followed by a very short power spike from increased moderation with the closing of the MSIVs. The power spike leads to the reactor scram. About 100 seconds after initiation, there is a sharp increase in IC water supply, which levels off in about 20 seconds.

15.2.5.2.2 Technical Evaluation

The pressure remains well below the AOO limit of 110 percent of the design value, the OLMCPR remains well above 1.30, and the reactor is shut down. CRD high-pressure injection controls the water level. CRD and IC injection ensure core cooling. Since the core is covered, in a stable state, and cooled, the staff concludes that the regulatory requirements are met.

15.2.5.2.3 Conclusion

Loss of all nonemergency ac power to station auxiliaries leads to turbine trip and reactor shutdown with IC and high-pressure CRD pump activation and injection. After a short pressure-and-power pulse, the vessel depressurizes and power reduces to zero. In the transition to shutdown, the OLMCPR remains well above the SLMCPR; therefore, the results of this transient meet the acceptance criteria, and the transient is acceptable.

SRP Section 15.0 states that transient and accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the SAFDLs and RCPB pressure limits as a result of these events. Based on this finding, the NRC staff concludes that the ESBWR will meet the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.2.5.3 <u>Loss of All Feedwater Flow</u>

15.2.5.3.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.1 are used in evaluating this AOO.

Loss of all FW flow could result from inadvertent operator errors, pump failure, or reactor trip signals. The sequence of events is similar to that described in DCD Tier 2, Section 15.2.5.2, “Loss of Non-Emergency Power,” in that the reactor subcooling decreases, causing reduction in core power, pressure, and water level. DCD Tier 2, Table 15.2-22 lists the sequence of events, and DCD Tier 2, Figure 15.2-16 shows the variation of reactor parameters as a function of time.

If the FW pumps trip, the ensuing reduction of FW flow will scram the reactor and initiate IC

operation. At about 5 seconds, the FW flow decays to zero, vessel water level drops to RPV Level 2, and the high-pressure CRD injection initiates at 20 seconds while the ICs reach full flow at 33 seconds and the MSIVs close at 40 seconds into the transient. At about 100 seconds, water level recovers to about 13 meters (43 feet) above TAF, and the core is stabilized.

15.2.5.3.2 Technical Evaluation

The prompt action by the RPS to scram the reactor and initiation of the ICs removes the possibility of core uncovery and ensures fast recovery of normal water level. During this transient, the vessel pressure quickly drops below normal operating values to about 70 percent of normal in less than 200 seconds.

The OLMCPR remains well above 1.30 and the reactor is shut down, with the core covered in a stable cooled state. Therefore, the results of this transient satisfy the acceptance criteria.

15.2.5.3.3 Conclusion

Loss of all FW flow results in a fast reactor shutdown and simultaneous IC and CRD high pressure injection activation. This transient violates none of the acceptance criteria and the reactor is shut down; therefore, the transient analysis is acceptable.

SRP Section 15.0 states that transient and accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the SAFDLs and RCPB pressure limits as a result of these events. Based on this finding, the NRC staff concludes that the ESBWR will meet the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.3 Analysis of Infrequent Events (IEs)

This section covers the material in DCD Tier 2, Section 15.3. IEs are defined as the events with an expected frequency of less than 10^{-2} pry of operation. The expected frequency for these events is in DCD Tier 2, Section 15A.3. The staff's evaluation of the event frequency determination is provided in Section 15A of this report.

The staff based this review on the acceptance criteria listed under SRP Section 15.0, "Transient and Accident Analysis."

The applicant based some events analyses in DCD Tier 2, Section 15.3 on an assumed OLMCPR of 1.30 and a bounding number of damaged fuel rods of 1000. Since the DCD does not include the SLMCPR, the applicant will need to address these hypotheses. The DCD treats the 1.30 as an SLMCPR because they calculate dose rates whenever MCPR $\leq$ 1.30. This is addressed in RAI 15.0-16 which the staff is currently reviewing. **RAI 15.0-16 is being tracked as an open item.**

The staff reviewed DCD Tier 2, Revision 1, Section 15.3 and found it had not provided a complete source term for the radiological consequence analysis for the infrequent events identified in the DCD. In RAI 15.3-25, the staff requested that the applicant revise DCD Tier 2, Tables 15.3-13 and 15.3-16 with applicable information pertaining to radiological consequence analysis for those infrequent events listed in the DCD. GEH's response to this RAI is under staff review. **RAI 15.3-25 is being tracked as an open item.**

In RAI 15.3-26, the staff noted that since only the limiting events will undergo analysis during the COL licensing phase, design certification requires analyses of all IEs. The applicant needs to revise DCD Tier 2, Table 15.3-1, to show the results of all IEs. The applicant needs to analyze the events described in Sections 15.3.7 to 15.3.12 and 15.3.14. **RAI 15.3-26 is being tracked as an open item.**

15.3.1 Loss of Feedwater Heating with Selected Control Rod Run-In Failure

15.3.1.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

A loss of FW can occur in at least two ways, including (1) closing of the steam extraction line to the heater, and (2) FW flow bypassing the heater. The DCD states that the ESBWR design is such that no single failure or operator error will cause loss of FW heating greater than 100 °F. Loss of FW heating will increase core water density, which will result in increased core neutron moderation, which in turn will increase power. The FWCS monitors and compares FW temperature to a reference temperature, and when the system detects increased subcooling, it alerts the operator and signals the RC&IS to activate the SCRRI to automatically reduce reactor power and avert a scram. In this case, the analysis assumes that the SCRRI fails to insert the rods.

The calculated results indicate that the addition of void reactivity is counterbalanced by fuel temperature reactivity. The transient attains a power level of 120 percent of normal power, but the pressure remains at about the normal level, with FW flow increasing by about 5 percent. The value of the MCPR falls below the designated OLMCPR of 1.30.

The DCD assumes that this event will damage 1000 fuel rods, a number that bounds the expected radiological consequences. The estimated frequency of this event is 1.488×10^{-3} pry, which classifies it as an IE, as indicated in DCD Tier 2, Section 15A.3.6.3.

15.3.1.2 Technical Evaluation

The calculated results indicate that coolant pressure will remain within normal limits; however, the thermal safety limits are exceeded at about 75 seconds into the transient when the MCPR becomes less than the assumed safety limit. The applicant estimates that 1000 fuel rods are affected. This is a bounding number for all transients in this chapter. An evaluation of the radiological consequences will be performed elsewhere. However, the evolution of this transient is hypothetical because the reactor would normally scram in such circumstances. Because the reactor achieves a steady state achieved with an MCPR less than the safety limit value, continued operation at that state will increase the number of damaged rods. The DCD

states that recalculation of this transient will take place at the COL stage.

15.3.1.3 Summary and Conclusion

The maximum pressure remains within the limits of normal operating pressure, and the transient stabilizes at about 150 seconds into the transient. Radiological consequences will be reviewed separately. Therefore, the transient resulting from loss of FW heating with SCRRI failure to insert meets the regulatory requirements of an IE and is acceptable.

SRP Section 15.0 states that the accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the radiological and RCPB pressure limits as a result of these events. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.3.2 FW Controller Failure—Maximum Demand

15.3.2.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

The DCD, referencing the design of the FWCS, states that FW controller failure requires several failures to result in FW failure. For the purpose of reactor systems review of FW failure, maximum demand is assumed when such a failure takes place. The estimated frequency is $5x10^{-4}$ pry, which classifies it as an IE (DCD Tier 2, Section 15A.3.5).

With excess FW flow, the water level rises to the high reference point (Level 8) where FW pumps initiate a runback, the main turbine trips, and the reactor scrams. DCD Tier 2, Figure 15.3-2 depicts the sequence of events as a function of time.

The calculated results indicate that the FW flow is ramped up to 170 percent of normal flow in about 2.5 seconds. The main TBVs open at 12.3 seconds to relieve vessel pressure. The reactor reaches the Level 8 water level at 15.3 seconds; turbine trip, reactor scram, and FW pump runback activate at 16.15 seconds. At 16.35 seconds, control rod insertion and IC steamflow initiate. Vessel pressure reaches its maximum, slightly over normal, operating pressure.

At about 18.0 seconds the scram is complete and the reactor is stable on IC cooling. The value of the MCPR remains higher than the designated OLMCPR of 1.30; therefore, no fuel damage or radioactive releases are anticipated.

15.3.2.2 Technical Evaluation

The DCD states that the excessive FW flow transient requires more than one failure to take place. As such, the anticipated frequency is lower than $1x10^{-2}$ and it is included in the IE

category. The calculated results indicate that the excessive FW flow transient will cause minimal disturbance to the reactor in that there is a small and short power peak and a corresponding small pressure peak and the value of the MCPR will remain well above the designated safety limit. The system inserts control rods to scram the reactor, which remains under stable IC cooling. The results of this transient meet the acceptance criteria in that the vessel does not overpressurize, the cladding remains intact because the MCPR remains above the safety limit, and the reactor is maintained in a stable state.

15.3.2.3 Conclusion

In this transient the excessive FW flow causes an insignificant perturbation to vessel pressure but does not violate the fuel OLMCPR, and at the end of the transient, the reactor is at a stable condition. Therefore, the results of the analysis are acceptable.

SRP Section 15.0 states that the accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the radiological and RCPB pressure limits as a result of these events. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.3.3 Pressure Regulator Failure: Opening of All Turbine Control and Bypass Valves

15.3.3.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

The SB&PC system controls vessel pressure and steam turbine bypass. In Section 15.2 of this report, the staff examined the accidental opening of a turbine control or bypass valve. The DCD states that the electronic logic aspects of the SB&PC system are such that it would take multiple failures to accidentally open all of the turbine control or bypass valves. Therefore, the applicant considers this event as having a very small probability of occurrence and categorizes it as an IE. DCD Tier 2, Section 15A.3.1 estimates that the frequency of this event is $5x10^{-4}$ pry, which classifies it as an IE.

DCD Tier 2, Table 15.3-4 and DCD Tier 2, Figure 15.3-3 illustrate the calculated results of the time-dependent evolution of the transient. At 17.89 seconds into the transient, turbine low pressure will initiate MSIV closure, which in turn will initiate reactor scram and IC operation.

Because of increased steamflow, water level decreases, reaching the RPV L2 level at 28.14 seconds. At 34.17 seconds, the IC begins to return condensate coolant to the vessel, and at 38.14 seconds, the high-pressure CRD injection starts and vessel water level recovery initiates.

As stated in the DCD, the ESBWR has a 105-percent bypass capacity. Opening all of the bypass valves produces rapid depressurization resulting in an increase of the void fraction that

reduces power. In the first few seconds, the FW system attempts to stabilize operation by increasing FW flow. The reactor scrams at 20 seconds, zeroing power and reducing FW flow to 20 percent of normal flow. Simultaneously, the IC steamflow increases to about 20 percent of normal steamflow because the MSIVs are closed. At this time, reactor operation stabilizes with IC cooling and the achievement of normal water level. The MCPR value stays well above the safety limit value and increases during the transient.

15.3.3.2 Technical Evaluation

This transient has a very low probability of occurrence and is classified as an IE. The important feature is that vessel depressurization leads to decreasing power, reactor scram, and initiation of IC cooling. Assuming that the required instrumentation and systems will operate as designed and as expected, the results of this transient meet the acceptance criteria (i.e., maximum pressure remains below 110 percent of design pressure, no cladding damage occurs, and the transient evolves into a stable situation).

Revision 3 of DCD Tier 2, Section 15.3.3.1, "Pressure Regulator Failure-Opening of All Turbine Control and Bypass Valves," states that "the event is considered as a limiting fault." As stated in RAI 15.3-29, the staff does not agree with this characterization of the event. This event is an IE as referred to in other parts of the DCD, and in RAI 15.3-29, the staff requested that the applicant revise this section of the DCD to characterize it as an IE rather than as a limiting fault. GEH has not responded to this request. **RAI 15.3-29 is being tracked as an open item.**

15.3.3.3 Conclusion

Inadvertent opening of the TCV and/or bypass valves from power results in fast depressurization, decrease in power, reactor scram, and IC initiation. The calculated results indicate that the vessel pressure remains below operating values, the MCPR is well above the designated OLMCPR of 1.30, and the reactor, being cooled by the ICs, assumes a stable state. The results of this transient satisfy the acceptance criteria; therefore, this transient is acceptable (subject to resolution of the open item).

SRP Section 15.0 states that the accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the radiological and RCPB pressure limits as a result of these events. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis.

15.3.4 Pressure Regulator Failure: Closure of All Turbine Control and Bypass Valves

15.3.4.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

This transient assumes failure of the SB&PC system with closure of all of the TCVs and all bypass valves. The DCD states that for this transient to occur, more than a single failure is necessary and the probability is exceedingly low; therefore, this transient qualifies as an IE. As the TCVs and the bypass valves begin to close, vessel pressure increases, void collapse increases moderation, power increases, and the ESBWR reaches the neutron high-flux control rod insertion setpoint at 1.78 seconds. Control rod insertion initiates at 2.03 seconds. TCV closure is completed at 2.5 seconds into the transient. CRD high-pressure injection activates on RPV L2 to recover vessel water level. In DCD Tier 2, the calculation is carried to 50 seconds. At the end of this time, the reactor has recovered water level and dome pressure is about 50 percent of normal operating pressure. Vessel pressure peaks at 6.0 seconds at 114 percent of normal operating pressure, but the MCPR remains above the OLMCPR of 1.30 throughout the transient. IC initiation does not take place because neither the high dome pressure nor the low water level signals are in effect for the 10 or more seconds and 6 or more seconds, respectively, required for IC initiation. DCD Tier 2, Section 15A.3.2 estimates the event frequency as $5x10^{-4}$ pry which classifies it as an IE.

15.3.4.2 Technical Evaluation

Amendment 26 to General Electric Standard Application for Reactor Fuel (GESTAR) II, dated March 29, 2000, approved the change of this transient from moderate frequency to IE for BWR/6 plants. It stated that "the classification of the pressure regulator downscale failure (PRDF) as an AOO was also reevaluated and it was concluded that the expected frequency of the single initiating failure was below the moderate frequency event definition, and was an infrequent event."

The applicant based the categorization of this transient as an IE on the performance of the SB&PC system. The peak pressure reaches 114 percent of operating pressure (i.e., remains below the 110 percent of design pressure). The MCPR value remains above the OLMCPR of 1.30; thus, no fuel damage is expected during this transient. The reactor recovers water level, and the operator has a number of choices for long-term cooling.

DCD Tier 2, Figure 15.3-4a indicates a sharp rise in total power (although the corresponding simulated power peak is not as pronounced), similar to that shown in the control rod drop transient. In RAI 15.3-11, Supplement 1, the staff requested the applicant to calculate the total power deposition and the corresponding cladding strain. GEH has not responded to this request. **RAI 15.3-11 is being tracked as an open item.**

15.3.4.3 Conclusion

Due to the open item discussed above, the staff cannot finalize its conclusions regarding this Infrequent Event.

15.3.5 Generator Load Rejection with Total Turbine Bypass Failure

15.3.5.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

Significant reduction in generator load initiates a signal for fast closure of the TCVs to avoid turbine overspeed. At the same time, the SB&PC system signals the TBVs to open (in the fast mode). This transient analysis examines total failure of the turbine bypass system. The DCD states that the frequency of this event is estimated to be 1.98×10^{-4} pry, which classifies it as an IE.

The calculated results indicate a sharp and narrow peak in total power, a dip in FW flow (5 seconds) resulting from increased pressure, and a wide pressure peak that decays slowly after 5 seconds. The simulated thermal power exceeds the neutron high-point scram signal, initiating a reactor scram at about 6 seconds to be completed 27 seconds into the event. Peak dome pressure remains lower than the SRV activation pressure. The calculation ends at about 50 seconds, at which time the FW flow is at about 50 percent. At about 0.6 seconds, the MCPR falls below the safety limit of 1.30 to reach a minimum value of about 1.23 at 1.20 seconds or about the end of the neutronic power peak.

15.3.5.2 Technical Review

On sensing loss (or partial loss) of electrical load, the system commands the TCVs to close in the very rapid mode, causing a sudden reduction in steamflow, void collapse, and pressure and power spikes. The calculated results (DCD Tier 2, Figure 15.3-5a) show a very narrow high-power peak.

As in the transient described in Section 15.3.4 of this report, there is a very fast energy deposition for this event. The staff requested, in RAI 15.3-11, that the applicant calculate the energy deposition along with the pellet-clad mechanical interaction for cladding strain, or else it should explain why this is not necessary. **RAI 15.3-11 is being tracked as an open item.**

The FW dips to about 60 percent of normal at about 3 seconds, and the simulated thermal power peaks above the high neutron flux setpoint—a fraction of a second after TCV closure—initiating a scram. Dome pressure peaks also at the minimum of the FW flow but remains below the SRV setpoint. The increased pressure, decrease in FW flow, and void collapse reduce the WR water level below the RPV L2 level for about 20 seconds. This is about the minimum time required to initiate IC operation. In this case, the analysis shows that the ICs do not start, but the CRD high-pressure injection initiates. Finally, the MCPR falls below the OLMCPR of 1.3 between 0.7 to 1.7 seconds into the event.

The DCD states that the number of fuel rods affected will remain below 1000 and the offsite dose will remain less than 2.5 rem TEDE. The core recovers water level and remains stable and shutdown.

15.3.5.3 Conclusion

Due to the open item discussed above the staff cannot finalize its conclusions regarding this Infrequent Event.

15.3.6 Turbine Trip with Total Turbine Bypass Failure

15.3.6.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

A variety of causes, such as loss of control fluid pressure, large vibrations, low condenser vacuum, and vessel high water level, can result in a turbine trip. Turbine trip is followed by fast opening of the bypass valves. Failure of all bypass valves to open would require multiple failures. DCD Tier 2, Section 15A.3.3 estimates the frequency of this event to be 5.7×10^{-4} pry, which classifies it as an IE.

The sequence of events is nearly identical to that described in Section 15.3.5 of this report for total loss of electrical load. DCD Tier 2, Figure 15.3-6 shows the calculated sequence of events as a function of time. The results indicate a sharp and narrow peak in total power, a dip in FW flow (5 seconds) resulting from increased pressure, and a wide pressure peak that decays slowly after 5 seconds. The calculation ends at 10 seconds, at which time the FW flow is still increasing and is about 120 percent of normal flow. The high-pressure CRD injection probably initiates at this time. At about 0.6 seconds, the MCPR falls below the OLMCPR of 1.30 to reach a minimum value of about 1.23 at 1.2 seconds or at the end of the total power peak. Unlike the generator load rejection event, the simulated thermal power registers only a small rise above normal at about 1 second into the event followed by decay to less than 50 percent at 10 seconds. The system initiates shutdown at 0.45 seconds. In the long term, CRD injection initiates to recover RPV level. At 10 seconds (i.e., the end of the event analysis), the WR and the two-phase level readings are still decreasing.

Finally, the dome pressure peaks at about 4 seconds into the event but remains well below the SRV setpoint and well below the 110 percent of the design value.

15.3.6.2 Technical Evaluation

This event is almost identical to that of the load rejection with turbine bypass failure discussed in Section 15.3.5 of this report. The results of the analyses are the same and the conclusions are the same.

The analysis presented in previous DCD revisions covered just the first 10 seconds in this event, which seemed to the staff to be too brief a time to establish that the reactor is stable. In RAI 15.3-12, the staff requested the applicant to run this event analysis for a longer period to show that the reactor has stabilized and long-term cooling has been established. For example, DCD Tier 2, Figure 15.3-5a shows a dip in FW flow at 38 seconds with no explanation of its origin. The applicant was requested to address whether the same is true for this event and, if so, what is causing the dip.

In response, the applicant addressed the length of time the transient was run (i.e., 20 seconds) and concluded that it was stabilized even though the FW flow was increasing and the dome

pressure was decreasing. (See DCD Tier 2, Figures 15.3-5a and 15.3-5d.) However, the applicant ran the same transient for 50 seconds in DCD Tier 2, Revision 3 (same figure numbers), showing that the FW flow had indeed decreased and the pressure was lower and stable. Therefore, RAI 15.3-12 is resolved.

As in the previous two events, there is a pulse-like power event, and therefore, the staff requested, in RAI 15.3-11, that the applicant calculate the energy deposition along with the associated pellet-cladding mechanical interaction. If the applicant determines that energy deposition and pellet-cladding interaction should not be calculated, it should state the reason for this decision. **RAI 15.3-11 is being tracked as an open item.**

15.3.6.3 Conclusion

Due to the open item discussed above, the staff cannot finalize its conclusions regarding this Infrequent Event.

15.3.7 Control Rod Withdrawal Error During Refueling

15.3.7.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

The DCD states that there is no postulated set of circumstances that result in an inadvertent control rod withdrawal error while in the refueling mode. The applicant based this conclusion on system interlocks that ensure against inadvertent criticality. In addition, removal of the highest worth control rod or two rods in the same hydraulic control unit will not make the reactor critical. This is an ESBWR design feature.

To minimize the possibility of inserting fuel into any cell without control rods inserted, the design requires that all control rods be fully inserted before fuel is loaded into the core. The design achieves this protection through the use of interlocks. When the mode switch is in the “refuel” position, the interlock prevents the reload platform from moving over the core if a control rod is withdrawn and fuel is in the hoist, and it also prevents rod withdrawal. Control rod withdrawal in the refueling mode can occur if the refueling platform is not over the core and the hoist is not loaded with fuel. The feasible selections (in the RC&IS) are “single” and “gang.” In this case, the interlock prevents a second rod from moving in the “single” setting or a pair of rods in the “gang” mode. Also, the physical design of the fuel that needs to be removed (four assemblies) before the control rod is removed prevents upward control rod removal from a cell.

The estimated frequency of this event is less than 1.0×10^{-3} pry, which classifies it as an IE, as indicated in DCD Tier 2, Section 15A.3.11.3

15.3.7.2 Technical Evaluation

Since the design precludes inadvertent criticality and multiple failures are necessary to cause a criticality under refueling conditions, the analysis is acceptable. However, the applicant did not explain the statement that multiple failures are needed to cause criticality. For example, the ESBWR control rod system consists of mechanical, electrical, and pneumatic/hydraulic

components and is therefore subject to mechanical, electrical, pneumatic/hydraulic, and operator error failures. DCD Tier 2, Section 15A.3.11 does not indicate how operator error or a combination of equipment failure and operator error could result in control rod withdrawal causing criticality during refueling.

The description of the interlocks effective during refueling in current BWR-4 plants is similar to the description of those in the ESBWR. A recent announcement disclosed that in 1999, a BWR-4 plant in Japan experienced inadvertent withdrawal of three control rods during refueling, which caused criticality. According to the description, this event resulted from a combination of mechanical failures and operator error.

In DCD Tier 2, , Section 15.3.7, the applicant states that there is no postulated set of circumstances that results in an inadvertent RWE during refueling because of interlocks and design improvements. In RAI 15.3-19, the staff asked the applicant to provide the basis, using applicable information, for reaching this conclusion and the analysis demonstrating the magnitude of the consequences for this event under refueling conditions. GEH has not responded to this RAI. **RAI 15.3-19 is being tracked as an open item.**

15.3.7.3 Conclusion

Due to the open item discussed above the staff cannot finalize its conclusions regarding this event.

15.3.8 Control Rod Withdrawal Error During Startup

15.3.8.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

In this analysis, the applicant assumed that during startup a control rod assembly or a single control rod is either inadvertently withdrawn or the automated control rod system malfunctions. The RC&IS prevents the withdrawal of any out-of-sequence rod. Also, if a rod assembly withdrawal sequence is violated, the RC&IS will initiate a rod block. In addition, the startup range neutron monitor (SRNM) has a period-based reactor trip function that will either initiate a rod block if the event period is between 10 and 20 seconds or initiate a reactor scram if the period is shorter than the 10 seconds.

For this event, the assumption is that a single rod or a rod bundle is continuously withdrawn from close to zero power. At about 14 seconds, the SRNM initiates a rod block and, at about 25 seconds, initiates a period scram. About 2 seconds later, the event is terminated. Assuming adiabatic heat deposition, the analysis calculated the total energy generated and deposited in the fuel. The rod reactivity worth, the rod speed (and reactivity insertion rate), and initial fuel and coolant temperature come from the advanced boiling-water reactor (ABWR) core and parameters. Under these conditions the calculated peak fuel enthalpy is about 70 joules per gram (J/g). If the SRNM did not function (or the event initiated at a higher power), the average power range monitor (APRM) startup mode would provide a scram at 15 percent power. In this case the calculated peak fuel enthalpy would be 146.5 J/g, which is less than the limit of 711 J/g. This event does not challenge either of the barriers.

The estimated frequency of this event is 1.5×10^{-7} pry, which classifies it as an IE, as indicated in DCD Tier 2, Section 15A.3.12.3.

15.3.8.2 Technical Evaluation

The DCD states that multiple failures (or an inadvertent operator action) are necessary to cause an uncontrolled rod (or rod assembly) withdrawal. The analysis included calculations to assess the impact from about zero power. The energy deposition model assumed adiabatic heating, which is a conservative assumption. The analysis derived the scram reactivity shape from core design, which is assumed to be realistic. Other assumptions included six delayed neutron groups and a 3.5 meter (11.5 foot) ABWR core and rod withdrawal speed of 30 millimeters per second. The overly conservative results and the similarity of the ABWR and the ESBWR cores justify the use of the ABWR core and parameters.

After event initiation, the reactor promptly scrams on the period trip function. The average at-peak axial location enthalpy increase is 0.63 J/g for a final value of 69.5 J/g. This is very small compared to the limit value of 711 J/g (SRP Section 4.2). The corresponding pressure and MCPR values remain negligibly small. Even if the period trip failed, the reactor would scram from the APRM at 15 percent power. At that level, the enthalpy values are still small. Therefore, the acceptance criteria are satisfied.

15.3.8.3 Conclusion

In the control rod (or rod assembly) withdrawal error at startup, the reactor should scram on the period meter at a very early stage and generate a small amount of energy deposition. The acceptance criteria are satisfied.

SRP Section 15.0 states that analyses use staff-approved acceptable analytical methods. The applicant used PANAC11 for the accident analyses. PANAC11, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the radiological and RCPB pressure limits as a result of these events. This conclusion is dependent on the closure of the open items on PANAC11 in Chapter 21 such that PANAC11 does not change in a way that materially affects this analysis.

15.3.9 Control Rod Withdrawal Error During Power Operation

15.3.9.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

The ESBWR is equipped with an ATLM, which is a subsystem of the RC&IS. The ATLM has two channels and monitors the MCPR and maximum linear heat generation rate. Should the reactor reach either of these limits, the system will remove the rod withdrawal permissive. Potential causes of rod (or rod assembly) withdrawal include procedural operator error and malfunction of the automated rod withdrawal sequence control logic. DCD Tier 2, Section 15A.3.13 estimates the probability of an RWE during power operation to be

$2.44x10^{-5}$ pry. The calculation distinguishes between an error during automatic rod movement and manual rod movement. In the first case, the calculation takes credit for the ATLM, which is a subsystem of the RC&IS. The ATLM is based on actual core thermal limit information and, in the case of control rod withdrawal, would remove the rod motion permissive when the core reaches thermal limits. The estimated annual frequency for the automatic rod movement is $1.2x10^{-9}$ pry. For the manual rod movement, the applicant estimates the annual frequency as the product of the operator error times the probability of failure of the ATLM. The estimated value is $2.44x10^{-5}$ per year.

The DCD claims that in either case the ATLM system will hold the progression of the transient before any limits are violated. Therefore, there is no basis for the RWE to occur and, according to the DCD, no need to analyze this event.

15.3.9.2 Technical Evaluation

The DCD categorizes this event as a near impossibility. The SRP guidance is that the analysis of control rod withdrawal during power operation be performed, and that the event be classified as an AOO. The ESBWR is equipped with the ATLM system, which would be able to arrest rod withdrawal should it happen based on core data such as inlet temperature, core power, and coolant temperatures. The ATLM is a dual-channel subsystem, not subject to single failure. However, there is no reference regarding its classification as a safety grade system. In addition, there is no reference to TSs on this system. The guidance in SRP Section 15.0 and 10 CFR 50.36, "Technical Specifications," requires that structures, systems, and components related to protection of SAFDLs should be safety grade.

The staff requested that the applicant provide additional information regarding this event in RAIs 15.0-15 and 15.3-33.

In RAI 15.0-15, the staff requested the applicant to describe the basis for the reclassification of the RWE, including initiating actions/events and mitigating strategies from all modes of operation; to address the potential for a "gang" withdrawal error (e.g., multiple control rods); and to identify the proposed acceptance criteria for the new event classification.

In RAI 15.3-33, the staff requested the applicant to analyze the RWE at power event and provide an evaluation of the barrier performance.

RAIs 15.0-15 and 15.3-33 are being tracked as open items.

15.3.9.3 Conclusion

Due to the open items discussed above, the staff cannot finalize its conclusions regarding this event.

15.3.10 Fuel Assembly Loading Error, Mislocated Bundle

15.3.10.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

Mislocated assemblies involve at least two fuel assemblies at interchanged positions. If one is assumed to operate at a lower power, the other will operate at a higher power level. The plant is instrumented so that the core monitor will recognize the mislocated assemblies, allowing the operator to intervene and minimize the consequences of the fuel mislocation. That would be the case if the higher power assembly is next to an automatic fixed in-core probe or a local power range monitor. However, should another fuel assembly be located between the instrument and the higher power assembly, then the core monitor will not recognize the mislocation. In this case, the possibility exists that the assembly will operate above the thermal limits. Should a mislocated assembly suffer thermal-mechanical damage resulting in leaking fuel rods, the application of existing leak detection methods can suppress the local power and the radioactive leakage.

The maximum power at which the mislocated assembly will operate is limited by the detection capability of the core monitoring system. Analyzing each case for a core as large as the ESBWR may be wasteful. The applicant pursued the analysis of a very conservative case to be clearly bounding for any mislocated assembly. First, it assumed that all fuel rods in the affected assembly will be damaged and become leakers. Then it is assumed that all four assemblies surrounding the affected assembly experience damage to all their rods. In addition, it added a factor of 1.4 to account for fission product inventory differences over an operational cycle, and it added a factor of 2.5 to account for variation of cycle-dependent bundle power as a ratio of the end-of-cycle average bundle power. This amounts to a factor of 3.5 to bound the end-of-cycle fission product inventory.

The radiological consequences should be a small part (defined as 10 percent) of the 10 CFR 50.34 exposure limits (i.e., 2.5 rem TEDE whole body and 30 rem to the thyroid for an individual located at any point in the exclusion boundary for 2 hours from the beginning of emission). Both limits meet the guidance in SRP Section 15.4.7.

The calculation of the dose to the site boundary depends on whether or not the plant is equipped with a main steamline high radiation isolation trip. The topical report “GESTAR II Amendment 28, Revision 1, Mis-loaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan 15.4.7” (which has been approved by the staff), meets the above criteria.

The boundary dose for plants without high radiation isolation trip depends on their long-term meteorological parameters. Individual plants must periodically verify their meteorological conditions to ascertain that they are within the limits.

The estimated frequency of this event is 9.56×10^{-4} pry, which classifies it as an IE, as indicated in DCD Tier 2, Section 15A.3.14.3.

15.3.10.2 Technical Evaluation

The SRP Section 15.4.7, Revision 3 (issued March 2007),, acceptance criteria are set forth in 10 CFR Part 100, “Reactor Site Criteria.” Amendment 28 to GESTAR II allowed the category change from AOO to Infrequent, Acceptance criteria 10 percent of 10 CFR Part 20 (SER dated January 26, 2006).

As stated above, the primary safeguards against fuel loading errors are design features and

loading procedures to minimize the probability of a misloading event. The applicant has implemented these safeguards in the fuel and plant design. In addition, GDC 13 requires the provision of instrumentation to monitor local operating power versus anticipated power levels. Both of the above have been implemented.

In addition, the applicant noted that a mislocated fuel bundle in the immediate vicinity of an automatic fixed in-core probe or a local power range monitor will be readily detected (after startup) and power will be suppressed within normal operating range. Once removed, the mislocated assembly from a detector may not be detected. But, the power mismatch is limited. On such occasions, it is possible that the mislocated bundle will operate outside its thermal mechanical limits and damage the cladding. Fuel leakage (from any cause) is detected by monitoring and suppressing power to the segment with the leakers to diminish or eliminate the leak.

To estimate the potential site boundary dose (rate), the applicant performed a very conservative calculation to bound the potential emissions and the site boundary exposure. In this manner, plants with main steamline high activity trip satisfy the requirements of 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," as described (and approved) in Attachment A to GESTAR II, Amendment 28, Revision 1.

For plants without the steamline high reactivity trip feature, the calculation of the site boundary exposure is based on the dispersion factor (χ/Q). The dispersion factor is site dependent. Attachment B to GESTAR II, Amendment 28, Revision 1, back calculates the dispersion factor to meet the limits of 10 CFR 100.11. The value is 1.67×10^{-3} seconds per cubic meter (s/m^3). Therefore, the objective is to maintain a dispersion factor equal to or less than 1.67×10^{-3} s/m^3.

15.3.10.3 Conclusion

In this section, the applicant analyzed the fuel misloading. The analysis assumed that one of two interchanged assemblies is operating in a location of higher power and is one location removed from a detection device and, therefore, is subject to potential thermal mechanical damage. The DCD makes a bounding calculation and estimates the site boundary exposure for plants with main steamline high reactivity trip and for plants without this feature. In the first case, the DCD demonstrates that the exposure criteria are satisfied. In the second case (which depends on the site dispersion factor), the DCD back calculated the minimum dispersion factor necessary to meet the 10 CFR 50.34 criteria.

In addition, the review established that the design has the required instrumentation and controls to monitor and control local power as required by GDC 13. The staff finds the fuel misloading analysis to be acceptable.

15.3.11 Fuel Assembly Loading Error, Misoriented Assembly

15.3.11.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

The probability that a misoriented assembly will be placed in a core position and will not be

detected is very small. The five different visual indications for the proper fuel orientation: (1) the fastener springs and spacers to maintain channel clearance are located in the corner toward the center of the control rod; (2) identification on the assembly handle points toward the adjacent control rod; (3) the channel spacing buttons are adjacent to the control rod passage area; (4) the assembly identification numbers located on the fuel assembly handles are readable from the center of the assembly; and (5) there is cell-to-cell replication. Based on the above, the staff considers that the probability of a misoriented assembly's not being detected is very small.

DCD Tier 2, Reference 15.3-1 suggests that the analysis is the same as that for the mislocated assembly, described in Section 15.3.10.2 of this report.

15.3.11.2 Technical Evaluation

The staff's review of DCD Tier 2, Section 15.3.10.2 indicates that the analysis is a conservative bounding calculation for assembly location and fuel burnup; therefore, it is acceptable. For the same reasons, the analysis of the misoriented assembly is acceptable.

The estimated frequency of this event is 2.0×10^{-3} pry, which classifies it as an IE, as indicated in DCD Tier 2, Section 15A.3.15.3.

15.3.11.3 Conclusion

In this section, the applicant analyzed the fuel assembly misorientation. The DCD makes a bounding calculation and estimates the site boundary exposure for plants with main steamline high reactivity trip and for plants without this feature.

In the first case, the applicant demonstrated that the exposure criteria are satisfied. In the second case, which depends on the site dispersion factor, it back calculated the minimum dispersion factor necessary to meet the criteria.

In addition, the review established that the design has the required instrumentation and controls to monitor and control local power as required by GDC 13. The staff finds that the fuel misloading analysis is acceptable.

15.3.12 Inadvertent Shutdown Cooling Function Operation

15.3.12.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

This transient is concerned with power increase resulting from misoperation of the RWCU system or the SDC system either during power operation or during startup. Malfunction of the SDC leads to lower temperature cooling water's entering the core, resulting in mild reactivity insertion and power increase. The DCD does not quantify the resulting temperature differences or the reactivity insertion. However, the DCD states that, if there is no operator action, the system will assume a new power level but with or without operator action the system will not violate the thermal limits. During startup, RWCU/SDC malfunction will increase the reactivity

insertion rate and may result in a scram. Either way the system will not violate the thermal limits.

15.3.12.2 Technical Evaluation

The DCD states that the transient does require analysis. In RAI 15.3-34, the staff requested the applicant to quantify the range of expected temperature limits and the resulting reactivity and reactivity rate to justify this statement. GEH has not responded to this RAI. **RAI 15.3-34 is being tracked as an open item.**

15.3.12.3 Conclusion

Due to the open item discussed above, the staff cannot finalize its conclusions regarding Inadvertent Shutdown Cooling Function Operation.

15.3.13 Inadvertent Opening of a Safety/Relief Valve

15.3.13.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

An SRV opening could result from a valve malfunction or an operator error. SRV discharge is directed to the suppression pool, which could overheat (if the operator does not close the SRV), triggering reactor scram. In this case, the analysis assumes no operator action. The calculated results show that in about 30 seconds the reactor will assume a new power level and in 412.5 seconds the reactor will scram on high suppression pool temperature (DCD Tier 2, Figure 15.3-8). The vessel pressure settles to slightly lower operating pressure and the MCPR remains well above the operating limit of 1.30. DCD Tier 2, Section 15A.3.8 estimates the frequency of this event as 2.81×10^{-3} pry, which classifies it as an IE.

RAI 15.3-16 referred to a factor attributed to the triplicate electronic control system. In DCD Tier 2, Sections 15.3.1, 15.3.3, and elsewhere, the applicant based valve failure probability on the electronic portion of the control and ignored the mechanical aspects of valve failure. GEH letter MFN 07-264 documents the applicant's responses regarding the mechanical aspects of TBVs and SRVs in accident analyses. The applicant's responses answered the questions raised in RAI 15.3-16; therefore, this RAI is closed.

15.3.13.2 Technical Evaluation

The analytical results indicate that this is an inconsequential transient with or without operator intervention to close the discharging SRV. Neither fuel damage nor overpressurization take place, and the MCPR remains well above the operating limit of 1.30 (DCD Tier 2, Figure 15.3.8). These results satisfy the acceptance criteria for fuel damage and overpressurization; therefore, the transient analysis is acceptable, subject to the RAIs that follow. However, primary water will enter the suppression pool, introducing radioactivity. After plant stabilization the operator has several choices for disposing of the activity. Exposure at the site boundary, should radioactivity be released in compliance with relevant TSs, would likely be nearly negligible.

The calculation of the event frequency (DCD Tier 2, Section 15A.3.8) assumes 0.0 pry for incorrect setpoint or spring adjustment, spring relaxation, and operator error. (Operator error setting or adjusting the valve spring?).

15.3.13.3 Conclusion

Assuming that the reactor will shut down on high suppression pool temperature, this event meets the pressure and fuel damage acceptance criteria. Therefore, the staff considers it acceptable.

SRP Section 15.0 states that accident analyses use staff-approved acceptable analytical methods. The applicant used TRACG for the accident analyses. TRACG, however, is the subject of several open items set forth in Chapter 21 of this report. Therefore, the analysis is acceptable pending closure of these open items. The NRC staff further concludes that the applicant has demonstrated that the ESBWR will not exceed the radiological and RCPB pressure limits as a result of these events. This conclusion is dependent on the closure of the open items on TRACG in Chapter 21 such that TRACG does not change in a way that materially affects this analysis. This conclusion cannot be finalized until the other remaining open items are resolved.

15.3.14 Inadvertent Opening of a Depressurization Valve

15.3.14.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

Opening of a depressurization valve could result from a valve malfunction or operator error. The difference between the depressurization valve and the SRV, examined in Section 15.3.13 of this report, is that the depressurization valve is bigger and discharges into the drywell, where it could raise the temperature (within a few seconds) to the reactor scram setpoint. Opening of a depressurization valve amounts to a depressurization event in that the SB&PC system will close the TCVs to stabilize the reactor vessel pressure to a slightly lower pressure and the reactor will resume operation at a slightly lower than normal power. However, the depressurization valves discharge into the drywell, and the reactor will scram on high drywell temperature. DCD Tier 2, Section 15A.3.9 estimates the frequency of this event as 5.75×10^{-4} pry, which classifies it as an IE.

15.3.14.2 Technical Evaluation

This is an inconsequential event in the sense that the plant does not get close to fuel damage or overpressurization. The acceptance criteria are satisfied and the event analysis is acceptable.

15.3.14.3 Conclusion

This event meets the pressure and fuel damage acceptance criteria. Therefore, it is acceptable.

15.3.15 Stuck-Open Safety Relief Valve

15.3.15.1 Summary of Technical Information

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

A stuck-open safety relief valve is attributed to valve malfunction (either electronic or mechanical) whether the opening resulted from inadvertent operator action or a high-pressure signal. More specifically, in this event the applicant assumed that the SRV remains open after issuance of a scram signal and the reactor is shutting down. The calculated sequence of events (after scram) indicates that depressurization begins at 10 seconds; vessel water level reaches RPV L2 at 19.3 seconds, activating high-pressure CRD injection; and low steamline pressure activates MSIV closure, which is completed at 124.8 seconds. At 154 seconds, the ICs are in full operation. As expected, vessel pressure keeps falling at a steady rate, and the MCPR value continues to increase above the normal operating value. The estimated frequency of a stuck-open relief valve is 2.24×10^{-4} pry, classifying this transient as an IE. (See DCD Tier 2, Section 15A.3.10.1.)

15.3.15.2 Technical Evaluation

This is a post-shutdown event that depends on successful removal of decay heat. With successful operation of the high-pressure CRD and the ICs, the system ensures decay heat removal and water level recovery. In this case neither the pressure nor the MCPR value comes close to critical values; therefore, the acceptance criteria are satisfied and the results of the analysis are acceptable.

In RAI 15.3-23, supplement 1, the staff requested that the applicant modify the DCD to provide justification of the following comments:

1. DCD Tier 2, Section 3.9.1.4, regarding ASME Class 2 and 3 Valves, states: "Elastic analysis methods and standard design rules used for evaluating faulted loading conditions...are obtained from NC/ND-3400 of the Code. These allowables are above elastic limits."

   Enclosure MFN 07-011, "The acceptance criteria...require...that after testing...the valve shall not exhibit any deformation that would degrade its performance beyond the specification prescribed limits." The main cause of valve malfunction after opening is failure to reseat properly due to deformation. If the allowable criteria are beyond the elastic limit how do you expect the valve to reseat properly? Please clarify these statements.

2. DCD Tier 2, Section 15A.3.8.2, discussing operator error states: "He should not be opening the SRVs inadvertently and he cannot do it accidentally because a deliberate action is required to open the SRVs." This statement does not make sense, please revise the DCD to clarify this statement and to provide justification to support the conclusion that the probability of an inadvertent opening of a relief valve resulting from operator action is judged to be negligible.

GEH has not responded to this request. **RAI 15.3-23 is being tracked as an open item.**

15.3.15.3 Conclusion

Due to the open item discussed above, the staff cannot finalize its conclusions regarding this event.

15.3.16 Liquid-Containing Tank Failure

15.3.16.1 Regulatory Criteria

The requirements summarized in Section 15.1.1.2 are used in evaluating this IE.

The staff reviewed DCD Tier 2, Revision 3, Section 15.3.16, "Liquid Containing Tank Failure," in accordance with the guidance and acceptance criteria described in SRP Section 11.2 and Branch Technical Position (BTP) 11-6, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures," issued March 2007. The requirements for this analysis were initially located in SRP Section 15.7.3 with the same title. The requirements have not changed as the approach, content, and format of BTP 11-6 are consistent with that of SRP Section 15.7.3. The following acceptance criteria are applicable:

- 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," and 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public," as they relate to limits for liquid effluent concentrations in unrestricted areas; these criteria apply to releases resulting from the liquid waste management system (LWMS) during normal plant operations and AOOs,

- GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the design of LWMS components and structures housing the LWMS to control releases of liquid radioactive effluents.

The relevant requirements of the regulations identified above are met by using the regulatory positions and guidance contained in the following:

- SRP Chapter 11.2, "Liquid Waste Management System,"

- SRP Chapter 11.2, BTP 11-6,

- SRP Chapter 15.7.3, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures," issued July 1981 (with the updated SRP (March 2007), the requirements of SRP Chapter 15.7.3 have been relocated to BTP 11-6),

- Regulatory Guide (RG) 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," as it relates to the design of the LWMS and structures housing this system, as well as the provisions used to control leakages.

15.3.16.2 Summary of Technical Information

DCD Tier 2, Revision 3, Section 11.2, describes the design of the LWMS and its functions in

controlling, collecting, processing, storing, and disposing of liquid radioactive waste generated as a result of normal operation, including AOOs. DCD Tier 2, Revision 3, Figure 11.2-1, provides a LWMS process diagram depicting major subsystems. DCD Tier 2, Revision 3, Figure 11.2-2, provides an LWMS process stream information directory and simplified flow diagram.

The LWMS and its components are housed in the radwaste building and located in radiologically controlled access areas. DCD Tier 2, Revision 3, Figure 11.2-1, shows the tanks, processing equipment, pumps, valves, ion exchangers, filters, and other components. All LWMS tank overflows are routed to building sumps and drains, which are pumped to their respective drain tanks. The LWMS treatment system components are arranged in shielded enclosures and compartments to minimize exposure of plant personnel during operation, inspection, and maintenance. The COL holder will subject the LWMS to preoperational tests, and there are provisions for periodic inspections of major components to ensure the integrity of the LWMS subsystems and components.

Each subsystem of the LWMS incorporates one or more tanks to hold liquid wastes. The equipment drain subsystem includes three collection tanks, each with a capacity of about 140,000 liters (37,000 gallons), and two sample tanks, each with the same capacity. The floor drain subsystem consists of two collection tanks, each with a capacity of about 130,000 liters (34,000 gallons), and two sample tanks, each with the same capacity. The chemical drain subsystem consists of one collection tank with a capacity of about 4,000 liters (1,060 gallons). The detergent drain subsystem includes two collection tanks, each with a capacity of about 15,000 liters (4,000 gallons), and two sample tanks with the same capacities.

The LWMS comprises several subsystems such that the system can segregate the liquid wastes from various sources and process them separately. The system maintains the segregation to support the most appropriate treatment of the waste by the LWMS. Cross-connections between subsystems provide additional flexibility in processing wastes by alternate methods and provide redundancy if one subsystem becomes inoperative. The LWMS normally operates on a batch basis. There are provisions for sampling at important process points. The detection and alarm of abnormal conditions and administrative controls provide protection against accidental discharge.

15.3.16.3 Staff Evaluation

The staff conducted its evaluation of a potential release of radioactive liquid waste following the postulated failure of a tank and its components, located outside of containment, as part of its review of DCD Tier 2, Revision 3, Section 15.3.16, with information drawn from DCD Tier 2, Revision 3, Sections 11.2 and 12.2. Section 12.2 of DCD Tier 2, Revision 3, presents information on the expected inventory of radioactive materials in LWMS tanks. The staff reviewed the LWMS in accordance with the guidance of SRP Section 11.2 and BTP 11-6 (March 2007), or equivalently with SRP Section 15.7.3 (July 1981). Staff acceptance of the postulated impact of a failure of a LWMS tank containing radioactive materials is based on the design's meeting the requirements of GDC 60; the effluent concentration limits of Table 2 (Column 2) of Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20, "Standards for Protection Against Radiation"; and RG 1.143, as it relates to the design of structures housing LWMS components and provisions

used to control leakage and minimize spills into the environment.

In reviewing DCD Tier 2, Revision 1, the staff could not confirm that the approach used in assessing the impact of tank failure was consistent with guidance of the SRP and SRP acceptance criteria. The radiological source term postulated to be released in an unrestricted area is the radioactivity contained in one of several tanks that are part of the LWMS. The evaluation considered the impact of the release of radioactive materials on the nearest potable water supply located in an unrestricted area and whether the impact results in the presence of radioactivity in potable water above the concentration limits of Appendix B (Table 2, Column 2) to 10 CFR Part 20. The staff asked the applicant to provide additional information. The following paragraphs discuss the staff's evaluation of the applicant's responses to the RAI.

A review of DCD Tier 2, Revision 3, Section 15.3.16, indicates that the technical approach is not consistent with that described in SRP Section 11.2 (BTP 11-6) or SRP Sections 15.7.3.II and 15.7.3.III. The analysis assumes that such tanks are located in compartments with sealed concrete walls designed to hold the expected amounts of liquid wastes in the event of a tank failure. DCD Tier 2, Revision 3, Section 15.3.16.1, states that because of these design features, it is unlikely that a major event would result in the release of liquid radioactive wastes into the environment. The approach takes credit for the presence of coated concrete surfaces which contain the volume of the tank in the compartment where the tank is located. The proposed approach is inconsistent with the SRP, which states that "Credit for liquid retention by unlined building foundations will not be given regardless of the building seismic category because of the potential for cracks." The SRP does not allow credit for retention by coatings or leakage barriers outside of the building foundations. Also, DCD Tier 2, Revision 3, states that this design feature applies only to tanks containing "high-level liquid radwaste." The implication is that tanks containing low-level liquid radwaste would not be located in compartments that afford the same level of protection. As a result, the applicant's analysis considers only a single pathway involving the airborne volatilization of radioactivity via the heating, ventilation, and air conditioning (HVAC) system and releases into the environment via the plant stack. Finally, DCD Tier 2, Revision 3, Sections 11.2 and 11.4, emphasize the use of liquid waste processing systems located in treatment bays so as to facilitate truck access and loading and unloading. These design features are in contrast to those designed to minimize spills and leaks into the environment, and it is not clear if the placement of radwaste processing systems in treatment bays provides the same level of protection as that provided for tanks located in cubicles.

In RAIs 15.3-4 and 15.3-5, and in a related supplemental RAI (RAI 2.4-29 S01), the staff asked the applicant to address these inconsistencies with the NRC's guidance and acceptance criteria of SRP Section 11.2 and BTP 11-6, as the SRP precludes the assumption of sealed concrete walls in containing releases of liquid radioactive waste. In addition, the staff requested the applicant to provide additional details on "special design features" to support the approach and update the radiological assessment and to discuss why the release of the postulated inventory of radioactive materials to surface or ground water is not limiting as compared to the current case where only the volatile airborne fraction of radioactivity (as radioiodines) is assumed to be released in the environment. The staff also instructed the applicant to describe the method, basis, assumptions, and parameters used in the analysis; update the text and tables in DCD Tier 2, Revision 3, Section 15.3.16; and update the text and tables as they apply to DCD Tier 2, Section 2.4.13 and Table 2.0-2 of DCD Tier 2, Revision 3.

In its response, the applicant agreed that BTP 11-6 does not allow credit for sealing concrete

walls to contain releases of liquid wastes from tanks, and it committed to use steel liners in cubicles where liquid radwaste tanks are located. The commitment also includes provisions, where sumps are located in tank cubicles, to pump liquids from such sumps to the radwaste system for processing. The applicant is updating Section 11.2.2.3 of DCD Tier 2, Revision 4, to indicate that rooms where tanks are located will be lined with steel to prevent accidental releases of radioactivity in the environment. Similar revisions are proposed for DCD Tier 2, Revision 4, Sections 15.3.16.1 and 12.2.1.4. The staff finds that the inclusion of a steel liner in tank cubicles and the use of sumps to collect and pump liquids to the radwaste system are acceptable mitigating features, consistent with BTP 11-6 and RG 1.143, and in compliance with GDC 60 in controlling releases of radioactivity in the environment during normal operations and AOOs. In its RAI response, the applicant committed to placing this information in DCD Tier 2, Revision 4. RAIs 15.3-4 and 15.3-5 are confirmatory items.

Given that the proposed design precludes the likelihood of a release of radioactivity in ground or surface water, the staff evaluated the applicant's analysis that considers the release of the volatile fraction of radioactivity contained in water and comprised of radioiodines, and the impact on members of the public in unrestricted areas, based on the assumptions given in DCD Tier 2, Revision 3, Tables 15.3-17 and 15.3-18. The amounts of radioiodines consist of the cumulative radioactivity inventory contained in seven tanks, ranging in capacity from 4 to 140 cubic meters (m^3) (about 1,100 to 37,000 gallons). The analysis assumes that the entire inventory of radioiodines is released in the radwaste building and vented outdoors, with no credit taken for filtration and treatment. The analysis assumes an atmospheric dispersion factor of 2.0×10^{-3} s/m^3 for a receptor located at the exclusion area boundary (EAB). The applicant's results, presented in DCD Tier 2, Revision 3, Table 15.3-19, indicate an inhalation dose (TEDE) of 0.07 rem (0.7 millisievert (mSv)) for the offsite receptor. The staff confirmed the result and concludes that the dose complies with the dose limit of 0.1 rem (1 mSv) for members of the public under 10 CFR 20.1301.

The staff finds that the inclusion of special design features to mitigate the consequences of a failure of a tank and its associated components is acceptable, consistent with BTP 11-6 and RG 1.143, and in compliance with GDC 60 in controlling releases of radioactivity into the environment. The basis for the staff's acceptance is the capability of these design provisions to prevent radioactivity from entering a potable water supply system and to prevent the plant from exceeding the limits of 10 CFR Part 20, Appendix B, Table 2 (Column 2), in the nearest source of potable water located in an unrestricted area. The applicant's alternate analysis of a postulated failure of a tank indicates that doses to members of the public from the release and inhalation of volatile radioiodines complies with the dose limit of 10 CFR Part 20. Therefore, the staff concludes that the design provisions incorporated by the applicant are acceptable in mitigating the effects of the failure of a tank and its associated components involving radioactive liquids.

15.3.16.4 Conclusions

Pending the staff's closure of the two confirmatory items discussed above, the staff finds that the analysis and impact of a postulated failure of a tank and its components, located outside of containment (as a permanently installed system), are consistent with the NRC's requirements and guidance. The applicant has met the requirements of GDC 60 with respect to the control of releases of radioactive materials to the environment by providing controls to reduce the potential impact of the failure of a radioactive liquid-containing tank and its associated

components. Such a release will not result in concentrations of radioactive materials exceeding the limits of 10 CFR Part 20, Appendix B, Table 2 (Column 2), in the nearest source potable water located in an unrestricted area.

The staff concludes that the applicant has evaluated the postulated failure of a tank and its associated components and that the design is acceptable, meets the requirements of GDC 60 for the control of releases of radioactive materials to the environment, and provides an adequate level of safety during normal reactor operation, including AOOs. Based on the above review, the staff determines that the ESBWR LWMS design meets the guidelines of SRP Section 11.2 and BTP 11-6 and, therefore, is acceptable.

15.3.17 Combined License Information Summary

The potentially limiting IEs will be evaluated for the (COL applicant) initial core and (COL holder) subsequent reload core design changes:

- loss of FW heating with failure of SCRRI—generator load rejection with total turbine bypass failure,
- COL applicant—confirm the applicability of the startup control rod withdrawal error analysis to the initial core design,
- COL holder—confirm the applicability of the generic radiological dose assessment for misloaded fuel bundles to the site meteorological characteristics,

This was identified as a COL information item in DCD Tier 2, Revision 3, in Section 15.2.0.

In RAI 15.3-2, DCD Tier 2, Revision 3, Section 15.2.0 lists several “COL applicant assumptions” that are applied in the TRACG calculations. A COL information item was provided in DCD Tier 2, Section 15.2.7 for the COL applicant to confirm the applicability of these assumptions. Since the assumptions are also applied to DCD Tier 2, Sections 15.3 and 15.5.5, similar COL information items should be added to DCD Tier 2, Sections 15.3.17 and 15.5.8 for completeness.

15.4 Analysis of Accidents

15.4.1 Design-Basis Accidents

In DCD Tier 2, Section 15.4, “Analysis of Accidents,” the applicant performed radiological consequence assessments of the following six reactor design basis accidents (DBAs) using the hypothetical set of atmospheric dispersion factors (χ/Q values) provided in DCD Tier 1, Table 5.1-1, and DCD Tier 2, Table 2.0-1. Given that all other aspects of the design are fixed, these χ/Q values determine the required minimum distances to the EAB and the low-population zone (LPZ) for a given site in order to provide reasonable assurance that the radiological consequences of a DBA will be within the dose limits specified in 10 CFR 50.34(a)(1)(ii)(D). The analyzed DBAs in DCD Tier 2 include the following:

(1) fuel-handling accident (Section 15.4.1);

(2) loss-of-coolant accident (Section 15.4.4);
(3) main steamline break outside containment (Section 15.4.5);
(4) failure of small lines carrying primary coolant outside containment (Section 15.4.8);
(5) failure of reactor water cleanup system line outside containment (Section 15.4.9); and
(6) feedwater line break outside containment (Section 15.4.7).

Both SRP Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," and RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" list the BWR control rod drop accident as a DBA and require its radiological consequences to be analyzed. In DCD Tier 2, Section 15.4.6, "Control Rod Drop Accident," the applicant stated that the radiological consequence of a control rod drop accident need not be considered because such an accident is extremely unlikely with the improved design of the ESBWR and, furthermore, there is no technical basis for the control rod drop to occur.

The staff has considered GEH's control rod drop event frequency evaluation provided in response to RAI 4.6-23 S01. Based upon the potential consequences of an unrestricted reactivity excursion and to ensure compliance with 10 CFR 50 Appendix A GDC 28, the staff concludes that the ESBWR design must demonstrate reactor coolant pressure boundary integrity and acceptable radiological consequences for the control rod drop accident. More detailed regulatory criteria and guidance is provided in SRP Section 4.2 Appendix B. This regulatory position necessitates updates to ESBWR DCD Tier 1 Section 2.2.2, Tier 2 Section 4.6, and Tier 2 Section 15.4.6. **RAI 4.6-23 is being tracked as an open item.**

In DCD Tier 2, Section 15.4.10, "Spent Fuel Cask Drop Accident," the applicant stated that the radiological consequences of a spent fuel cask drop accident need not be considered because the fuel building (FB) design is such that a spent fuel cask drop height of 9.2 meters (30 feet) cannot be exceeded. SRP Section 15.7.5, "Spent Fuel Cask Drop Accidents," requires a design-basis radiological consequence analysis only if a cask drop exceeding 30 feet can be postulated. In response to RAI 15.4-5, the applicant provided FB figures showing spent fuel cask movements and lifting heights. They are within the 9.2-meter height limit specified in SRP Section 15.7.5. The staff agrees with the applicant's response and neither the staff nor the applicant analyzed the radiological consequences for a spent fuel cask drop accident. RAI 15.4-5 is resolved.

In DCD Tier 2, Section 15.4.7, "Feedwater Line Break Outside Containment," the applicant provided its radiological consequence analysis. The staff considers the radiological consequence resulting from this DBA to be bounded by that resulting from the main steamline break (MSLB) accident outside containment for all light-water BWRs; therefore, it is neither listed in nor required by SRP Section 15.0.3 or RG 1.183. Nevertheless, the staff reviewed the applicant's analysis and finds that the calculational methods used for the radiological consequence assessment are reasonable and acceptable. Furthermore, the staff confirmed that the radiological consequences calculated by the applicant are indeed bounded by those resulting from the MSLB accident outside containment as analyzed by the applicant for the ESBWR.

In DCD Tier 2, Section 15.4.9, "Reactor Water Cleanup (RWCU)/Safe Shutdown Cooling (SDC) Line Failure Outside Containment," the applicant provided its radiological consequence

analysis. This event is not listed as a DBA in either SRP Section 15.0.3 or RG 1.183 and is not required to be analyzed for its radiological consequences. During promulgation of Appendix A to 10 CFR Part 52, the Advisory Committee on Reactor Safeguard (ACRS) Subcommittee on ABWR, specifically recommended that the applicant analyze this event as a DBA to determine the radiological consequences. Accordingly, the applicant analyzed this event for the radiological consequences for the ESBWR, and the staff reviewed this event as analyzed and documented by the applicant.

Therefore, the staff concludes that the selection of six DBAs identified above and analyzed by the applicant are consistent with those identified in SRP Section 15.0.3 and RG 1.183, and therefore, finds it to be acceptable.

In DCD Tier 2, Section 15.4, the applicant concluded that the ESBWR design will provide reasonable assurance that the radiological consequences resulting from any of the above five DBAs will be within the offsite dose criteria, specified in 10 CFR 50.34(a)(1)(ii)(D), of 0.25 sievert (Sv) (25 rem) total effective dose equivalent (TEDE) and the CR operator dose criterion, specified in GDC 19, "Control Room," of Appendix A to 10 CFR Part 50, of 0.05 Sv (5 rem) TEDE. The applicant reached this conclusion by using reactor accident source terms based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and RG 1.183 and a set of hypothetical χ/Q values (see Section 2.3.4 of this report for more detail). No specific site is associated with the ESBWR design.

The χ/Q values are the relative atmospheric concentrations of radiological releases at the receptor point in terms of the rate of radioactivity release. In lieu of site-specific meteorological data, the applicant provided a reference set of hypothetical χ/Q values for the ESBWR. DCD Tier 1, Table 5.1-1, and DCD Tier 2, Table 2.0-1, list the ESBWR hypothetical χ/Q values. The χ/Q values indicate the atmospheric dilution capability. Smaller χ/Q values are associated with greater dilution capability, resulting in lower radiological doses. The radiological consequence doses are directly proportional to the χ/Q values. The hypothetical χ/Q values in the DCD are back calculated from the dose acceptance criteria to minimize the fission product removal credit assumed for the engineered safety feature (ESF) systems in the ESBWR design.

Therefore, any COL or construction permit (CP) applicant that references the ESBWR design should show that its proposed site-specific χ/Q values fall within the reference set of hypothetical χ/Q values used by the applicant in DCD Tier 2 in order to demonstrate that the COL or CP application meets the offsite dose criteria specified in 10 CFR 50.34(a)(1)(ii)(D) and the CR operator dose criterion specified in GDC 19 of Appendix A to 10 CFR Part 50. This was identified as a COL information item in DCD Tier 2, Section 2.0.1.

15.4.2 Fuel-Handling Accident

15.4.2.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Section 15.4.1, "Fuel Handling Accident," in accordance with SRP Section 15.0.3 and RG 1.183. The staff evaluated the radiological consequences of a fuel-handling accident (FHA) against the dose acceptance criteria, specified in SRP Section 15.0.3 and RG 1.183, of 0.063 Sv (6.3 rem) TEDE at the EAB for any 2-hour period following the onset of the postulated fission product release and 0.063 Sv (6.3 rem) TEDE at

the outer boundary of the LPZ for the duration of exposure to the release cloud. The staff used a criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences from a postulated FHA in the CR of the ESBWR design, pursuant to GDC 19 of Appendix A to 10 CFR Part 50.

RG 1.183 provides guidance on radiological consequence analyses to licensees of operating power reactors that choose to implement an alternative source term (AST) pursuant to 10 CFR 50.67, which specifies the same regulatory dose criteria as 10 CFR 50.34(a)(1)(ii)(D) (0.25 Sv (25 rem) TEDE) and GDC 19 (0.05 Sv (5 rem) TEDE). Although RG 1.183 was written to apply to currently operating power reactors, the staff finds that its guidance on radiological acceptance criteria, formulation of the source term, and DBA radiological consequence analysis modeling also applies in the review of the ESBWR design.

15.4.2.2 Summary of Technical Information

In DCD Tier 2, Section 15.4.1, the applicant presented its analyses of the radiological consequences of a postulated FHA. An FHA is postulated to result from a failure of the fuel assembly lifting mechanism, leading to a raised fuel assembly being dropped onto the reactor core or into the spent fuel storage pool. Any fission products released as a result of a fuel assembly drop in the refueling pool will be released into the reactor building (RB) atmosphere and then to the environment. Fission products released as a result of a fuel assembly drop onto the reactor core are assumed to be released directly to the environment by means of the cask doors on the west side of the FB. The applicant provided the FHA radiological consequence analyses for a fuel assembly drop onto the reactor core and into the spent fuel storage pool in its response to RAI 15.4-1, Supplement 1. The applicant assumed, in accordance with the guidance in RG 1.183, that fission products are directly released to the environment within a 2-hour period without credit for any fission product removal processes.

15.4.2.3 Staff Evaluation

The staff has reviewed the applicant's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria specified in SRP Section 15.0.3 and RG 1.183.

The applicant conservatively postulated that a total of two spent fuel assemblies incur damage to the cladding of all fuel rods. One fuel assembly is dropped either into the spent fuel storage pool or onto the reactor core impacting fuel assemblies (equivalent to one fuel assembly) in the pool or in the reactor core. The applicant assumed that these two damaged fuel assemblies had undergone 24 hours of decay time and that all fission products in the gap of every rod in the two damaged fuel assemblies were instantaneously released. The ESBWR TS 3.9.7, "Decay Time," requires the reactor to be subcritical for at least 24 hours before refueling operation. Therefore, the FHA could occur no earlier than 24 hours following reactor shutdown. The applicant assumed a radial peaking factor of 1.7 for the damaged rods. The kinetic energy developed in this drop is conservatively assumed to be dissipated in damage to the cladding of all fuel rods in two fuel assemblies. All fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident.

Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods (8 percent of iodine-131, 10 percent of krypton-85, and 5 percent of other iodine and noble gas are assumed to be in all fuel rod gaps in the reactor core) is assumed to occur, with the released gases bubbling up through the fuel pool water (with an effective decontamination factor of 200 for total iodine). These gap fractions and the effective decontamination factor used are consistent with the guidance provided in RG 1.183. The applicant assumed that iodine in the particulate form is not volatile; therefore, it is not released. In accordance with the RG 1.183 guidance, the applicant assumed that the particulate cesium iodide (CsI) is instantaneously converted to the elemental form of iodine when it is released from the fuel into the pool water.

The applicant assumed that the CR will not be isolated during the postulated FHA and the control room emergency filtration unit (CREFU) will not be operational. The applicant further assumed that the normal control room area ventilation system (CRAVS) will be operational during this event with no credit for fission product removal. The applicant used a normal CRAVS flow rate of 200 liters per second (l/s) (424 cubic feet per minute (cfm)) as an unfiltered air in-leakage rate into the CR envelope for conservatism.

The applicant evaluated the maximum 2-hour TEDE to an individual located at the EAB, the 30-day TEDE to an individual at the outer boundary of the LPZ, and the 30-day TEDE to an individual in the CR. The resulting doses are less than the dose acceptance criteria specified in RG 1.183 and SRP Section 15.01. The staff performed an independent confirmatory dose calculation and found the staff's results are in agreement with the applicant's values. Both, the applicant and the staff's results met the relevant dose acceptance criteria at the EAB, LPZ, and CR.

15.4.2.4 Conclusion

The staff concludes that the ESBWR design, as bounded by the hypothetical χ/Q values proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated fuel handling accident at the EAB and LPZ will be well within the dose criteria set forth in 10 CFR 50.34 (i.e., 25 percent or 0.063 Sv (6.3 rem) TEDE) and that the radiological consequences to an individual in the CR as a result of a postulated fuel handling accident will be within the dose criterion set forth in GDC 19 (0.05 Sv (5 rem) TEDE). Therefore, the staff finds the radiological consequence analysis provided by the applicant to be acceptable.

15.4.3 Loss-of-Coolant Accident

15.4.3.1 Regulatory Criteria

The requirements summarized in Section 15.1.1.3 are used in evaluating this DBA.

The staff reviewed DCD Tier 2, Section 15.4.4, "Loss-of-Coolant Accident Inside Containment Radiological Analysis," in accordance with SRP Section 15.0.3 and RG 1.183.

The staff evaluated the radiological consequences of a LOCA against the dose criteria specified in 10 CFR 50.34(a)(1)(ii)(D) of 0.25 Sv (25 rem) TEDE at the EAB for any 2-hour period

following the onset of the postulated fission product release and 0.25 Sv (25 rem) TEDE at the outer boundary of the LPZ for the duration of exposure to the release cloud. The staff used a criterion of 0.05 Sv (5 rem) TEDE to evaluate the radiological consequences from DBAs in the CR of the ESBWR design, pursuant to GDC 19 of Appendix A to 10 CFR Part 50. The staff used the applicable guidance in RG 1.183 in its review of the ESBWR DBA radiological consequence analyses.

15.4.3.2 Summary of Technical Information

In DCD Tier 2, Section 15.4.4, the applicant analyzed a hypothetical design-basis LOCA. The applicant concluded that certain bounding sets of hypothetical χ/Q values specified in DCD Tier 1, Table 5.1-1, and DCD Tier 2, Table 2.0-1, in conjunction with the use of the passive containment cooling system (PCCS) in the containment, the natural deposition of fission product aerosol in the containment, an essentially leaktight containment barrier, and the control of the pH of the water in the containment to prevent iodine evolution, are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated design basis LOCA will be within the relevant dose criteria established in 10 CFR 50.34 and GDC 19.

To support its conclusion, the applicant submitted the following licensing technical topical report and three research reports as its enclosures to supplement DCD Tier 2, Section 15.4.4:

- Licensing Topical Report, NEDE-33279P, Revision 1, "ESBWR Containment Fission Product Removal Evaluation Model" (GE Licensing Topical Report), dated August 2007.

- Research Report, VTT-R-04413-06, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment—Part 1" dated October 2006 (hereafter referred to as the VTT report),

- Research Report, VTT-R-04413-06, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment—Part 2", dated December 2006 (hereafter referred to as the VTT report),

- Research Report, VTT-R-06771-07, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment—Part 3" dated October 2007 (hereafter referred to as the VTT report).

The staff found that the key aspects concerning fission product distribution, transport, and removal following the postulated LOCA involve (1) the PCCS operation, (2) natural deposition of fission product aerosol within the containment, and (3) control of the pH of the water in the containment to prevent iodine evolution. The four reports listed above provide information on these aspects. Therefore, the staff requested, in RAI 15.4-6, that the applicant incorporate the radiological consequence analyses provided in these reports into DCD Tier 2, Section 15.4.4 or incorporate the reports into DCD Tier 2, Chapter 15 as appendices. **RAI 15.4-6 is being tracked as an open item.**

The applicant postulated the following three LOCA scenarios:

(1) RPV bottom drain line break with automatic depressurization system (ADS) operating and degraded low-pressure makeup system,

(2) RPV bottom drain line break with ADS failure and degraded high-pressure makeup system,

(3) loss of preferred power with ADS operating and degraded low-pressure makeup system.

The applicant originally proposed accident scenarios 1 and 2 above, stating that the reactor core uncovery and fission product release timing is shortest. For accident scenario 1, the use of a fully depressurized, low-pressure accident sequence in conjunction with the source term described in NUREG-1465 is appropriate because the release fractions for the source terms presented in NUREG-1465 are intended to be representative or typical of those associated with a low-pressure core melt accident. Accident scenarios 1 and 2 have the same initiating event with different accident sequences.

The staff accepted the accident scenarios proposed by the applicant but requested the applicant to add one additional accident sequence, "loss of preferred power with ADS operating and degraded low-pressure makeup system," since it is the dominant contributor to the core damage frequency for the ESBWR. The applicant accepted the staff's request and agreed to evaluate the above three accident scenarios as representative of the spectrum of ESBWR LOCAs.

In RAI 15.4-17, the staff requested that the applicant describe each of the above three LOCA accident scenarios in more detail, complete with the sequence of events; operation and availability of the ESF systems, including the suppression pool; fission product transport pathways; and fission product release timing. **RAI 15.4-17 is being tracked as an open item.**

The applicant performed and provided the radiological consequence analysis only for accident scenario 1 above. In RAI 15.4-7, the staff requested that the applicant complete and provide the same radiological consequence analyses for accident scenarios 2 and 3 above as it did for accident scenario 1. The staff asked the applicant to incorporate these two remaining radiological consequence analyses into NEDE-33279P and DCD Tier 2, Section 15.4.4. In addition, the staff requested that the applicant compare and discuss the results of the radiological consequences and fission product removal rates in the containment for all three accident scenarios. **RAI 15.4-7 is being tracked as an open item.**

DCD Tier 2, Section 15.4.4 stated that the applicant's radiological consequence analyses are based on the NUREG-1465 alternative source terms (ASTs) and the methodology in RG 1.183. On the other hand, the applicant also stated in DCD Tier 2, Section 15.4.4.2.1, that the core remains covered throughout the accident, and there is no fuel damage. The statement in DCD Tier 2, Section 15.4.4.2.1, is inconsistent with NUREG-1465 and RG 1.183. In RAI 15.4-8, the staff requested that the applicant rectify the inconsistencies in these statements. Specifically, the staff asked the applicant to review NEDE-33279P and Section 15.4.4 to ensure that no further discrepancies exist. **RAI 15.4-8 is being tracked as an open item.**

In response to RAI 15.3-25, the applicant provided in tabular form a complete fission product

inventory of the core at 4590 megawatts thermal (MWt), considering a licensing power 2 percent above the design level, along with its technical bases in DCD Tier 2, Section 15.3. In RAI 15.4-9, the staff requested that the applicant include this information in DCD Tier 2, Section 15.4.4. The applicant revised Appendix 15B of DCD Tier 2, Revision 3, to include information complying with RG 1.183, Section 3.1 for core thermal power, fuel bum-up, and fuel enrichment. The staff agrees with this response. RAI 15.4-9 is resolved.

All of the fission product releases caused by a postulated LOCA are either the result of containment atmosphere leakage, RB leakage, or main steamline isolation valve leakage. The ESBWR design does not have ESF systems outside of the containment; therefore, the applicant did not consider leakage from the ESF systems as part of its radiological consequence analysis.

The ESBWR containment consists of a drywell, a wetwell, a PCCS, and supporting systems to limit fission product leakage following a postulated LOCA, with rapid isolation of all pipes and ducts that penetrate the containment boundary. It is designed to prevent the uncontrolled release of fission products to the environment. The applicant stated that the containment will be built and tested periodically to ensure a leak rate at design pressure of less than 0.5 percent by weight per day (wt%/d) at the calculated peak containment pressure associated with a LOCA for the entire duration of the accident (30 days). Both, the applicant and the staff used this leak rate in its respective radiological consequence analysis.

The applicant stated that 2 percent of this 0.5 wt%/d containment leak rate (0.01 wt%/d overall containment leak) is assumed to leak through the PCCS into the air space directly above the PCCS and subsequently leak directly to the environment without mixing with the RB atmosphere (RB bypass). In response to RAI 16.4-11, the applicant provided the information requested by the staff concerning the fission product source term through this pathway and confirmed that this leakage will be included as an inspection, test, analysis, and acceptance criterion (ITAAC) item and in the TSs as a surveillance requirement. RAI 15.4-11 is resolved. The ESBWR design provides neither an ESF filtration (e.g., charcoal adsorbers) nor a safety-related containment spray system in the containment.

The RB is a reinforced concrete structure that forms an envelope completely surrounding the containment and is designed to seismic Category 1 criteria. The RB isolation is designed to be tested under accident conditions. During normal plant operation, the potentially contaminated areas of the RB are maintained at a slightly negative pressure relative to the adjoining areas by exhausting the RB air through the non-safety-related normal RB HVAC system. Following a postulated DBA, the RB is automatically isolated to provide a holdup for the decay of airborne fission products. The normal RB HVAC system will continue to operate following the postulated LOCA, if power is available. Neither the applicant nor the staff claimed fission product mitigation by the normal RB HVAC system.

The applicant assumed, however, that 40 percent of the RB volume will be available for mixing for holdup and decay of fission products before leaking from the RB and that an overall RB leakage rate will be less than 50 percent per day. The staff is currently reviewing the technical basis proposed by the applicant for the 40-percent mixing of the RB volume, and the staff's evaluation and open items regarding this subject are provided in Section 6.2.3.3 of this report.

The applicant stated that the RB envelope is not intended to provide a leaktight barrier against

radiological fission product release; however, the RB is capable of periodic testing to ensure that the leakage rates assumed (50 percent per day) in the radiological consequence analyses are met. The staff requested in its RAI 15.4-26 that the applicant (1) identify the flowpaths to be isolated and the method to be used to verify the leak rate, (2) state whether the leakage rate test to meet the 50-percent-per-day limit is specified in the ESBWR TS, and (3) include this leak rate verification in Tier 1 as an ITAAC item to be confirmed at the COL stage. In its response to RAI 15.4-26, the applicant identified the flowpaths to be isolated and the method to be used to verify the leak rate and stated that the leakage rate test to meet the 50-percent-per-day limit is specified in the ESBWR TS 3.6.3.1.4. The applicant included this leak rate verification in DCD Tier 1, Table 2.16.5-2, Revision 4, as an ITAAC item. Therefore RAI 15.4-26 is resolved.

The PCCS is designed to remove decay heat and fission products from the containment atmosphere following a postulated LOCA. The PCCS heat exchangers receive a steam-gas mixture and airborne fission products from the drywell atmosphere, condense the steam, and return the condensate with condensed fission products to the RPV though the GDCS pools. The noncondensibles, including noble gases and volatile fission products, are drawn to the suppression pool through a submerged vent line driven by the differential pressure between the drywell and wetwell. The noncondensibles will become airborne into the wetwell air space and flow back into the drywell during vacuum breaker openings. In RAI 15.4-7, the staff requested additional information concerning fission product removal rates by the PCCS heat exchangers for accident scenarios 2 and 3. **RAI 15.4-7 is being tracked as an open item.**

NEDE-33279P includes the radiological consequence analyses, complete with fission product removal rates in the containment, for accident scenario 1. In RAI 15.4-7, the staff asked the applicant to complete and provide the same radiological consequence analyses, including fission product removal rates in the containment and associated pH evaluation, for accident scenarios 2 and 3, as discussed in Section 1.3 of NEDE-33279P. The staff also asked the applicant to incorporate these two remaining radiological consequence analyses into NEDE-33279P and Section 15.4.4 and to compare and discuss the results of the radiological consequences and fission product removal rates in the containment for all three accident scenarios. **RAI 15.4-7 is being tracked as an open item.**

The ESBWR design provides a suppression pool to condense steam and remove fission products following a postulated LOCA. The sequence of a postulated LOCA include, among other things, the operation and availability of the suppression pool as a passive fission product control and removal system. The accident scenarios evaluated involve the reactor bottom drain line breaks that result in a blowdown of the RPV liquid and steam to the drywell by means of the severed pipe. The resulting pressure buildup drives the mixture of steam, water, and other gases down through vents to the downcomers and into the suppression pool water, thereby condensing the steam and reducing the pressure. Because of the postulated loss of core cooling, the fuel heats up and melts, resulting in the release of fission products.

The fission product release occurs in phases over a 2-hour period. The initial blowdown to the suppression pool will not include significant quantities of fission products. Subsequent fission product releases from reactor safety valves to the suppression pool will remove some fission products by the suppression pool water. The applicant assumed a decontamination factor of 10 for any particulate fission product and for iodine in its elemental form. In RAI 16.4-7, the staff requested additional information regarding fission product removal rates by the

suppression pool as a function of time (i.e., for a period of 30 days) for accident scenarios 2 and 3. **RAI 15.4-7 is being tracked as an open item.**

The applicant assumed leakage of the MSIVs at the TS limit of 0.0623 cubic meter per minute total (200 cubic feet per hour). In RAI 15.4-10, the staff asked whether the MSIV leakage in the turbine building (TB) is included in the total containment leakage rate of 0.5 wt%/d. The applicant responded to this RAI, stating it did not include the total containment leakage rate of 0.5 wt%/d. RAI 15.4-10 is resolved.

The main steamlines are classified as seismic Category 1 from the RPV interface to the outboard seismic restraint outboard of the downstream MSIV. The steamlines and their associated branch lines outboard of the last RB seismic restraint, including the main steam drainlines, are dynamically analyzed to safe-shutdown earthquake (SSE) conditions that determine the flexibility and structural capabilities of the lines under SSE conditions.

The main condensers are also dynamically analyzed to SSE conditions to ensure that fission products leaked through the MSIVs are enclosed. In its response to RAI 15.4-19, the applicant stated that (1) the main steamlines and drainlines are designed to meet SSE criteria and analyzed to dynamic loading criteria, (2) the MSIV fission product leakage path to the main condenser is analyzed to demonstrate structural integrity under SSE loading conditions, and (3) the ITAAC in DCD Tier 2, Chapter 12, Table 2.11.1-1, now require the turbine main steam system piping and MSIV fission product leakage path to be able to withstand an SSE without loss of structural integrity. This response satisfies RAI 15.4-19 and, therefore, RAI 15.4-19 is resolved.

15.4.3.3 Staff Evaluation

15.4.3.3.1 Accident Source Terms

In SECY-94-302, “Source Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs,” dated December 19, 1994, the staff proposed to use only the “coolant,” “gap,” and “early in-vessel” releases from NUREG-1465 for the radiological consequence assessments of DBAs for the passive advanced light-water reactor (ALWR) designs. These source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel. These scenarios represent the most severe reactor accidents from which the plant could be expected to return to a safe-shutdown condition. As stipulated in 10 CFR 50.34(a)(1)(ii)(D), an applicant performing a radiological consequence of accident analysis shall assume a fission product release from the core into the containment. Note 6 to this regulation states the following:

> The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

The staff considered the inclusion of the “ex-vessel” and the “late in-vessel” source terms in

NUREG-1465 to be unduly conservative for DBA purposes. Such releases will only result from core damage accidents with vessel failure and core-concrete interactions. For passive ALWRs, the estimated frequencies of such scenarios are low enough that they need not be considered credible for the purpose of meeting the requirements of 10 CFR 50.34. In SECY-94-302, the Commission approved the staff-recommended technical position to use only the coolant, gap, and early in-vessel releases from NUREG-1465 for the radiological consequence assessments of DBAs for passive ALWR designs.

The objective of NUREG-1465 is to define an accident source term for regulatory application for future light-water reactors (LWRs). The intent was to capture the major relevant insights available from severe accident research to provide a more realistic portrayal of the amount of the postulated accident source term. These source terms were derived from examining a set of severe accident sequences for LWRs of current design. Because of general similarities in plant and core design parameters, these results are considered to be applicable to passive LWR designs. The NRC has used this source term in evaluating the Westinghouse AP600 and AP1000 standard reactor design certification applications.

RG 1.183 provides guidance to licensees of operating power reactors on acceptable applications of ASTs pursuant to 10 CFR 50.67. This RG establishes an acceptable AST based on insights from NUREG-1465 and establishes the significant attributes of other alternative source terms that may be found acceptable by the NRC staff for operating LWRs. RG 1.183 also identifies acceptable radiological analysis assumptions for use in conjunction

with the accepted AST for operating power reactors. The applicant followed the applicable guidance in RG 1.183 for the ESBWR design.

15.4.3.3.2 Radiological Consequence Analysis

In DCD Tier 2, Section 15.4.4, the applicant analyzed a hypothetical design-basis LOCA. The applicant concluded that certain bounding sets of hypothetical χ/Q values specified in DCD Tier 1, Table 5.1-1, and DCD Tier 2, Table 2.0-1, in conjunction with the use of the passive containment cooling system (PCCS) in the containment, the natural deposition of fission product aerosol in the containment, an essentially leaktight containment barrier, and the control of the pH of the water in the containment to prevent iodine evolution, are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated design-basis LOCA will be within the relevant dose criteria established in 10 CFR 50.34 and GDC 19.

The staff will complete its independent confirmatory radiological consequence analyses when the remaining open items are unresolved.

All of the fission product releases resulting from the postulated LOCA to the environment are the result of (1) primary containment atmosphere leakage, (2) main steamline isolation valve leakage, and (3) RB leakage. The ESBWR design does not have ESF systems outside containment. Therefore, the radiological consequence analyses do not consider leakage from the ESF systems.

15.4.3.3.2.1 <u>Primary Containment Atmosphere Leakage</u>

The ESBWR design does not provide an active containment atmosphere cleanup system. Instead, the design relies on natural aerosol removal processes for deposition in the containment structural surfaces and the PCCS condensers, such as gravitational settling and plateout through diffusiophoresis and thermophoresis. The GEH LTR NEDE-33279P and VTT reports discuss the removal of airborne activity from the containment atmosphere. The applicant provided a non-safety-related containment spray system as part of the fuel and auxiliary pool cooling system. The system can be manually operated 72-hours following a LOCA to aid in post-accident recovery. However, the containment spray system is not credited for mitigation of radiological releases in the radiological consequence assessment.

(1) Iodine Removal

The ESBWR passive containment design utilizes a unique passive containment cooling system in order to transport decay heat from a damaged reactor core to a water-pool heat sink and thereby to reduce the containment pressure. Following any initial pressure transients associated with reactor vessel blowdown, long term heat rejection in the ESBWR is accomplished by heat rejection to the PCCS water pools by the flow of steam drawn into the cool condenser tubes of the PCCS. Steam produced by boiling in the reactor vessel enters the containment by way of the open direct pressure vent lines and flows into the PCCS. The condensate from the PCCS returns to the gravity drain condensate system (GDCS) pool and subsequently returns to the reactor vessel. Thus, water is maintained in the reactor vessel by a supply from the GDCS pool.

While the design of the PCCS should prevent reactor core damage, the applicant assumed substantial meltdown of the core as a result of postulated LOCA with subsequent release into the containment of appreciable quantities of fission products as stipulated in 10 CFR 50.34(a)(1) and in 10 CFR 100.21.

Condensation occurring in the PCCS tubes driven by the boiling of water in the reactor vessel serves a very effective means of scrubbing radioactive aerosol in the drywell, and in time most of drywell aerosol will be captured in the PCCS condensate. Since the PCCS condensate drains back into the reactor vessel, most of radioactive aerosol will reside in the water of the reactor vessel. NUREG-1465 specifies that, after an accident, iodine entering the containment from the reactor core is composed of at least 95-percent cesium iodine (CsI), with the remaining 5 percent comprising elemental iodine and a small amount of hydriodic acid. However, about 3 percent of elemental iodine in contact with some organic compounds will produce organic iodides. Therefore, the iodine in the containment will consist of 95-percent particulate iodine as CsI, 4.85-percent elemental iodine (I_2), and 0.15-percent organic iodine. The composition of the iodine in the ESBWR is consistent with the composition specified in NUREG-1465 and RG 1.183.

Both gaseous and particulate iodine can be scrubbed from the drywell in the PCCS condenser tubes and delivered back to the reactor vessel by the draining condensate. Within the boiling vessel, the cesium iodine in particulate/aerosol form will subsequently dissociate to form Cs^+ and I^-. Here, the aqueous I_2 and methyl iodine (CH_3I) together with the dissociated I^- may undergo complex chemical reactions in the high radiation environment of the boiling reactor vessel, producing a wide range of chemical and ionic forms of iodine, including volatile I_2. Dissolved I_2, much of which was originally in the form of CsI when initially released to the containment atmosphere, can subsequently return to the containment atmosphere as gaseous

iodine at the surface of the water pool in the reactor vessel, and subsequently be carried to the containment atmosphere by the steam leaving the reactor vessel.

This ionized iodine again flows into the PCCS, where it can be dissolved into the condensate and re-introduced to the reactor vessel. Therefore, there is a continuous refluxing of iodine from the PCCS to the reactor vessel, and from the reactor vessel back into the containment atmosphere, and back into the PCCS tubes. Meanwhile, airborne volatile iodine in the containment atmosphere will be adsorbed on the walls and wetted surfaces of the containment and removed by gravitational settling and plateout through diffusiophoresis and thermophoresis. The staff believes that the combination of production (sources) and removal (sinks) will lead to a steady-state concentration of gaseous iodine in the containment atmosphere that will leak to the environment at a design basis leak rate.

In RAI 15.4-29, the staff requested that the applicant explain the iodine transport phenomena in the ESBWR containment and perform a rate analysis of steady-state iodine transport within the containment including iodine re-volatilization (source) from the reactor vessel and iodine removal by the PCCS condenser and by natural deposition (sink). **RAI 15.4-29 is being tracked as an open item.**

(2) Aerosol Removal

Applying credit for aerosol removal through the PCCS requires input from thermal-hydraulic (T-H). The basis document defining the revised accident source term, NUREG-1465, does not specify an associated T-H scenario, methodology, or acceptance criteria for fission product removal. The AST regulatory guidance, RG 1.183, also does not specify these items. NUREG-1465 describes a source term derived from an examination of a set of severe accident sequences for LWRs and is intended to be representative or typical and does not imply a specific scenario, much less the worst case.

In the past, the staff and industry evaluated aerosol removal through well-established models of spray removal or condensation. The ESBWR application relies on natural deposition processes in the PCCS that depend strongly on local T-H conditions. While gravitational settling is relatively easy to understand, aerosol removal through diffusiophoresis and thermophoresis is much more complex. Diffusiophoresis is associated with steam condensation on the heat sinks and depends on the condensation steam mass flux. Thermophoresis relies only on the temperature gradient close to the surface on which the particles will be deposited.

Thermophoresis is more subtle than the other two natural deposition processes. Because the temperature gradient cannot be measured or easily calculated, its model uses the heat flux at the surface divided by the thermal conductivity of the gas adjacent to the surface as an equivalent measure of the driving force. Simultaneous occurrence of the two phoretic processes introduces an additional level of complexity.

The applicant used the MELCOR code to establish T-H boundary conditions and to estimate fission product removal rates in the containment by the PCCS. The MELCOR code is an NRC severe accident code, and it is a fully integrated, engineering-level computer code with the primary purpose of modeling the progression of a severe reactor accident and estimating fission product source terms. In DCD Tier 2, Table 15.4-5, the applicant provided aerosol

removal coefficient values starting at the onset of a gap release through the first 24 hours into a DBA. The values ranged from 0.1 to 5.0 per hour for accident scenario 1. In RAI 15.4-7, the staff requested that the applicant provide aerosol removal coefficient values for accident scenarios 2 and 3. **RAI 15.4-7 is being tracked as an open item.**

The staff conducted an independent evaluation of aerosol removal coefficients. In its evaluation, the staff considered the same natural processes for removing aerosols from the containment atmosphere as described above. These processes include the sedimentation mechanism of gravitational settling, such as aerosol agglomeration, and the phoretic mechanisms of diffusiophoresis and thermophoresis in the PCCS.

For the staff's independent evaluation of aerosol removal coefficients, the staff contracted with Sandia National Laboratory (1) to evaluate aerosol removal coefficients and (2) to perform quantitative analyses of uncertainties in predicting the aerosol removal rates. The analysis used the three accident scenarios described in Section 15.4.3.2 above as the initiating events. As part of the staff's review, the aerosol behavior in the ESBWR containment was predicted using the MELCOR integrated accident analysis code, which includes the MAEROS aerosol mechanics code. The staff used the results of a fully integrated MELCOR analysis of the ESBWR accident scenarios, performed by the staff's contractor, Sandia National Laboratories, to develop a simplified containment model for the uncertainty analysis. The NUREG-1465 radiological source term for the gap release and in-vessel release phases were used in place of the source term predicted in the fully integrated analysis. The uncertainty analysis considered those MELCOR parameters known to affect aerosol settling and depletion to be uncertain within a range of values, represented by an assumed distribution function. A Monte Carlo method, which randomly samples the uncertain parameters and performs a large number of separate MELCOR analyses, estimated the effect of the uncertain physics parameters on the aerosol removal rate.

In its evaluation of aerosol removal rates, the staff (contractor) used the containment geometry (e.g., volume, upward-facing surface area) provided by the applicant and the fission product release timing, fractions, and release rates described in NUREG-1465. The staff's analyses considered the following principal uncertainties in aerosol properties and aerosol behavior:

- aerosol size and distribution,
- aerosol void fraction and particle shape factors,
- aerosol material density,
- nonradioactive aerosol mass,
- particle slip coefficient,
- sticking probability for agglomeration,
- boundary layer thickness for diffusion deposition,
- thermal accommodation coefficient for thermophoresis,
- ratio of thermal conductivity of particle to gas,
- turbulent energy dissipation, and
- multipliers on heat and mass transfer to containment shell.

The applicant's calculation of aerosol removal coefficients is based on an analysis of several T-H scenarios and uses a single aerosol model without providing an uncertainty analysis. The applicant also used T-H conditions associated with the three accident sequences described

above. The staff will confirm the applicant's aerosol removal rates upon receipt of its response to RAI 15.4-7 comparing them with those calculated by the staff. Although the selection of accident scenarios is acceptable, the staff believes an evaluation of the associated uncertainties is necessary. The staff used an alternative T-H code (MELCOR) as an input to a Monte Carlo sampling (150 realizations) of the above-listed parameters affecting aerosol behavior. After several discussions between the staff and the contractor, engineering judgment was used to choose parameters, as well as the range and distribution of their values. The resultant distribution of possible aerosol removal rates will be compared with aerosol removal rates to be provided by the applicant.

(3) Containment Pool Water Chemistry

Iodine in the form of CsI is soluble in the containment pool water. However, some of it may be converted into the elemental form, which can be released into the containment atmosphere. The released radioactive iodine may leak out of the containment atmosphere to the RB and, subsequently, to the environment. To minimize formation of elemental iodine, the pH of the containment water should be kept basic. Basic pH will also prevent stress-corrosion cracking of the stainless steel components.

The ESBWR design includes three water pools—the PCCS pool in the RB and the GDCS pool and suppression pool in the containment. During normal plant operation, the pH of these pools will be between 6 and 7. In RAI 15.4-28, the staff requested that the applicant provide pH values in each pool (i.e., the PCCS pool, GDCS pool, and suppression pool), including the RPV and lower drywell, following the postulated LOCA for the duration of the entire accident period (30 days). **RAI 15.4-28 is being tracked as an open item.**

The pH of the containment sump water after a LOCA is determined by acidic and basic chemical species released to the containment from different sources in the plant. The most significant effect on reducing containment water pH results from the hydrochloric acid produced by radiolytic decomposition of electric cable jackets. In RAI 15.4-14, the staff requested that the applicant provide the amount of cable insulation material used in the ESBWR containment. The applicant responded to this RAI stating that DCD Tier 1, Revision 3, Subsection 2.15.1 and Table 2.15.1-1 will be revised to include exposed cable mass and that DCD Tier 1, Subsection 2.15.1 and Table 2.15.1-1, Revision 4 will include exposed cable mass. Therefore, RAI 15.4-14 is resolved. Other sources of chemical species that are formed in the containment during the 30 days following a core damage accident include nitric acid produced by radiolytic formation from air dissolved in the sump water and cesium hydroxide (CsOH) from the damaged core. CsOH, being a strong base, will contribute to the increase of pH. In RAI 15.4-13, the staff requested that the applicant provide a sensitivity analysis of pH as a function of the amount of CsOH formation. **RAI 15.4-13 is being tracked as an open item.**

15.4.3.3.2.2 Main Steamline Isolation Valve Leakage

The MSIVs automatically isolate the four main steamlines that penetrate the drywell. There are two MSIVs on each steamline, one inside the drywell (i.e., inboard) and one outside the drywell, (i.e., outboard). The MSIVs are functionally part of the primary containment boundary, and design leakage through these valves provides a leakage path for fission products to bypass the RB and enter the environment as a ground-level release.

The applicant assumed that the inboard MSIV failed to close in one of four main steamlines and its outboard MSIV leaks at a maximum allowable MSIV leakage of 200 scfh specified in the ESBWR TS in DCD Tier 2, Chapter 16. The applicant modeled one main steamline with the leak as a single main steamline and the three remaining non-leaking main steamlines were lumped together into one equivalent main steamline. This leak rate is based on a design-basis LOCA maximum peak containment pressure of 48 psig. The applicant did not credit any reduction in drywell pressure or the MSIV leakage rate of 200 scfh after 24 hours following the postulated LOCA. Leakage rates were held constant for the entire duration of the accident (30 days) for conservatism. The ESBWR TS specifies the maximum allowable MSIV leak rate.

The applicant's analysis did not take credit for aerosol and iodine removal in the main steamlines or in the main steam drainlines. The applicant's analysis did take credit for aerosol and iodine removal in the main condensers, referencing BWR Owner's Group Topical Report NEDC-31858P, "BWROG Report for Increasing MSIV Leakage and Elimination of Leakage Control System." In 1996, the staff accepted this topical report in reactor licensing for reactor plants that use the accident source terms specified in the Atomic Energy Commission's Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," issued in 1962. However, the ESBWR design uses the ASTs to meet the radiological consequence evaluation factors as expressed in TEDE as required by 10 CFR 50.34(a)(1) and 10 CFR 100.21, "Non-Seismic Site Criteria." Therefore, the use of TID-14844 accident source terms is no longer acceptable to the staff. In RAI 15.4-22, the staff requested that the applicant provide the model, method, and assumptions used for fission product removal in the main condensers and justify the use of a TID-14844 accident source term for this pathway in estimating its radiological consequences. The applicant has not provided the response to this RAI. **RAI 15.4-22 is being tracked as an open item.**

15.4.3.3.2.3 Reactor Building Leakage

Section 6.2.3, "Reactor Building Functional Design," of DCD Tier 2 describes the RB functional design, including RB leakage.

GDC 16, "Containment Design," states that reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. The applicant stated that GDC 16 does not apply because the RB is not considered to be a leaktight barrier.

The staff is considering the applicant's statement with respect to the applicability of GDC 16. The applicant makes two assumptions in the design-basis radiological consequence analyses that impact the control of fission product release. The first assumption is that the primary containment leakage into the RB is diluted by 40 percent of the RB volume before it is leaked to the environment. The second assumption is that the RB leakage to the environment is 50-percent volume per day. These two assumptions directly affect the results of the design-basis radiological consequence analyses required by 10 CFR 34(a)(1) and the CR operator dose stated in GDC 19. The applicant needs to demonstrate how the ESBWR meets the safety function of controlling fission product releases to the environment if it is not a leaktight barrier.

In a response to RAI 15.4-26, the applicant provided (1) the maximum leak rate that could occur from the RB under design-basis conditions (50-percent volume per day) and (2) the method to be used to test RB leakage. The leakage rate test is specified in the ESBWR TS 3.6.3.1.4 and it is identified in DCD Tier 1, Section 2.16.5, "Reactor Building," as an ITAAC item. Therefore, RAI 15.4-26 is resolved.

Paragraph 4.4 of Appendix A to RG 1.183 states that "[C]redit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise the leakage from the primary containment should be assumed to be transported directly to the exhaust systems without mixing." The applicant has not demonstrated an adequate means to accomplish a 40-percent dilution of the source term before its release from the RB and has not addressed the uncertainties. RAIs 6.2-165 address this issue. **RAIs 6.2-165 is being tracked as an open item** See section 6.2.3.3 of this report for the staff's evaluation of the reactor building design.

15.4.3.3.2.4 Control Room Radiological Consequence Analysis

In DCD Tier 2, Section 15.4, "Analysis of Accidents," the applicant reported the results of its radiological consequence analysis for personnel in the main control room (MCR), relying on the CREFU to limit the radioactivity to which personnel may be exposed. Section 6.4, "Control Room Habitability Systems," of DCD Tier 2 describes the CREFU design and Section 6.4 of this report provides the staff's evaluation.

The original ESBWR design in DCD Tier 2, Revisions 0 through 2, included a passive control room emergency bottled air breathing system (EBAS) and did not provide an ESF atmosphere cleanup filtration system for the CR. In DCD Tier 2, Revision 3, the applicant changed its ESBWR design to provide the CREFU as an active containment ESF atmosphere cleanup filtration unit, designed to remove fission products from the CR habitability area and to pressurize the CR with nonradioactive air following postulated DBAs. The CREFU is a safety-related system and is a subsystem of the control building HVAC system located in the control building and is designed to seismic Category 1 criteria. The CREFU, in ESBWR DCD Tier 2, Revision 3, replaces the passive control room emergency air breathing system provided in the original ESBWR design in DCD Tier 2, Revisions 0 through 2.

The applicant assumed an unfiltered air in-leakage rate of 0.113 cubic meter per minute (DCD Tier 2, Table 15.4-5) in its CR radiological consequence analysis. In RAI 15.4-30, the staff requested that the applicant include preoperational testing of assumed CR unfiltered air in-leakage rates as an ITAAC item in Section 14.2.8.2.19, "HVAC System Performance Test," in Chapter 14, "Initial Test Program," and its surveillance requirements in Section 3.7.2, "Control Room Habitability Area Heating, Ventilation, and Air Conditioning Subsystem," in Chapter 16, in accordance with guidance provided in Technical Specification Task Force 448, "Control Room Habitability," dated July 1, 2003.

In its Revision 4 of DCD Tier1, the applicant has specified the CREFU tests in its Section 2.16.2.3 and Table 2.16.2-6 as ITAAC items. Therefore, RAI 15.4- 30 is resolved.

In Revision 3 to the DCD, the applicant did not provide complete figures and tables showing the design features that will be needed by the COL applicant to generate site-specific CR χ/Q

values at the COL stage. In RAI 2.3-9, the staff asked the applicant to provide figures showing CR intake, unfiltered in-leakage, and postulated DBA release locations to the environment. These figures are intended to provide a basis for determining the distances and directions between potential accident release pathways and intake and in-leakage pathways to the CR necessary to evaluate the radiological consequences. **RAI 2.3-9 is being tracked as an open item.**

In Revision 3 to the DCD, GEH revised the CR χ/Q values in DCD Tier 1, Table 5.1-1, and Tier 2, Table 2.0-1, listing them as standard plant site design parameters. Two sets of CR χ/Q values are provided for the RB, PCCS/RB roof, and TB release pathways; one set represents unfiltered in-leakage and the second set represents the filtered air intake. In RAI 15.4-31, the staff requested that the applicant state which set of CR χ/Q values it used for the CR radiological consequence analysis and why. **RAI 15.4-31 is being tracked as an open item.**

In DCD Tier 2, Sections 6.4 and 15.4, the applicant reported the results of its radiological consequence analysis for personnel in the MCR during a design-basis LOCA, relying on the CREFU to limit the radioactivity to which the personnel may be exposed. Due to the open items discussed above,, the staff cannot finalize its conclusions regarding whether the dose criteria of 0.25 Sv (25 rem) TEDE in 10 CFR 50.34 and 0.05 Sv (5 rem) TEDE in GDC 19 are met for the postulated LOCA.

15.4.3.3.2.5 Hypothetical Atmospheric Dispersion Factors

Because no specific site is associated with the ESBWR design, the applicant defined the offsite boundaries (EAB and LPZ) only in terms of various hypothetical atmospheric dispersion factors (χ/Q values). DCD Tier 1, Table 5.1-1, and DCD Tier 2, Table 2.0-1, list the hypothetical reference χ/Q values used in the radiological consequence analyses for the ESBWR design. Section 2.3.4 of this report provides the staff's discussion of the hypothetical atmospheric dispersion factors. The applicant stated that these χ/Q values were "back calculated" to determine the bounding values that will result in radiological consequence doses just under regulatory limits. The staff will independently confirm radiological consequence analyses by utilizing hypothetical reference χ/Q values.

The staff will perform an independent assessment of short-term (less than or equal to 30 days) atmospheric dispersion factors for potential accident consequence analyses on a site-specific basis for a COL application that references the ESBWR design. If site-specific atmospheric dispersion factors exceed the reference χ/Q values used in this evaluation (e.g., poorer dispersion characteristics), a COL applicant may have to consider compensatory measures, such as increasing the size of the site or providing additional ESF systems to meet the relevant dose limits set forth in 10 CFR 50.34 and GDC 19.

15.4.3.4 Conclusion

Due to the open items discussed above, the staff cannot finalize its conclusions.

15.4.4 Main Steamline Break Outside Containment

15.4.4.1 Regulatory Criteria

The requirements summarized in Section 15.1.1.3 are used in evaluating this DBA.

The staff reviewed DCD Tier 2, Section 15.4.5, "Main Steamline Break Accident Outside Containment," in accordance with SRP Section 15.0.3 and applicable guidance in RG 1.183, Appendix D.

The staff will evaluate the radiological consequences of this DBA against the dose acceptance criteria, specified in SRP Section 15.0.3 and RG 1.183, of 0.025 Sv (2.5 rem) TEDE for an accident-initiated iodine spike and 0.25 Sv (25 rem) TEDE for a pre-accident iodine spike at the EAB for any 2-hour period following the onset of the postulated fission product release. The staff will use a criterion of 0.05 Sv (5 rem) TEDE to evaluate the radiological consequences from a postulated MSLB accident in the control room of the ESBWR design, pursuant to GDC 19 of Appendix A to 10 CFR Part 50. In its review of this event, the staff used applicable guidance in RG 1.183 for the radiological consequence analyses. Although RG 1.183 was written to apply to currently operating power reactors, its guidance on radiological acceptance criteria, formulation of the source term, and DBA radiological consequence analysis modeling is applicable in the review of the ESBWR design because it is an ABWR.

15.4.4.2 Summary of Technical Information

The applicant postulated that one of the four main steamlines will rupture between the outer isolation valve and the turbine control valves (TCVs). The radiological consequences of a break outside containment will bound those from a break inside containment. The accident evaluated is the complete severance of a main steamline outside the containment at a location downstream of the outmost main steam isolation valve. The applicant presented its analyses of the radiological consequences of a postulated MSLB accident outside containment in DCD Tier 2, Section 15.4.5 and Tables 15.4-10 through 15.4-131. The MSIVs are assumed to isolate the break within 5 seconds, as specified in the ESBWR DCD Tier 2, Chapter 16, TS 3.6.1.3. No other release mitigation (i.e., plateout, holdup, dilution) is assumed. Following isolation of the main steam supply system, (i.e., MSIV closure), the ADS initiates. Once the reactor system has depressurized, the gravity drain cooling system automatically begins reflooding the reactor vessel and therefore, no fuel damage is projected to occur. All the radioactivity in the released coolant is assumed to be released to the environment instantaneously from the TB as a ground-level release.

In RAI 15.4-20, the staff requested that the applicant provide certain design parameters (i.e., pipe diameter, length, thickness, volume, steam pressure and temperature) for the main steamlines, drainlines, and condensers for the staff to perform an independent confirmatory radiological consequences through these pathways. **RAI 15.4-20 is being tracked as an open item.**

The applicant concluded in DCD Tier 2, Revision 3, that no more than 8.2328×10^4 kilograms (kg) of reactor coolant will be lost through the break before automatic isolation and less than 4.705×10^3 kg of that will be lost as steam. However, in its response to RAI 15.4-2, the applicant stated that GEH is preparing a formal MSLB outside of containment analysis to determine exact mass release values. In RAI 15.4-2, Supplement 1, the staff requested that the applicant provide, among other things, revised steam and water mass releases for the MSLB accident. The applicant has recently provided its response and the staff is currently reviewing. **RAI 15.4-**

2 is being tracked as an open item.

In DCD Tier 2, Revision 4, the applicant evaluated the dose to operators in the CR. The resulting 30-day TEDE to an individual in the control room is less than the GDC 19 dose criteria. The applicant assumed that the control room will not be isolated during this event, the CREFU will not be operational, and the normal CRAVS will be operational for this event. The applicant used an in-leakage rate of 200 L/s (424 cfm) of unfiltered air into the CR envelope for conservatism; this assumption allows for any unfiltered air in-leakage.

15.4.4.3 Staff Evaluation

The staff will perform an independent radiological consequence calculation for the following two scenario cases for the MSLB accident upon receipt of the applicant's radiological consequence analysis for this event in forthcoming Revision 5 to DCD Tier 2.

For Case 1, the staff will assume that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by the MSLB accident. Before the accident, the ESBWR reactor is assumed to operate the equilibrium limit of 7.4 kilobecquerels per gram (kBq/g) (0.2 microcuries per gram (µCi/g)) for dose equivalent iodine-131 (DEI-131) in the primary coolant as specified in the ESBWR TS. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate results in an increasing concentration in the primary coolant during the course of the accident.

For Case 2, the staff will assume that previous reactor operation had resulted in a primary coolant iodine concentration equal to the maximum instantaneous ESBWR TS limit of 0.148 MBq/g (4 µCi/g) for DEI-131.

RAI 15.4-24 noted that Section 4.4, "Main Steam Isolation Valve Leakage," of NEDE-33279P and Section 5.0, "Offsite Dose Calculations," stated that "fuel damage does not occur until…." This is contrary to the guidance provided in RG 1.183 for an AST. The staff requested that the applicant explain how it determined the fission product release timing for the radiological consequence analyses and how it used this value throughout NEDE-33279P and Section 15.4.4 of the proposed DCD Tier 2, Revision 3. **RAI 15.4-24 is being tracked as an open item.**

15.4.4.4 Conclusion

Due to the open items discussed above the staff cannot finalize its conclusions for this accident.

15.4.5 Failure of Small Lines Carrying Primary Coolant outside Containment

15.4.5.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Section 15.4.8, "Failure of Small Lines Carrying Primary Coolant Outside Containment," in accordance with NUREG-0800, SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside

Containment," Revision 2 and SRP Section 15.0.3. RG 1.183 does not provide guidance about or list this event as a DBA.

The staff will evaluate the radiological consequences of this DBA against the dose acceptance criterion, specified in SRP Section 15.0.3, of 0.025 Sv (2.5 rem) TEDE at the EAB for any 2-hour period following the onset of the postulated fission product release. The staff will use a criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences in the CR of the ESBWR design, pursuant to GDC 19 of Appendix A to 10 CFR Part 50.

GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," contains a provision to ensure isolation of all pipes that are part of the reactor containment pressure boundary and which penetrate the containment building. Exempted from these specifications are small-diameter pipes (instrument lines) that must be continuously connected to the primary coolant system to perform their necessary functions. For these lines, methods of mitigating the consequences of a rupture are necessary because the lines cannot be automatically isolated.

15.4.5.2 Summary of Technical Information

The requirements summarized in Section 15.1.1.3 are also used in evaluating this DBA.

For the ESBWR design, the applicant postulated an instantaneous and circumferential rupture of an instrument line that is connected to the primary coolant system outside the containment but inside the RB at a location where it may not be isolated automatically for 30 minutes at normal reactor operating temperature and pressure. Following the 30 minutes from the initiation of this event, the applicant assumed that the operator detects the event, and then scrams the reactor and initiates reactor depressurization.

The applicant estimated that 1.48×10^4 kg of primary coolant will be released through the break before it is isolated until the reactor is depressurized and that 4×10^3 kg of the primary coolant will flash to steam and be available for release. All of the iodine available in the flashed steam is assumed to be released via the RB to the environment without any mitigation. Furthermore, iodine in the primary coolant is assumed to be at the maximum equilibrium limit of 0.148 MBq/g (4 μCi/g) for DEI-131 specified in the ESBWR TSs.

The applicant evaluated the dose to operators in the CR. The applicant assumed that the CR will not be isolated during this event, the CREFU will not be operational, and the normal CRAVS will be operational for this event. The applicant used an in-leakage rate of 200 L/s (424 cfm) of unfiltered air into the CR envelope for conservatism; this assumption allows for any unfiltered air in-leakage. The applicant analyzed the CR dose over a 30-day period. The resulting 30-day TEDE to an individual in the CR is less than the GDC 19 dose criteria.

15.4.5.3 Staff Evaluation

While performing past licensing reviews, such as those for the AP600, AP1000, and ABWR, the staff determined that a small line break accident is expected to result in radiological consequences less than a small fraction of the dose reference values specified in 10 CFR 50.34(a)(1). Furthermore, the staff believes that this event may be subsumed by the MSLB and the reactor water cleanup line failure outside containment. However, upon receipt of a response to RAI 15.4-3, Supplement 1 from the applicant, the staff will perform an

independent evaluation of this event because this application is the first involving the ESBWR standard design with hypothetical site boundaries. In RAI 15.4-3, Supplement 1, the staff requested the applicant to provide the duration of this event, fission product release point, and site boundary and CR atmospheric dispersion values used. **RAI 15.4-3 is being tracked as an open item.**

15.4.5.4 Conclusion

Due to the open item discussed above, the staff cannot finalize its conclusions for this accident.

15.4.6 Failure of Reactor Water Cleanup System Line Outside Containment

15.4.6.1 Regulatory Criteria

This event is not listed as a DBA in either SRP Section 15.0.3 or RG 1.183 and is not required to be analyzed for its radiological consequences. However, during promulgation of Appendix A to 10 CFR Part 52, the ACRS Subcommittee on ABWR specifically recommended that the applicant analyze this event for the radiological consequences as a DBA for the ABWR. Therefore, the applicant analyzed, and the staff will review, this event for the radiological consequences for the ESBWR using the guidance provided in SRP Section 15.0.3 and RG 1.183 for the MSLB accident.

The staff will evaluate the radiological consequences of this DBA against the dose acceptance criteria, specified in SRP Section 15.0.3 and RG 1.183, of 0.025 Sv (2.5 rem) TEDE for an accident-initiated iodine spike and 0.25 Sv (25 rem) TEDE for a pre-accident iodine spike at the EAB for any 2-hour period following the onset of the postulated fission product release. The staff will also use a criterion of 0.05 Sv (5 rem) TEDE to evaluate the radiological consequences in the CR of the ESBWR design, pursuant to GDC 19 of Appendix A to 10 CFR Part 50.

15.4.6.2 Summary of Technical Information

The requirements summarized in Section 15.1.1.3 are used in evaluating this DBA.0.

In DCD Tier 2, Section 15.4.9, the applicant presented its analyses of the radiological consequences of a postulated Reactor water Cleanup/ Shutdown Cooling (RWCU/SDC) system line failure outside containment.

The applicant assumed that the break will be instantaneous and circumferential, and will occur on the downstream side of the outermost containment isolation valve, but on the upstream side of the RWCU demineralizer. The applicant assumed 66 seconds of break flow time (46-second built-in delay time for flow differential pressure instrumentation to activate an isolation signal and 20 seconds for the motor-operated isolation valve to close). The applicant further assumed that no fuel damage will result as a consequence of this event. The applicant determined that the initial break flow rate will be limited to 2.218×10^3 kg per second assuming (1) two-phase critical flow for limiting diameter piping inside containment and (2) no more than 1.23×10^5 kg of reactor coolant will be lost through the break before automatic isolation and less than 4.9×10^4 kg of that will be lost as steam.

15.4.6.3 Staff Evaluation

The staff provided no specific regulatory guidance for evaluating the radiological consequences for this event in RG 1.183 or SRP Section 15.0.3. Therefore, upon receipt of response to RAI 15.4-4, Supplement 1 from the applicant, the staff will review this event using the guidance provided in SRP Section 15.0.3 and RG 1.183 for the MSLB accident.

The staff will perform an independent radiological consequence calculation for the following two scenario cases for this event. For Case 1, the staff will assume that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by this event. Before the accident, the ESBWR reactor is assumed to operate the equilibrium limit of 7.4 kBq/g (0.2 μCi/g) for DEI-131 in the primary coolant as specified in the ESBWR TS. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate results in an increasing concentration in the primary coolant during the course of the accident. For Case 2, the staff will assume that previous reactor operation had resulted in a primary coolant iodine concentration equal to the maximum instantaneous ESBWR TS limit of 0.148 MBq/g (4 μCi/g) for DEI-131.

The applicant evaluated the dose to operators in the CR. The applicant assumed that the CR will not be isolated during the postulated MSLB accident, the CREFU will not be operational, and the normal CRAVS will be operational for this event. The applicant used an in-leakage rate of 200 l/s (424 cfm) of unfiltered air into the CR envelope for conservatism; this assumption allows for any unfiltered air in-leakage. The applicant analyzed the CR dose over a 30-day period. The resulting 30-day TEDE to an individual in the CR is less than the GDC 19 dose criteria. The staff will perform an independent confirmatory dose calculation upon receipt of the applicant's response to RAI 15.4-4. In RAI 15.4-4, Supplement 1, the staff requested the applicant to provide the duration of this event, fission product release point, and site boundary and CR atmospheric dispersion values used. **RAI 15.4-4 is being tracked as an open item.**

15.4.6.4 Conclusion

Due to the open item discussed above, the staff cannot finalize its conclusions for this event.

15.5 Special Events

Historically, non-DBEs that are evaluated in BWR safety analysis reports or DCDs have been termed, "special events." The applicant retained this classification for the ESBWR. The applicant established the following criteria for special events:

- postulated in 10 CFR Part 50 to demonstrate some specified prevention, coping, or mitigation capabilities, without specifically requiring a radiological evaluation,

- include a common-mode equipment failure or additional failure(s) beyond the single-failure criterion.

The applicant provided, in DCD Tier 2 Revision 3, Chapter 15, analyses of special events. In some cases, these events form the technical bases for conclusions drawn in other sections of

this report. In such instances, the applicant presented results in the corresponding section of the DCD. The staff will correspondingly reference the appropriate SER section.

15.5.1 Overpressure Evaluation

Section 5.2.2 of this report presents the results of the staff's overpressure evaluation.

15.5.2 Shutdown without Control Rods

The standby liquid control system (SLCS), which is evaluated in Section 9.3.5 of this report, provides for reactor shutdown without control rods.

15.5.3 Shutdown from outside the Main Control Room

Section 7.4.2 of this report evaluates shutdown from outside the MCR.

15.5.4 Anticipated Transient Without Scram (ATWS)

An ATWS is an AOO, as defined in Appendix A to 10 CFR Part 50, followed by the failure of the reactor trip portion of the protection system specified in GDC 20. Since protection systems (e.g., the reactor trip system) must satisfy the single-failure criterion, the assumed failure of the reactor trip must be caused by multiple failures or a common-mode failure. The probability of an AOO coincident with multiple failures or a common-mode failure is much lower than the probability of any of the other events that are evaluated in Chapter 15. Therefore, an ATWS event cannot be classified as either an AOO or a DBA.

The failure of the reactor to shut down during certain transients can lead to unacceptable RCS pressures, fuel conditions, and/or containment conditions. For a conventional BWR, AOOs with failure to scram could lead to unacceptable conditions, such as closure of MSIVs or turbine trip with bypass available, if unmitigated unstable power oscillations are allowed to grow.

Safety issues associated with ATWS have been evaluated since the early 1970s. During NRC evaluations of vendor models and analyses addressing such events, ATWS was formally identified as Unresolved Safety Issue (USI) A-9, "Anticipated Transients Without Scram." NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," issued in 1980 presents the NRC staff studies and findings regarding Unresolved Safety Issue (USI) A-9. In 1986, USI A-9 was resolved through promulgation of 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," also known as the ATWS rule. The ATWS rule does not require ATWS analyses. SECY 83-293, "Amendments to 10 CFR Part 50 Related to Anticipated Transients Without Scram Events," dated July 19, 1983, and the *Federal Register* notice of the final rule (49 FR 26036) present the bases for current regulatory requirements related to ATWS, including the associated regulatory evaluation.

15.5.4.1 Acceptance Criteria

Prescriptive requirements for ATWS are specified in 10 CFR 50.62. This regulation requires BWRs to have the following mitigating features for an ATWS event:

- a SLCS capable of injecting a borated water solution with reactivity control equivalent to the control obtained by injecting 86 gallons per minute (gpm) of a 13 percent by weight sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a reactor vessel with a 251-inch inside diameter,

- an alternate rod insertion (ARI) system that is designed to perform its function in a reliable manner and that is independent from sensor output to the final actuation device,

- a SLCS initiation that is automatic and designed to perform its function in a reliable manner for plants granted a CP after July 26, 1984,

- equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

The staff determined that this latter requirement does not apply to the ESBWR because the ESBWR does not contain recirculation pumps. Therefore, the staff reviewed the ESBWR DCD to determine that comparable actions were provided.

BWR performance during an ATWS was also compared to the criteria used in the development of the ATWS safety analyses described in NEDO-24222, “Assessment of BWR Mitigation of Anticipated Transients Without Scram,” issued December 1979. The criteria include the following:

- limiting the peak vessel bottom pressure to less than the ASME Service Level C limit of 1500 psig,

- ensuring that the peak cladding temperature, maximum cladding oxidation, and coolable geometry remain within the limits specified in 10 CFR 50.46,

- limiting peak suppression pool temperature to less than the containment design temperature,

- limiting the peak containment pressure to a maximum of the containment design pressure.

Finally, SRP Chapter 15.8 provided guidance for the staff’s review of BWR ATWS. SRP Chapter 15.8 provides the applicable GDC that form the regulatory basis of the ATWS rule, as listed below:

- GDC 12, “Suppression of Power Oscillations,” which requires that oscillations are either not possible or can be reliably and readily detected and suppressed,

- GDC 13, “Instrumentation and Control,” requiring a control and monitoring system to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions,

- GDC 14, “Reactor Coolant Pressure Boundary,” which requires an extremely low probability of failure of the coolant pressure boundary,

- GDC 16, which requires that containment design conditions important to safety are not exceeded as a result of postulated accidents,

- GDC 35, “Emergency Core Cooling,” which specifies that fuel and clad damage, should it occur, must not interfere with continued effective core cooling and that clad metal-water reactions must be limited to negligible amounts,

- GDC 38, “Containment Heat Removal,” which requires that the containment pressure and temperature be maintained at acceptable low levels following any accident that deposits reactor coolant in the containment,

- GDC 50, “Containment Design Basis,” which requires that the containment not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment.

The ATWS rule specifies two requirements: (1) light-water cooled plants must have prescribed systems and equipment that have been determined to acceptably reduce risks attributable to ATWS events and (2) licensees must demonstrate the adequacy of the features specified in the rule. In addition, all required equipment and systems must be designed to perform their functions in a reliable manner. Design and quality assurance criteria for the required systems and equipment should meet or exceed the criteria established in conjunction with the ATWS rulemaking, as described in Appendix A to SRP Section 7.1A, December 4, 1997, to ensure adequate independence, diversity, and reliability as required by the ATWS rule.

15.5.4.2 Summary of Technical Information

For ATWS prevention/mitigation, the ESBWR provides the following:

- 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 FW runback that operates under conditions indicative of an ATWS,

- An SLCS that automatically initiates under conditions indicative of an ATWS.

The mitigation of ATWS events is accomplished by a multitude of equipment and procedures. These include ARI, FMCRD run-in, FW runback, ADS inhibit, and SLCS. The following are the initiation signals and setpoints for the above responses:

- ARI and FMCRD run-in

    - High pressure, or
    - Level 2, or

    - Either RPS scram command, or SCRRI/SRI command and elevated power levels exist after time delay
    - Manual

- SLCS initiation

    - High pressure and Startup Range Neutron Monitor (SRNM) ATWS permissive for 3 minutes, or
    - Level 2 and SRNM ATWS permissive 3 minutes, or
    - Manual ARI/FMCRD run-in signals and SRNM ATWS permissive for 3 minutes

- Feedwater runback

    - High pressure and SRNM ATWS permissive, or
    - Either RPS scram command, or SCRRI/SRI command and elevated power levels persist after time delay
    - Manual ARI/FMCRD run-in
    - ADS inhibit
    - High pressure and Average Power Range Monitoring (APRM) not downscale for one minute, or
    - Level 2 and APRM not downscale

- HP CRD

    - Level 2 with maximum 10 s delay
    - Level 2 with maximum delay of 145 s during loss of off-site power
    - IC
    - Closure of MSIV
    - High pressure for 10 s
    - Level 2 with 30 s delay or Level 1

15.5.4.3 Staff Evaluation

The ESBWR has an SLCS capable of automatically injecting 291 gpm of sodium pentaborate solution into the RPV with the simultaneous operation of both accumulators. The 86-gpm equivalency specified in the ATWS rule (10 CFR 50.62) for the 251-inch RPV (sodium pentaborate decahydrate solution of 13 percent by weight at 86 gpm for a 251-inch vessel) is satisfied by the 291 gpm provided for the 278-inch EBSWR vessel. The staff evaluated compliance with this portion of 10 CFR 50.62, as described in Section 9.3.5 of this report, and concluded that these requirements were satisfied.

The ATWS rule requires that the SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984. Section 9.3.5 of this report provides a detailed evaluation of the SLCS.

GDC 26–28 require the SLCS, which is described in Section 9.3.5. Because the new CRD design eliminates the previous common-mode failure potential and because of the very low probability of simultaneous common-mode failures of a large number of FMCRDs, a failure to achieve shutdown is deemed unlikely. The staff position in this regard is that the provisions of

10 CFR 50.62 continue to require the SLCS. In addition, the ESBWR incorporates automatic initiation of the SLCS under conditions indicative of an ATWS in order to meet the rule specified in 10 CFR 50.62.

The ESBWR incorporates electric-hydraulic FMCRDs, which provide motor-driven scram and hydraulic scram. In response to a scram signal, the control rods are inserted hydraulically by means of the stored energy in the scram accumulator, similar to the currently operating BWR CRDs. In the ESBWR, a scram signal is also given simultaneously to insert the FMCRD electrically by means of the FMCRD motor drive. This diversity (i.e., hydraulic and electric methods of scramming) provides a high degree of assurance of rod insertion on demand.

The ESBWR has an ARI system that is independent of the RPS from sensor output to the final actuation device. The ARI system has redundant scram air header exhaust valves. The ARI system is designed to perform its function in a reliable manner and is independent of the existing RPS system from sensor output to the final actuation device. Chapter 7 of this report provides a detailed evaluation of the ARI and RPS.

As stated in the evaluation criteria, the ATWS rule incorporates prescriptive requirements because it clearly reflected the BWR use of forced core flow circulation. Because the ESBWR uses natural circulation, there are no recirculation pumps to be tripped. Hence, no recirculation pump trip (RPT) logic can be implemented in the ESBWR. An ATWS automatic FW runback feature is implemented, which provides a reduction in water level, core flow, and reactor power, similar to the RPT in a forced circulation plant. This feature prevents reactor vessel overpressure and possible short-term fuel damage for the most limiting ATWS events. The staff finds that the FW runback feature is comparable to the RPT feature provided in BWRs with forced recirculation with respect to the requirements of 10 CFR 50.62(c)(5).

The ATWS rule is also specific as to the use of locking-piston CRDs. The ESBWR, however, uses the FMCRD design with both hydraulic and electrical means to achieve shutdown. Section 4.6 of this report describes this control rod drive system. The use of this design reduces the common-mode failure potentials of the existing locking-piston CRD by eliminating the scram discharge volume (potential mechanical common-mode failure) and by having an electric motor run-in diverse from the hydraulic scram feature. This latter feature allows rod run-in, if scram air header pressure is not exhausted because of a postulated common-mode electrical failure and simultaneous failure of the ARI system, and thus satisfies the intent of 10 CFR 50.62.

The staff issued RAI 15.5-5 regarding the manner in which operator actions are credited by the applicant. The staff sought clarification because the TRACG analysis of ATWS does not appear to include operator actions; discussion in the DCD suggests otherwise. The applicant provided that the TRACG analysis of ATWS MSIV Closure transient response evaluation assumes operator action to:

1) Maintain level at TAF + 5ft (1.524 m) after the initial automatic feedwater runback.

2) Depressurize the reactor, if the Heat Capacity Temperature Limit (HCTL) curve is reached.

Staff agrees with this response. RAI 15.5-5 is resolved.

The applicant analyzed several classes of transients to provide assurance that, based on a low estimated frequency of occurrence, unacceptable plant conditions will not occur in the event of an ATWS. The applicant demonstrated that RCS pressures will not exceed ASME Service Level C limits—120 percent of the RPV design peak pressure of 1500 pounds per square inch. This analysis was performed using the systems code TRACG. The NRC staff is currently reviewing the applicability of TRACG for the ESBWR ATWS analysis as presented in NEDE 33083-P, Supplement 2, "TRACG Application for ESBWR Anticipated Transients Without Scram Analyses." The staff's evaluation is provided in Section 21.6 of this report, and in the Staff's SER with open items regarding NEDE-33083P, Supplement 2. Approval of the analysis and results presented in DCD Tier 2, Section 15.5.4 are dependent on closure of all open items associated with the staff's review of NEDE-33083P, Supplement 1.

The applicant also used TRACG to analyze ESBWR stability during ATWS scenarios. Chapter 4 of this report addresses ESBWR stability during ATWS scenarios.

15.5.4.4 Analysis

To establish compliance with the criteria identified in Section 15.5.1, the applicant performed analyses of ATWS scenarios initiated by the following AOOs:

- MSIV Closure—The maximum values from this event are, in most cases, bounding of all events considered;
- Loss of Condenser Vacuum—Pressurization rate and energy addition to the pool may be as severe as those in the MSIV closure scenario;
- Loss of FW Heating—This scenario may be limiting in terms of peak cladding temperature;
- Loss of Normal AC Power to Station Auxiliaries—This scenario could challenge the capability of the plant to mitigate an ATWS event because of reduced available equipment;
- Loss of FW Flow—This event is analyzed to demonstrate the plant's capability to mitigate ATWS events initiated by low level trips;
- Generator Load Rejection with Single Failure in the Turbine Bypass System.

The applicant analyzed these events using the computational code TRACG. As discussed above, the staff is currently reviewing the application of TRACG to analyze ATWS scenarios for ESBWR.

As discussed in Section 21.6 of this report, there are several open RAIs regarding concerns about the capability of TRACG to analyze boron-mixing phenomena in the ESBWR core bypass. In addition, the staff is performing a confirmatory computational fluid dynamics calculation using the commercial code FLUENT. These confirmatory calculations are discussed in Section 21.5 of this report. The staff's Safety Evaluation with Open Items for Application of the TRACG Computer Code to Anticipated Transients Without Scram for the ESBWR Design NEDE-33083P, Supplement 2," December 2007 (ML073190036)."

The staff is also performing a confirmatory neutronics analysis to determine the effects of localized boron concentration on the effective multiplication factor of the ESBWR core. For this

analysis, the staff will be using the Monte Carlo N-Particle Transport Code and a flux-squared, adjoined weighting factor to determine core criticality in situations with limited boron distribution. These calculations are also discussed in Section 21.6 of this report.

15.5.4.5 Conclusion

The staff's findings are subject to closure of the open items, in Section 21.6 of this report, and in the Staff's SER with open items regarding NEDE-33083, Supplement 2. Pending closure of these open items, the staff concludes that the plant design adequately addresses ATWS events and meets the requirements of 10 CFR 50.62. This conclusion is based on the following:

- The applicant's plant design includes ATWS risk reduction features prescribed by the ATWS rule.
- These features are independent and diverse from the reactor trip system and are designed to be reliable as required under the ATWS rule.
- The applicant has also proven or referenced information, analyses, and/or risk assessments that demonstrate that it has considered limiting ATWS transient and event sequences and determined that features included in the design pursuant to the ATWS rule result in reasonable assurance, based upon a low estimated frequency of occurrence, that unacceptable plant conditions, as defined during the ATWS rulemaking, will not occur as a result of ATWS events.
- The applicant has provided an acceptable diverse scram system.

The staff analyzed ESBWR ATWS assuming the reactor is deployed with the core as described in DCD Tier 2, Chapter 4. Any changes to fuel design or core design will require confirmation of the ATWS analysis.

15.5.5 Station Blackout

As required by 10 CFR 50.63, each light-water-cooled nuclear power plant must be able to withstand and recover from an station blackout (SBO) (i.e., loss of the offsite electric power system concurrent with reactor trip and unavailability of the onsite emergency ac electric power system) of a specified duration. In particular, 10 CFR 50.63 requires that, for the SBO duration, the plant be capable of maintaining core cooling and appropriate containment integrity. It also identifies the factors that must be considered in specifying the SBO duration.

15.5.5.1 Acceptance Criteria

As required by 10 CFR 50.63(a)(2):

> "[T]he reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of an SBO for the specified duration. The capability for coping with an SBO of specified duration shall be determined by an appropriate coping

analysis."

As noted in RAI 15.5-8, Regulatory Position 3.2.7 of RG 1.155, "Station Blackout," states that the ability to maintain appropriate containment integrity during a loss of all ac power should be addressed. The applicant addressed containment integrity in terms of design limits on pressures and temperatures. The staff requested that the applicant add a discussion to Section 15.5 of the ESBWR DCD explaining what provisions are present to ensure valve position indication and closure for containment isolation valves that may be in the open position at the onset of an SBO. **RAI 15.5-8 is being tracked as an open item.**

The staff performed its systems review of SBO using Position 3 of RG 1.155. This guidance has been incorporated into SRP Section 8.4, "Station Blackout," March 2007. The guidance is as follows:

- 3.2.1: Analysis performed should assume 100-percent rated thermal power for 100 days;
- 3.2.2: Core cooling and decay heat removal capability should be determined;
- 3.2.3: Adequate inventory should be ensured;
- 3.2.4 Design adequacy and capability should be evaluated, including potential failures of equipment necessary to cope;
- 3.2.5: Use of non-safety-related equipment should be considered;
- 3.2.6: Timely operator actions should be considered; and
- 3.2.7: The ability to maintain appropriate containment integrity should be addressed.

15.5.5.2 Summary of Technical Information

For its SBO analysis, the applicant used the following assumptions and inputs:

- The reactor is operating at 100 percent rated power and 100 percent rated nominal core flow, with nominal dome pressure and normal water level,
- The nominal American National Standards Institute/American Nuclear Society Standard 5.1-1994 decay heat model is assumed,
- SBO starts with loss of all ac power, which occurs at time zero. Auto bus transfer is assumed to fail,
- The loss of ac power trips the reactor, FW, condensate, and circulating water pumps. A turbine load rejection is also initiated,
- The reactor scram occurs at 2 seconds from the loss of power supply to the FW pumps because loss of FW flow results in a scram signal with a delay time of 2 seconds,

- Bypass valves open on load rejection signal and close 6 seconds later because of a loss of condenser vacuum and/or to control the reactor pressure when it begins to drop because of the reactor scram,

- The MSIVs close automatically 30 seconds after the water level reaches Level 2 or because of a loss of condenser vacuum; the valves are fully closed at 5 seconds,

- The CRD pumps are unavailable due to loss of ac power. No safety systems are credited, with the exception of three ICs,

- ICs are automatically initiated upon loss of FW pump power buses at 3 seconds to remove decay heat following the scram and isolation. IC drain flow provides initial reactor coolant inventory makeup to the RPV,

- The analysis credits no automatic or manual action when the vessel reaches Level 2 or 3,

- Vessel depressurization occurs; inventory of vessel and other components remains constant. Changes in level are observed as a result of changes in liquid temperature and pressure.

Using these assumptions and initial conditions, the applicant analyzed the SBO scenario using the TRACG computational code to conclude that, during a 72 hour coping period that credits no operator actions, the ESBWR is placed and maintained in a hot-shutdown condition. The coolant inventory is such that it remains above Level 1 in the vessel. As a result of ICS operation, coolant is not released into the drywell or wetwell. Therefore, the applicant asserts that containment integrity is maintained.

15.5.5.3 Staff Evaluation

The applicant used TRACG to analyze the SBO scenario. The staff had not previously determined that TRACG is qualified for this analysis. To establish qualification, the staff reviewed the existing TRACG qualification documentation that applies specifically to the ESBWR (i.e., NEDE-33083-P-A, “TRACG Application for ESBWR”). The staff determined that the conditions predicted during the SBO scenario are within the limits of a LOCA and ATWS scenarios for which TRACG approval is pending.

The TRACG qualification documentation states that TRACG is qualified to predict ESBWR system responses. The documentation also provides validation of the ability of TRACG to model IC behavior by comparison to test data from the PANDA facility. TRACG was verified in this respect by a comparison to staff confirmatory TRACE calculations. In the staff’s SER approving NEDE-33083P-A, the staff concluded that TRACG adequately modeled IC behavior.

In consideration of the nonlimiting nature of the reactor and system response during the SBO scenario, as well as the stated capability of TRACG to model IC performance, the staff concludes that the TRACG analysis adequately predicts ESBWR performance during an SBO.

The staff issued RAI 15.5-6 to verify that the SBO analysis assumed operation at 100-percent thermal power for 100 days. In response, GEH committed to provide this information in a revision to the DCD. DCD Tier 2, Revision 3, has been updated with this change. Staff has

reviewed and accepted the change. RAI 15.5-6 is resolved.

The selection of a coping time must be based on site-specific criteria, as required by 10 CFR 50.63. However, because passive plants will not have emergency AC power sources, applicants for such plants need not evaluate SBO coping duration as long as they are able to demonstrate that the design selected is capable of performing safety-related functions for 72 hours. The ESBWR is capable of maintaining the core in a hot-shutdown condition for at least 72 hours using three of the four ICs.

The TRACG analysis was carried out for 20,000 seconds to demonstrate that the IC system is capable of maintaining collapsed water level above the TAF and that a hot-shutdown condition can be achieved and maintained. The staff reviewed the applicant's analysis and determined that it showed that core cooling, decay heat removal capability, and coolant inventory are adequate. The SBO analysis indicates that appropriate containment integrity is maintained through the duration of the event.

Because an IC is assumed to be out of service, the staff concludes that potential failures of equipment necessary to cope have been considered. The use of non-safety related equipment is not assumed, and no operator actions are required.

15.5.5.4 Conclusion

Due to the open items discussed above, the staff cannot finalize its conclusions.

15.5.6 Safe-Shutdown Fire

The applicant credits TRACG analysis of SBO to provide conservative results for the safe-shutdown fire scenario because a manual scram is initiated before evacuation of the MCR. The staff reviewed the set of initial conditions for both scenarios and determined that, because all four ICs are assumed to be available in the fire scenario as well as CRD flow, the SBO scenario is bounding for the fire scenario during a CR fire. From a reactor systems standpoint, the SBO review demonstrates system response adequacy as applied to a safe-shutdown fire. The staff's evaluation of safe-shutdown fire from a fire protection perspective is provided in Section 9.5.1 of this report.

15.5.7 Waste Gas System Leak or Failure

Section 11.3.7 of this report evaluates waste gas system leak or failure.

15A Event Frequency Determination

15A.1 Summary of Technical Information:

The staff of the U.S. Nuclear Regulatory Commission reviewed the methodology used in the determination of the event frequency. The applicant, GEH, stated that it used the following types of analysis in determining the event frequency:

- For those initiating events explicitly modeled in the ESBWR probability risk assessment

(PRA), the frequency of the initiating events is taken directly from the PRA. The staff found only one discrepancy of this type. For the stuck-open relief valve event, Section 15A used a modified number lower than that used in the ESBWR PRA. The staff issued n RAI 15.0-18 (see discussion in Section 15.A.2.10 to ask GEH for clarification,

- The event frequency is determined from actual boiling-water reactor (BWR) operating experience, modified to reflect the ESBWR improved design features. Any cases involving COL application confirmation are identified. The staff verified that, to meet the infrequent event criterion (event frequency less than 0.01/yr), GEH identified that the reliability of the FW and pressure controllers are to be confirmed by the vendor,

- For events involving multiple independent hardware failures or human errors, the event frequency is based on conservative estimates of the hardware failures (including common-cause failures) and human errors. The staff verified that this approach was used for the events of turbine trip with total bypass failure, generator load rejection with total bypass failure, loss of FW heating with failure of SCRRI, and inadvertent shutdown cooling function operation.

15.A.2 Staff Evaluation

The staff compared event frequencies and failure probabilities used by the applicant in the analyses with data from operating reactors published in NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” issued February 2007, and the latest update to NUREG/CR-5750. The staff found that the parameter values used are consistent with operating experience and in many cases are reflective of the 95th percentile of the distribution of the operational data. In cases in which a comparison could not be made, the staff examined the impact of increasing the parameter by an order of magnitude to see whether such increases produced results that exceeded the staff’s acceptance criteria.

To account for modeling uncertainty, the staff verified that the final event frequencies are at least a factor of three above the criterion for infrequent events (i.e., less that 3.33X10-3 per year) A discussion of the staff’s evaluation of the frequency of each specific event follows.

15A.2.1 Pressure Regulator Failure—Opening of All Turbine Control and Bypass Valves

The steam bypass and pressure control (SB&PC) system controls the reactor pressure during plant operation. The SB&PC system is equipped with a triple-redundant, fault-tolerant digital controller (FTDC). The reliability of the FTDC will be confirmed by the vendor as part of the COL applicant commitment. Upon the vendor confirmation, the reliability of the SB&PC controller will meet the requirement that the mean time to failure (MTTF) be higher than 1000 years. The controller can either fail high causing maximum demand or fail low causing minimum demand. Assuming that both failure modes are equally possible, the frequency of the controller’s failing in a manner to cause maximum demand is estimated to be once in 2000 years ($5x10^{-4}$/yr).

In response to RAI 15.0-25, item A, GEH made an assessment of the mechanical failure of the pressure regulators and concluded that the likelihood of mechanical failure of the pressure regulators is negligible compared to the estimated overall failure frequency of $5x10^{-4}$/yr. The

staff agrees with this assessment. Therefore, RAI 15.0-25 is resolved.

Pending the successful confirmation of the reliability of the SB&PC controller, estimated fail frequency of the pressure controller is $5x10^{-4}$/yr. The staff agrees that this event frequency meets the criterion of being less than $1x10^{-2}$/yr.

15A.2.2 Pressure Regulator Failure—Closure of All Turbine Control and Bypass Valves

The SB&PC system controls the reactor pressure during plant operation. The SB&PC system is equipped with a triple-redundant FTDC. The reliability of the FTDC will be confirmed by the vendor as part of the COL applicant commitment. Upon the vendor confirmation, the reliability of the SB&PC controller will meet the requirement that the MTTF be higher than 1000 years. The controller can either fail high causing maximum demand or fail low causing minimum demand. Assuming that both failure modes are equally possible, the frequency of the controller's failing in a manner to cause minimum demand is estimated to be once in 2000 years ($5x10^{-4}$/yr).

In response to RAI 15.0-25, item A, GEH made an assessment of the mechanical failure of the pressure regulators and concluded that the likelihood of mechanical failure of the pressure regulators is negligible compared to the estimated overall failure frequency of $5x10^{-4}$/yr. The staff agrees with this assessment. Therefore, RAI 15.0-25 is resolved.

Pending the successful confirmation of the reliability of the SB&PC controller, estimated fail frequency of the pressure controller is $5x10^{-4}$/yr. The staff agrees that this event frequency meets the criterion of being less than $1x10^{-2}$/yr.

15A.2.3 Turbine Trip with Total Bypass Failure

In response to RAI 15.0-20, items (A)–(F), GEH has modified the model of the turbine bypass failure using the linked fault tree approach. The modeling of the bypass valves failures include the electro-hydraulic control system (EHC), related mechanical components, and supporting power supplies. Therefore, RAI 15.0-20 is resolved.

Based on the modified turbine bypass failure model and industry data for the frequency of turbine trip, the frequency of turbine trip with total turbine bypass failure is $5.7x10^{-4}$/yr. The staff agrees that this event frequency meets the criterion of being less than $1x10^{-2}$/yr.

15A.2.4 Generator Load Rejection with Total Turbine Bypass Failure

In response to RAI 15.0-20, items (A)–(F), GEH has modified the model of the turbine bypass failure using the linked fault tree approach. The modeling of the bypass valves failures included the EHC system, related mechanical components, and supporting power supplies. Staff agrees with this response.

In response to RAI 15.0-20, item (G), GEH proposed three alternatives to estimate the generator load rejection initiating event frequency. The staff agrees with the approach of using traditional generator load rejection frequency data as the initiating event frequency for generator load rejection. Therefore, RAI 15.0-20 is resolved.

Based on the modified turbine bypass failure model and traditional approach for estimating the initiating frequency for generator load rejection, the frequency of generator load rejection with total turbine bypass failure is 1.98×10^{-4}/yr. The staff agrees that this event frequency meets the criterion of being less than 1×10^{-2}/yr.

15A.2.5 Feedwater Controller Failure—Maximum Demand

The feedwater control system (FWCS) is equipped with a triple-redundant FTDC. The reliability of the FTDC will be confirmed by the vendor as part of the COL applicant commitment as found in DCD Tier 2, Revision 3, Section 15A.4. Upon the vendor confirmation, the reliability of the FWCS controller will meet the requirement that the MTTF be greater than 1000 years. If any one of the three controllers fails either high causing maximum demand or fails low causing minimum demand, the other two controllers will continue to function, and the frequency of two or three controllers failing in a manner to cause maximum demand is once in 2000 years ($5X10^{-4}$/yr).

Pending the successful confirmation of the reliability of the FWCS controller, estimated fail frequency of the feedwater controller is 5×10^{-4}/yr. The staff agrees that this event frequency meets the criterion of being less than 1×10^{-2}/yr.

15A.2.6 Loss of Feedwater Heating with Failure of Selected Control Rod Run-In

Based on the responses to RAIs 15.0-21 and 15.0-25, item (B.1), GEH has modified the initiating event frequency estimate by adding the selected rod insertion system to back up the SCRRI. The failure frequency calculation for this initiating event reflects the electrical, mechanical, and common-cause failure modes.

Based on the RAI responses and detailed modeling of the failure modes, the estimated failure frequency is 1.488×10^{-3}/yr. The staff agrees that this event frequency meets the criterion of being less than 1×10^{-2}/yr. Therefore, RAIs 15.0-21 and 15.0-25 are resolved.

15A.2.7 Inadvertent Shutdown Cooling Function Operation

In response to RAI 15.0-22, GEH used a linked fault tree approach to estimate the frequency of inadvertent shutdown cooling actuation. Valve functions, testing, and operator errors were modeled. Based on this approach, GEH estimates that the frequency of inadvertent SDC mode of operation is about 1.6×10^{-4}/yr.

Based on the RAI responses and improved modeling, the estimated event frequency of this event is 1.6×10^{-4}/yr. The staff agrees that this event frequency meets the criterion of being less than 1×10^{-2}/yr. Therefore, RAI 15.0-22 is resolved.

15A.2.8 Inadvertent Opening of a Safety/Relief Valve

In response to RAI 15.0-23, GEH included detailed failure modes of inadvertent opening of a safety/relief valve leading to vessel depressurization. Modeled failure modes are incorrect setpoints, vibration-induced failure, excess nitrogen pressure, spurious opening signal, operator error, and common-cause failures.

Based on the RAI responses and detailed modeling, the estimated event frequency of this event is 2.8×10^{-3}/yr. The staff agrees that this event frequency meets the criterion of being less than 1×10^{-2}/yr. Therefore, RAI 15.0-23 is resolved.

15A.2.9 Inadvertent Opening of a Depressurization Valve

In response to RAI 15.0-24, GEH has modified the modeling of this event by using the linked fault tree approach and including the common-cause failures.

Based on the RAI responses and improved modeling, the estimated event frequency of this event is 5.75×10^{-4}/yr. The staff agrees that this event frequency meets the criterion of being less than 1×10^{-2}/yr. Therefore, RAI 15.0-24 is resolved.

15A.2.10 Stuck-Open Relief Valve

In RAI 15.0-28, Section 15A.3.10, "Stuck Open Relief Valve," GEH estimated this initiating event frequency by taking credit for the availability of the isolation condenser (IC) system for the ESBWR. It is assumed that the probability of the IC being unavailable is less than 0.1. There is no justification for this number in this section. The staff has asked the applicant to provide the technical basis for this number.

The applicant's response to RAI 15.0-28 provided applicable information on the unavailability of the IC system and that the value (0.1) assumed in the analysis is conservative. Staff agrees with this response. Therefore, RAI 15.0-28 is resolved.

In RAI 15.0-29, Section 15A.3.10, GEH provided a best estimate value for the expected frequency of a stuck-open safety/relief valve (SRV) in an ESBWR of 3.28×10^{-4}/yr. However, the traditional number used for existing BWR plants is about 4.6×10^{-2}/yr (NUREG/CR-5750). In addition, the number used in the ESBWR PRA is 2.23×10^{-2}/yr (see NEDO-33201, Revision 2, Section 2). The staff have asked the applicant to explain why the best estimate of ESBWR frequency (i.e., 3.28×10^{-4}/yr) was not used in the ESBWR PRA.

The applicant addressed RAI 15.0-29 by providing the initiating event frequency of a stuck-open relief valve and explaining that the value used in the ESBWR is lower than that of the existing BWRs by crediting the IC system and eliminating the surveillance testing requirement of SRVs during power operation for the ESBWR. Staff agrees with this response. RAI 15.0-29 is resolved.

In response to RAI 15.0-18 (See RAIs 15.0-28 and 15.0-29 above.) GEH removed the four referenced analysis assumptions to be confirmed by the COL applicant from Appendix 15A to DCD Tier 2. These analysis assumptions have been relocated to other DCD sections and will be confirmed during the design certification process. **RAI 15.0-18 is being tracked as an open item.**

15A.2.11 Control Rod Withdrawal Error During Refueling

This event is initiated by one or more operator errors followed by failure of the refueling equipment interlocks. According to the GEH estimate, the frequency of a rod withdrawal error

during refueling is evaluated to be significantly less than once in 1000 years based on the multiple failures required for this event to occur.

In response to RAI 15.0-25, item B.2, GEH made an assessment of the mechanical failure of the FMCRD and concluded that the likelihood of mechanical failure of the FMCRD is negligible compared to the estimated overall fail frequency. The staff agrees with this assessment.

Based on the RAI responses and estimated failure frequency of $1x10^{-3}$/yr, the staff agrees that this event frequency meets the criterion of being less than $1x10^{-2}$/yr. Therefore, RAI 15.0-25 is resolved.

15A.2.12 Control Rod Withdrawal Error During Startup

It is postulated that, during reactor startup, a single control rod is inadvertently withdrawn continuously because of a procedural error by the operator during manual rod withdrawal or a gang of control rods is inadvertently withdrawn because of a malfunction in the automated rod movement control system (ganged rod operation) of the plant automation system, when in the automatic startup mode.

GEH estimates that the frequency of an automatic control rod withdrawal is about $1.20x10^{-6}$/yr, and the frequency of manual rod withdrawal is about $1.5x10^{-7}$/yr. With the consideration of uncertainty, these values are less than $1.0x10^{-2}$/yr.

In response to RAI 15.0-25, item B.2, GEH made an assessment of the mechanical failure of the FMCRD and concluded that the mechanical failure of the FMCRD is negligible compared to the estimated overall fail frequency.

The staff agrees with the RAI responses and the failure assessment of this event. Based on the estimated failure frequency of control rod withdrawal error during startup, the staff agrees that this event frequency meets the criterion of being less than $1x10^{-2}$/yr.

15A.2.13 Control Rod Withdrawal Error During Power Operation

The causes of a potential rod withdrawal error at power are either a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously.

GEH estimates that the frequency of an automatic control rod withdrawal is about $1.20x10^{-9}$/yr, and the frequency of manual rod withdrawal is about $2.5x10^{-5}$/yr. With the consideration of uncertainty, these values are less than $1.0x10^{-2}$/yr.

In response to RAI 15.0-25, item B.2, GEH made an assessment of the mechanical failure of the FMCRD and concluded that the mechanical failure of the FMCRD is negligible compared to the estimated overall fail frequency.

The staff agrees with the applicant's RAI responses and the failure assessment of this event. Based on the estimated failure frequency of control rod withdrawal error during power operation, the staff agrees that this event frequency meets the criterion of being less than $1x10^{-2}$/yr.

15A.2.14 Fuel Assembly Loading Error, Mislocated Bundle

The loading of a fuel bundle in an improper location with subsequent operation of the core requires three separate and independent errors:

(1) A bundle must be placed into a wrong location in the core.

(2) The bundle that was supposed to be loaded where the mislocation occurred is also put in an incorrect location or discharged.

(3) The misplaced bundles are overlooked during the core verification process performed following core loading.

Based on the industry survey data, GEH estimates that the mislocated bundle frequency is 9.6×10^{-4}/yr.

The staff has reviewed the failure assessment of not detecting the mislocated bundle and agrees with the estimated failure frequency. Based on the low probability of not detecting a mislocated bundle and the estimate of the frequency of a mislocated bundle, the staff agrees that this event frequency meets the criterion of being less than 1×10^{-2}/yr.

15A.2.15 Fuel Assembly Loading Error, Misoriented Bundle

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and ensured by verification procedures during core loading.

Based on the industry survey data, GEH estimates that the misoriented bundle frequency is 2.4×10^{-3}/yr.

The staff has reviewed the failure assessment of not detecting the misoriented bundle and agrees with the estimated failure frequency. Based on the low probability of a misoriented bundle's not being detected and the estimate of the frequency of a misoriented bundle, the staff agrees that this event frequency meets the criterion of being less than 1×10^{-2}/yr.

15.A.2.16 Liquid-Containing Tank Failure

Based on the industry survey data, GEH estimated that the frequency of this event is 3.3×10^{-4}/yr.

The staff agrees with the assessment that this is a low probability event. Based on the low probability of this event, the staff agrees that this event frequency meets the criterion of being less than 1×10^{-2}/yr.

15.A.3 Conclusions

Pending resolution of the open item discussed above, the staff agrees that the events reviewed for the following had frequencies less than 0.01/yr with the consideration of the uncertainty.

15.5.8 References

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