15. TRANSIENT AND ACCIDENT ANALYSES

15.1 Introduction

In the design control document (DCD), Tier 2, Revision 6, Chapter 15, “Safety Analyses,” the applicant discussed the analysis of various anticipated operational occurrences (AOOs) and accidents. The system response analyses are based on the equilibrium core (EC) described in Chapter 4 of DCD and core loading documented in NEDC-33239-P, “Global Nuclear Fuel, GE14 for ESBWR Nuclear Design Report,” Revision 4, March 2009. 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.

Because the ESBWR design is based on natural circulation, active components designed to ensure a continuous supply of cooling water (as in the current vintage of boiling-water reactors (BWRs)) are not used. Therefore, a number of transients and accidents do not apply to the ESBWR. In this sense, the ESBWR design is unique. In addition, GE-Hitachi Nuclear America LLC (GEH), the applicant, affected a series of instrumentation, mechanical, and electrical design improvements that modified the probability of occurrence of AOOs and accidents. This forced a recategorization of all events. DCD Section 15A was reviewed and approved by the staff in this report.

Because of the uniqueness of the ESBWR design and the recategorization of events in Chapter 15, this review does not strictly follow the SRP for all events. For example, according to Draft Revision 3 of the SRP (1996)[1], all the reactivity transients are AOOs, and the corresponding acceptance criterion of safety limit minimum critical power ratio (SLMCPR) is used. However, in the ESBWR design, most of the reactivity transients (except for control rod withdrawal during startup and power operation) are considered by the applicant and the staff to be in the infrequent category, based on event frequency (a subset of accident category), and hence, 2.5 rem total effective dose equivalent (TEDE) (10 percent of the dose acceptance criteria specified in Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(2)(iv)(A) is used as the acceptance criterion.

15.1.1 Event Categorization

The SRP divides events into AOOs and postulated accidents. The requirements of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” define AOOs as those conditions of normal operation expected to occur one or more times during the life of the nuclear power unit.

[1] The SRP referenced in each Section is the latest revision applicable to that Section.

SRP Section 15.0 defines postulated accidents as “Unanticipated conditions of operation (i.e., not expected to occur during the life of the nuclear power unit).” DCD, Tier 2, Revision 6, Section 15.0, “Analytical Approach,” presents the ESBWR transient and accident analysis methodology used by GEH.

GEH 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 lower frequency of event occurrence, and the unique design features, such as the four divisions of safety systems and three channels of process systems (feedwater control system (FWCS)), which are redundant and fault tolerant. These design features could reduce the frequency of design-basis events (DBEs). In the SRP Section 15.0, Revision 3 “Introduction-Transient and accident analyses” design-basis events are defined as follows;

Conditions of normal operation, including AOOs, design-basis accidents (DBAs), external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; (RCPB) the capability to shutdown the reactor and maintain in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures.

As part of the pre-application review, 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).

New initiating events that require consideration within the scope of accidents and transients may result from the new and unique design features of the ESBWR. For example, the original DCD did 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 issued RAI 15.0-1 to request the applicant to identify all possible transients and accidents that may result from the unique design features of the ESBWR. In its response GEH stated that it performed such study, and the results of this systematic review are listed in the RAI response Table 15.0-1. The categorization adheres to the guidance of Regulatory Guide (RG) 1.70 i.e., to ensure consideration of systems effects. In addition, GEH stated that the CRDS inadvertent initiation event consequences are bounded by the inadvertent initiation of the isolation condenser system (ICS). GEH also presented the results of a study confirming that all equipment, including the CRDS, in the ESBWR was reviewed to determine whether credible failures in the system or operator error could initiate a new type of DBE. In summary, GEH performed the requested study, which covered all the ESBWR systems and addressed possible new events resulting from the unique design features of the ESBWR; 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 the acceptance criteria for four groups of DBEs (AOOs, accidents—IEs, DBAs, and special events).

15.1.1.1 Anticipated Operational Occurrences

AOOs are expected during the life of the plant and require analyses to ensure that they will not cause damage to either the fuel or the 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 (as proposed by the applicant and accepted by the staff) of AOOs for the ESBWR includes events with a frequency greater 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,” in Appendix A to 10 CFR Part 50, 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 that 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 control 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 maintains the reactor water level above the top of active fuel (TAF).

- The plant design should maintain fuel cladding integrity by ensuring that the minimum critical power ratio (MCPR) remains above the applicable staff approved value of the SLMCPR.

- The plant design should maintain pressure in the reactor coolant and main steam systems below 110 percent of the design value (1,375 pounds per square inch gauge (psig)).

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

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

In RAI 15.0-16, staff requested GEH to include the SLMCPR in the technical specifications (TSs). It is the staff’s position that the SLMCPR numerical value should be kept as a safety limit in the TS as in the BWR Standard TS. GEH stated that for the ESBWR TRACG methodology, the transient delta- critical power ratio (CPR) uncertainty is inherently combined with the uncertainties included in the evaluation of the conventional BWR SLMCPR. This process allows for the direct calculation of the NRSBT for a transient occurring from an initial operating condition corresponding to the operating limit minimum critical power ratio (OLMCPR). Therefore, the number of rods subject to boiling transition (NRSBT) parameter becomes the cornerstone of the ESBWR TRACG methodology instead of the SLMCPR, which does not inherently exist for the ESBWR methodology. The staff reviewed the applicant’s response to the RAI and found it to be unacceptable.

The staff based its position on the following:

- Allowing the removal of the SLMCPR eliminates regulatory control of core analysis issues and takes away 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.

- Use of TRACG for calculating the 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 stated that this process allows for the direct calculation of the NRSBT for a transient. GEH maintained 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 BWRs/2–6, and the licensees who currently use the TRACG methodology for calculating the OLMCPR must still have an SLMCPR TS. Specifically, 10 CFR 50.36(d)(1)(i)(A) 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 TS.

The applicant revised their 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 criterion, but a safety limit.

GEH responded that although using the ESBWR TRACG fuel cladding integrity safety limit 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 occur only 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 revised its original response to RAI 15.0-16. The staff reviewed GEH's revised response to RAI 15.0-16, which includes TS Section 2.1.1.1 and the proposed value of the SLMCPR. Based on the applicant's response, RAI 15.0-16 is resolved.

15.1.1.2 Accidents or Infrequent Events (IEs)

SRP Section 15.0 defines DBEs as all transients with a frequency that is less than $1.0x10^{-2}$ pry that may occur during the lifetime of the plant. GEH defines IEs as events with a frequency of less than $1.0x10^{-2}$ pry; therefore the staff considers IEs as DBEs. The DBE criteria for acceptance include radiological consequence less than that of DBAs. DBAs are defined as postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components.

GEH submitted DCD, Tier 2, Revision 6, Appendix 15A, "Event Probability Analyses," providing the determination of the event frequency of the IEs. Section 15.A of this report presents the staff's evaluation of the event frequency determination.

The applicant proposed the inclusion of 16 events in this new category. These events include reactivity, power and pressure anomalies such as control rod withdrawal error (RWE), 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 BWRs, and some of them are new events. 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 TAF.
- The RCPB pressure is less than 1,500 psig (American Society of Mechanical Engineers (ASME) Code Pressure Service Level C, 1,500 psig, 120 percent of the RCS design pressure (1,250 psig).
- The radiological consequence is less than 2.5 rem TEDE, 10 percent of the dose acceptance criteria specified in 10 CFR 52.47(a). The dose acceptance criteria in Table 1 of SRP Section 15.0.3 are fractions of the 10 CFR 52.47(a) dose reference values for accidents other than the loss-of-coolant accident (LOCA), as historically presented. For events having a moderate frequency of occurrence, any release of radioactive material must be such that the calculated offsite doses are a small fraction of the 10 CFR 52.47(a) reference values. The staff has accepted "less than 10 percent" to be a small fraction of the 10 CFR 52.47(a) dose reference values, or 2.5 rem TEDE (SRP Section 15.0.3, Table 1). The DCD states that 1,000 fuel rods is a bounding number for the fuel damage that meets the 2.5-rem criteria.
- Staff requested in RAI 15.3-9 that the applicant provide the actual number of rods in transition boiling in DCD, Tier 2, Revision 1, Section 15.3.1. In response, GEH stated that the bounding number of rods failed is supported by an engineering evaluation of the number of rods under dryout. This number was estimated based on correlating the number of rods under dryout as a function of the core MCPR. The staff accepts this conclusion. Based on the applicant's response, RAI 15.3-9 is resolved.
- The estimated number of rods in boiling transition is less than half of the assumed 1,000 rods in boiling transition. The calculation of the TEDE assumes 1,000 failed rods. The technical bases for the OLMCPR, SLMCPR, and justification of the 1,000 failed rods are included in NEDE-33083, Supplement 3, "TRACG Application for ESBWR Transient Analysis" dated December 2007.
- The plant maintains containment and suppression pool pressures and temperatures below their design values.

- Control room personnel do not receive an accident dose 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 must have a small consequence (protection of the SLMCPR), while events assessed as having a lower frequency may have a more severe consequence (i.e., fuel damage may occur, but radiological dose must fall within the limits set forth in 10 CFR 52.47, “Contents of Applications; Technical Information”). For current operating BWRs, events with a frequency of less than 10^{-2} pry may result in cladding failure, fuel failures, or overpressurization. In the ESBWR, there are IEs, such as feedwater controller failure-maximum flow demand, control RWE, and loss of feedwater heating (LOFWH)), that have a frequency of occurrence of less than 1.0×10^{-2} pry, yet the consequences are similar to those of AOOs (i.e., the calculated MCPR is above the OLMCPR, reactor pressure is less than the relief valve set pressure, and the cladding strain is less than the allowed limit). Other IEs may result in fuel damage, overpressurization, or cladding damage.

GEH proposed ASME Code Service Level C (120 percent of the design pressure) as the criteria for RCPB pressure. DCD, Tier 2, Revision 6, 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 52.47(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). 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 and RAI 15.0-17 S01.

In RAI 15.0-17 S02, the staff stated the DCD should include a commitment to perform post-overpressurization event inspection testing to justify continued operation if any event causes an ESBWR RCS to exceed its ASME Code Service Level B (110 percent) pressure limit.

In response to RAI 15.0-17 S02, GEH stated that the safety analyses demonstrate that no DBE can cause an ESBWR RCS to exceed its ASME Code Service Level B pressure limit. However, ASME Code Section XI does require adequate inspections and 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-overpressure event inspections. The applicant updated DCD, Tier 2, Section 3.9.3.1.2, in Revision 5 in response to this request. The staff accepts this response based on the inspections. Based on the applicant’s response, RAI 15.0-17 is resolved.

In DCD, Tier 2, Section 15.0.1.2(3), the applicant stated, “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 less than 1.0×10^{-2} pry and a radiological consequence less than an accident.” In RAI 15.0-26, the staff indicated that the ESBWR IE classification is a subset of the accident category in the SRP and that the radiological consequence of an IE should be less than that of a DBA. The applicant agreed with the staff’s comment and revised the text in the DCD to read “radiological consequence less than a design-basis accident.” Therefore, based on the applicant’s response, RAI 15.0-26 is resolved.

The staff performed independent confirmatory calculations for limiting AOOs with the TRACE/PARCS computer code. Section 21.6 of this report presents the staff's evaluation of the applicant's analyses and the staff's independent calculation results.

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

15.1.1.3 Design-Basis Accidents

SRP Section 15.0 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 doses less than 2.5 rem TEDE, 6.3 rem TEDE, or 25 rem TEDE (see SRP Section 15.0.3, Table 1), the acceptable radiation dose criteria in 10 CFR 52.47(a)(2), depending on the accident-specific acceptance criterion in Chapter 15 of the SRP.

The DBA category includes the following:

- fuel-handling accidents (FHAs)
- main steamline break (MSLB) outside containment
- control rod drop accident (CRDA)
- feedwater line break outside containment
- failure of small line carrying primary coolant outside containment
- reactor water cleanup/shutdown cooling (RWCU/SDC) system line failure outside containment
- spent fuel cask drop accident

For LOCAs, the acceptance criteria for the emergency core cooling system (ECCS) 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 1,204.4 degrees Celsius (C) (2,200 degrees Fahrenheit (F)).
- For 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.

Section 6.3 of this report presents the staff evaluation of compliance with 10 CFR 50.46.

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 and coping, mitigation, and acceptance criteria specified in the NRC regulations and the SRP.

15.1.2 Analytical Methods

TRACG is a multidimensional, two-fluid reactor thermal-hydraulics (T-H) analysis code with a 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.

For nuclear analyses, the applicant's suite of codes includes a two-dimensional lattice physics code (TGBLA06) and a three-dimensional core simulator PANAC11. These codes are used in conjunction to perform several analyses to demonstrate ESBWR compliance with GDC. Sections 4.3 and 21 of this safety evaluation report (SER) include additional information on these codes.

GEH transient analyses used the TRACG evaluation model, described in licensing topical report (LTR) NEDE-33083P, "TRACG Application for ESBWR, Transient Analysis, and Supplement 3," issued December 2007, 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 IE analyses. In RAI 15.0-27, the staff requested GEH to include discussion of IE analyses.

In response to RAI 15.0-27, GEH submitted Supplement 3 of NEDE-33083, which contains analyses of AOOs, IEs, and special events. Based on the applicant's response, RAI 15.0-27 is resolved.

15.1.3 Non-Safety-Related Systems Assumed in the Analysis

In RAI 15.0-2, the staff requested GEH to provide a list of non-safety-related systems and equipment credited in the analyses. 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 Criterion 3 specified in 10 CFR 50.36(c)(2)(C), the TS must include LCOs for 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 RAI 16.2-33, the staff requested GEH to review the response to RAI 16.0-1 (bases for the TS) in light of the response to RAI 15.0-2 and identify any changes to TS. In its response to RAI 16.2-33, GEH stated the following:

> 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 (IC) and GDCS 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 TS)
- selected control rod run in (SCRRI) (included in the TS)
- fuel and auxiliary pool cooling system (FAPCS) (not included in the TS)
- FWCS (not included in the TS)
- rod control and information system (RC&IS) (rod worth minimizer (RWM) and automated thermal limit monitor (ATLM) are included in the TS)
- steam bypass and control system (included in the TS)

In RAI 15.0-2 S02, the staff requested that GEH confirm that all equipment credited in the analyses be included in the TS. In response to RAI 15.0-2 S02, GEH revised DCD, Tier 2, Tables 15.1-5 and 15.1-6, to show the list of non-safety-grade equipment for which credit was taken in accident analysis.

In RAI 15.0-1 S01 the staff requested GEH to justify why the high-pressure control rod drive (HPCRD) should not be in the TSs. In response to RAI 15.0-1 S01, the applicant stated that "In the case where HPCRD is unavailable for reactor vessel water level control, the system response is similar to the SBO events described in DCD Section 15.5.5 which demonstrates that the level can be maintained above the top of the active fuel with the ICS as the primary success path." DCD Section 15.5.5 presents a description and analysis of SBO where a loss of feedwater flow and control rod drive (CRD) flow is assumed. The SBO analysis bounds the events where operation of the HPCRD is required. Since the acceptance criteria can be met without the HPCRD system, this system is not required to be in the TS therefore; RAI 15.0-1 S01 is resolved.

In response to RAI 15.0-2, the applicant stated that the suppression pool cooling mode of FAPCS is credited with long-term cooling of the suppression pool following an inadvertent opening of a safety/relief valve (SRV). With no operation of the FAPCS in the suppression pool cooling mode, the pool would heat up to its scram setpoint and initiate a scram if one has not already occurred. Containment design limits will not be exceeded. Hence, FAPCS is not critical equipment and need not be in the TS.

The applicant, in response to RAI 16.2-33 S01, provided additional information related to the FWCS. The FWCS is credited in the limiting event, "Inadvertent Isolation Condenser Initiation" (IICI). Failure of the FWCS simultaneously with an IICI event is a detectable and non-consequential random, independent failure, and the automatic function of the FWCS is not in the primary success path for the mitigation of an IICI event.

The applicant stated that non-safety-related systems or components are assumed to be operational in the following situations:

- when assumption of a non-safety-related system results in a more limiting transient
- when a detectable and non-consequential random, independent failure must occur in order to disable the system
- when non-safety-related components are used as backup protection (e.g., the HPCRD system, which is not in the primary success path but is included to illustrate the expected plant response to the event)

In the above circumstances a non-safety system failure (1) will not result in a more limiting transient, (2) will occur only when a detectable independent failure disables the system and (3) non-safety-related systems will be used for backup protection. The staff finds acceptable the assumptions concerning the non-safety-related systems described above.

Based on the applicant's response, RAIs 15.0-2, 16.0-1, and 16.2-33 are resolved.

15.1. 4 Loss of Offsite Power Assumption

In RAI 15.0-4 the staff requested GEH to describe in detail how the ESBWR transient and accident analyses were performed to comply with GDC 17 and in particular " For new applications, loss of offsite power should not be considered as a single failure event; rather it should be assumed in the analysis of each event without changing the event category…" GEH addressed compliance with GDC 17 with regard to DCD, Tier 2, Chapter 15, analyses in its response to RAI 15.0-4. 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 loss of offsite power. Since the ESBWR has no reactor recirculation pumps that normally receive their power supply from off site, loss of offsite power is not a significant event for the ESBWR. Chapter 8 of this report includes a detailed evaluation of GDC 17. Based on the description of the ESBWR plant features in the applicant's response, RAI 15.0-4 is resolved.

15.1.5 Analysis of Anticipated Operational Occurrences and Infrequent Events for the Initial Core

GEH submitted NEDO-33337, "ESBWR Initial Core Transient and Accident Analyses," Revision 1, April 2009 (which is incorporated into the DCD via reference), which includes the analyses of the initial core and NEDO-33338, Revision 1, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," issued May 2009 (see Section 15.1.1.9). The staff, with the help of Brookhaven National Laboratory and ORNL respectively, reviewed these LTRs. A summary of the staff's evaluation of the initial core analyses follows.

The evaluation, based on the review of Chapter 15 of the DCD, incorporates (for each transient) a summary of limiting features from the evaluation of NEDO-33337, Revision 1, and NEDO-33338, Revision 1, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," issued May 2009 (see Section 15.1.1.9). The initial core loading (ICL) and the feedwater temperature operating domain (FWTOD) additions (see Figure 15.1-1 of this report) to the DCD were reviewed and evaluated independently of the reference core. It should be noted that the ICL is a simulation of the EC constructed with varying enrichment and poison loadings. These safety evaluations (Sections 15.2 and 15.3) are based on the reference core DCD, Tier 2, Revision 6, with additional segments on the ICL and the FWTOD for each transient. The ICL (NEDO-33337) and the FWTOD core (NEDO-33338) evaluations were reviewed separately.

This evaluation includes the limiting characteristics for each transient affected by the ICL and/or the FWTOD operation so that the most limiting parameters are included. Evaluation of the AOOs and the IEs is based on the acceptance criteria summarized in Sections 15.1.1.1 and 15.1.1.2 of this report.

GEH transient analyses used the TRACG evaluation model, described in LTR NEDE-33083P Supplement 3, to analyze most of the AOOs and IEs. This LTR is based on calculation results for ESBWR AOOs and IEs.

In NEDO-33337, the applicant discussed the analysis of various AOOs. GEH analyzed the following categories of AOOs:

- decrease in core coolant temperature
- increase in reactor pressure
- reactivity and power distribution anomalies
- increase in reactor coolant inventory
- decrease in reactor coolant inventory

In addition, the document discusses IEs and special events. IEs fall under the broad category of accidents, but reflect the unique passive cooling capabilities possible with the ESBWR design. Special events have an extremely low probability of occurrence and, in this case, include ATWS, SBO, etc. The evaluation of the results presented in this report for the ICL is based in large part on the comparison between these results and those obtained for the same transient for the EC.

The primary differences between the ICL and EC are fuel loading and cycle burnup. In an effort to mimic the EC, the ICL has many more fuel types regarding fissile material enrichment. In addition, the ICL has lower gadolinium concentration than the EC. Briefly, the ESBWR fuel assembly has a 10x10 fuel pin structure with two large water rods that correspond to 8 fuel pin positions. Thus, there are a maximum of 92 positions available for fuel pins. Several of these pins contain gadolinium as a burnable poison, with a variety of gadolinia loadings.

These differences result in a softer neutron energy spectrum and an axial power shape that is preferentially more peaked in the bottom half of the core for the initial core compared to the EC. These differences have the following implications for the transients to be considered here:

1) Scram Worth—The ICL has a more bottom-peaked axial power shape compared to the EC thus, as the control rods enter the core from the bottom, they have a more pronounced effect on core reactivity. In addition, the average neutron energy in the ICL core is lower (softer spectrum) than for the EC. This lower neutron energy enhances the neutron absorption in the control rods and their reactivity worth.

2) Void Reactivity—The ICL will have a lower void reactivity worth because the average neutron energy is lower in the ICL compared to the EC, which implies less under-moderation relative to the EC.

As a result of these two effects, the kinetic response of the initial core is expected to have lower power peaks for transients dominated by void collapse due to primary system pressure increases than the EC.

Finally, the decay heat contribution for the ICL will be lower than that corresponding to the EC. This is due to the lack of actinides in the fuel mix and the lower burnup period. This effect is expected to be small.

The CPR is also an important parameter that must be considered when comparing the initial and ECs. Briefly, the CPR is a measure of the allowable change or variation in the flow and power levels in a given assembly to avoid boiling transition. In the case of transient analyses, ΔCPR/ICPR (initial CPR) measures the change in the CPR as the transient progresses. This combination of parameters is determined for each AOO and IE analyzed.

The analyses of AOO transients are divided into the categories mentioned above. Each of the following is considered:

- Decrease in Inlet Coolant Temperature—one transient was analyzed in this group. This transient involves decrease in feedwater inlet temperature that results from failure in the feedwater heating system. In this case, the SCRRI/select rod insert (SRI) system is not credited in the analysis of the initial core, which results in a 16-percent increase in maximum neutron flux, a 19-percent increase in the maximum of the average heat flux, and a doubling of the ΔCPR/ICPR, compared to the EC. The EC analysis credits the SCRRI system.

- Increase in Reactor Pressure—the nine transients in this group have in common increase in reactor pressure resulting from closure of the main steam line isolation valve(s) (MSIVs). MSIV closure results in a sudden increase in reactor pressure that collapses the core voids. The MSIV closure rate and other mitigating factors characteristic of the transient being analyzed determine the extent of the in-core void collapse. This includes the amount of reactivity added to the core and the rate at which it is added. In all cases, the peak neutron flux is lower for the initial core than for the EC. This is due to the lower void reactivity worth associated with the initial core. The overall change between the ICL and EC analyses for core pressure increase and the maximum of the average heat flux is seen to be minimal. There are significant increases for the ΔCPR/ICPR for selected cases, which can be ascribed to changes in the SCRRI/SRI rod pattern used in the ICL compared to the EC.

- Reactivity and Power Distribution Anomalies—the EC analysis in the DCD applies to the initial core (except for the control RWE during startup with failure of control rod block for which the analysis used the initial core).

- Increase in Coolant Inventory—the two transients analyzed in this group are runout of one feedwater pump and inadvertent initiation of IC. The results for both the initial core and the EC are in good agreement.

- Decrease in Coolant Inventory—three transients are considered in this group. These transients are not characterized by a single theme, but are the result of increased flow out of the core or decreased flow into the core. The results for both the ICL and the EC are in good agreement.

In all but two cases, the results for the ICL and those for the EC agree. The two exceptions are generator load rejection and turbine trip with total bypass failure. These events are similar in that they both result in a turbine trip; the difference is in the timing of the sequences. In both cases, the maximum neutron flux determined for the ICL is about 20 percent lower than that corresponding to the EC. In addition, the value determined for ΔCPR/ICPR for the ICL is about

26 percent lower than that determined for the EC. Both of these deviations can be attributed to the lower ICL void reactivity worth. The other parameters, such as the reactor vessel pressures and maximum average heat flux determined for the ICL, are close to the corresponding EC values.

The analyses of IEs considered eight events. These events involved more than one system failure, thus making them less likely than the AOO transients discussed above, but potentially with more severe consequences.

The three events included in the special events category include an SBO and two ATWS events. The SBO assumes that the external power is cut off, and the station has to rely on standby power. After a 72-hour period, the calculated parameters for the ICL and those of the EC are essentially the same. The first ATWS event involves an MSIV closure with standby liquid control system (SLCS) activation, and the second ATWS event involves a loss of condenser vacuum with SLCS activation. In both cases, an initial power pulse is mitigated by feedwater runback and SLCS activation. In both cases, the results for the ICL have a maximum neutron flux that is about 10 percent lower than that determined for the EC.

Despite this difference, the primary system vessel pressure, suppression pool temperature, and containment pressure determined for the ICL case are very close to the results determined for the EC.

However, the calculated peak clad temperature for ICL is 835°C (1535°F) (Table 2.5-4-4a, NEDO-33337, Rev. 1) versus 928.25°C (1702.9°F) (DCD Table 15.5-4c) for the ICL and EC cases respectively. The lower ICL values relative to the EC are primarily the result of the lower reactivity insertion associated with the lower void reactivity worth of the initial core.

15.1.6 Feedwater Temperature Operating Domain

Appendix 15D to the DCD summarizes the effect of feedwater temperature (FWT) variation on AOOs, IEs, special events, and LOCAs. GEH also submitted NEDO-33338, Revision 1, which provides the analyses of the initial core and the EC of Chapter 15 transients for operation in the FWTOD. The staff, with the help of Oak Ridge National Laboratory, reviewed this LTR. The following summarizes the staff's evaluation of the analyses.

In LTR NEDO-33338, Revision 1, the applicant describes a broadening of the operating domain, which allows for increased flexibility of operation by adjusting the FWT. This increased flexibility is required to accommodate the so-called "soft" operating practices, which reduce the duty to the fuel and minimize the probability of pellet-clad interactions and associated fuel failures.

By adjusting the FWT, the operator can reduce or increase the reactor power without control blade motion and with minimum impact on the fuel duty. Control blade maneuvering can also be performed at lower power levels.

To control the FWT, the ESBWR design includes a seventh feedwater heater with high-pressure steam. FWT is controlled by either manipulating the main steam flow to the No. 7 feedwater heater to increase FWT above the temperature normally provided by the feedwater heaters with turbine extraction steam (normal FWT) or by directing a portion of the feedwater flow around the high-pressure feedwater heaters to decrease FWT below the normal FWT. An increase in FWT decreases reactor power, and a decrease in FWT increases reactor power. At 100 percent of rated thermal power conditions, the addition of the seventh stage feedwater heaters in full service provides an increase of approximately 36.7 degrees C (66 degrees F) in the FWT, which corresponds to a reduction of approximately 15 percent in the core power.

Figure 15.1-3 shows the ESBWR power FWTOD (P-FWTOD). It has two distinct regions: a feedwater temperature increase (FWTI) region and a feedwater temperature reduction (FWTR) region. The FWTI region is used to reduce the power before control blade maneuvering, both during startup and for normal rod-sequence exchanges. The FWTR region allows operating flexibility and could be used to control day-to-day burnup in a manner similar to the power-flow control with operating reactors.

Five major points are defined in the power FWTOD:

(1) SP0 is the nominal operating state point—100 percent power, 100 percent FWT 216 degrees C (420 degrees F).

(2) SP2 is the increased FWT state point—85 percent power and 252 degrees C (486 degrees F). This FWT corresponds to operation with the seventh feedwater heater at full capacity.

(3) SP1 is the reduced FWT state point—100 percent power and 160 degrees C (320 degrees F). This FWT represents a reduction of 55.56 degrees C (100 degrees F), which is the maximum credible FWT transient caused, by a single failure in the FWCS. SP1 is defined only for bounding calculations; the power flow relationship of FWTOD is limited by the SP1M state point.

(4) SP1M is the stability-bounding reduced FWT state point—100 percent power and 187 degrees C (370 degrees F). The FWT of point SP1M is defined on a cycle-specific basis and will be documented in the core operating limits report (COLR). Point SP1M is defined so that the reactor remains stable following an additional 16.6 degrees C (30 degrees F) FWT reduction caused by an inadvertent LOFWH.

    Note: LOFWH with temperature reductions greater than 16.6 degrees C (30 degrees F) would result in actuation of SCRRI/SRI control blade insertion, which would terminate the transient before instability is a concern.

(5) SP5 is the 50-percent power point at nominal FWT. At powers below 50 percent, the ESBWR will operate with the nominal feedwater heater configuration (i.e., heater Nos. 1 through 6 on, and no steam supply to heater No. 7).

One of the primary advantages of the FWTOD is realized during reactor startup. Figure 15.1-2 shows a typical startup path for the ESBWR. Up to point SP5, the operator simply pulls control blades with feedwater heaters Nos. 1 through 6 fully open and feedwater heater No. 7 closed, because rod heatup is not a concern when the power is below 50 percent. As the power increases, the turbine control valve (TCV) opens to provide higher steam flow to the turbine to maintain constant reactor pressure.

As the TCV opens, the pressure drop through the valve is reduced, and the turbine inlet pressure is higher, thus providing higher pressure steam to the feedwater heaters, which increases the shell-side steam-condensation temperature. Therefore, the tube-side FWT increases with power automatically.

After point SP5 is reached, the power is high enough that rod heatup limits and possible pellet clad interactions become a concern. To prevent fuel duty issues, the operator will alternate further control blade withdrawals with increases on the steam supply to feedwater heater No. 7. After all the rods are withdrawn following this sequence (rod pull, FWT increase), point SP2 will be reached with the target control blade pattern, while minimizing the duty to the fuel because the blade withdrawals occur at lower power levels. Finally, the full operating power (point SP0) is reached by slowly turning off feedwater heater No. 7 and decreasing the FWT to nominal conditions.

To optimize the core isotopics and burnup management, periodic control rod exchanges are performed with preplanned (preconfigured) control rod patterns. For this periodic rod sequencing, feedwater heater No. 7 is brought slowly back into operation, and the reactor maneuvers from point SP0 to SP2. The rod pattern exchange is performed at the lower power. After the rod exchange, the power is increased by slowly removing feedwater heater No. 7 from operation. Fine reactivity control may be achieved between rod sequence exchanges by partially bypassing feedwater heater No. 6 and operating between the points SP0 and SP1M at reduced FWT. During most of the cycle, gadolinium burnup results in a reactivity increase; thus, burnup control would require slowly increasing the FWT and moving towards point SP0. Towards the end of the cycle, uranium-235 burnup dominates and the reactivity decreases, thus requiring a reduction of FWT (i.e., moving towards state point SP1M).

Operation in the complete FWTOD is possible at some time during the cycle, although most of the operating time should occur at or near point SP0, where the balance of plant has been optimized for efficiency. The FWTI region in Figure 15.1-3 (i.e., point SP2) is expected to be used during startup and rod pattern exchanges. The FWTR region in Figure 15.1-3 (i.e., the line between SP0 and SP1M) may be used for fine reactivity control or for reactivity stretching towards end of cycle (EOC). In RAI 4.3-25, the staff requested whether it is possible to use the region between SP0 and SP1M to provide an-end-of cycle stretch. In response to the staff's RAI 4.3-25, the applicant specified that EOC stretch will not be accomplished by reducing FWT in the ESBWR. If in the future, a licensee intends to implement low-temperature EOC stretch, additional analyses will need to be performed and reviewed.

Figure 15.1-3 shows a detailed FWTOD map that defines not only the operating domain but also the protection system actuation lines, as well as the control system blocks on control rod and feedwater valve actuation. These blocks indicate the region where operation is restricted by means of the conventional control system. Unintended operation outside this region results in a control room alarm.

The applicant has provided an evaluation of the impact of varying FWT according to this map of the reactor transient response in NEDO-33338, Revision 1, for the initial core design documented in NEDO-33326, Revision 1, "GE14E for ESBWR Initial Core Design Nuclear Report," issued March 2009, and the DCD EC analysis in Chapter 15. The applicant has compared the results against the nominal operating point (SP0) reported in ESBWR DCD Chapter 15 for an EC and in NEDO-33337 for the ICL. All analyses have been performed using the approved version of the TRACG code.

The results of the analyses indicate that the most limiting AOO is the IICI at SP2 state point for both equilibrium and initial cores. The most limiting IE is the generator load rejection with total bypass failure at SP1 for the EC and at SP0 for the initial core.

However, the largest change in the CPR when the FWT is changed occurs for the LOFWH event, and therefore, it may have a larger impact on the operating limit MCPR at points SP1M and SP2.

The applicant has evaluated the impact of FWT and cycle-specific conditions for special events, including ATWS, ATWS stability, and LOCA. Ample margins to criteria are demonstrated, and the impact of cycle-specific conditions or FWT is calculated to be insignificant.

15.1.6.1 FWTOD Summary

The NRC staff concludes that the applicant has adequately accounted for the effects of the proposed FWTOD extension on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients and that the effects of postulated transients and accidents will not impair the capability to cool the core. Based on this evaluation, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable regulatory requirements. Therefore, the NRC staff finds the proposed -FWTOD extension acceptable

15.1.7 Post-Combined-License Activity

The staff's main conclusion is that the broadening of the ESBWR operating domain by adjusting the FWT is acceptable; however, the following post-combined-license (post-COL) actions are required per TS 5.6.3 "Core Operating Limits Report (COLR)" to satisfy applicable regulatory criteria.

- An operating limit should be established for the OLMCPR that is a function of FWT; thus, higher operating margin is provided at off-nominal FWTs. The OLMCPR is cycle dependent and will be documented in the COLR.

- The minimum FWT of operating point SP1M should be limited to ensure that stability criteria are satisfied. The FWT of point SP1M will be cycle dependent based on the result of stability analyses and will be documented in the COLR.

- In RAI 4.3-16, the staff requested the applicant to explain how the power-temperature operating domain will be defined. In response to RAI 4.3-16, GEH added a reference to NEDO-33338 in COLR Section 5.6.3 (b)(8) to indicate that the staff had approved the analytical method used for determining the cycle-specific operating thermal and stability limits.

- In RAI 4.3-25, the staff requested the applicant to explain whether it is possible to use the region between SP0 and SP1M to provide an end-of-cycle stretch. In response to the staff's RAI 4.3-25 (MFN 08-653 dated August 27, 2008), the applicant stated that if a future licensee or applicant intends to use the region between SP0 and SP1M for an end-of-cycle stretch, additional analyses similar to those required for end-of-cycle stretch for operating BWRs will be performed. If in the future, a licensee intends to implement low-temperature EOC stretch, additional analyses as stated above will need to be performed and reviewed. Since the applicant meets the regulatory requirements and based on the applicant's commitment to provide additional analyses for end-of-cycle stretch operation, RAIs 4.3-16 and 4.3-25 are resolved.

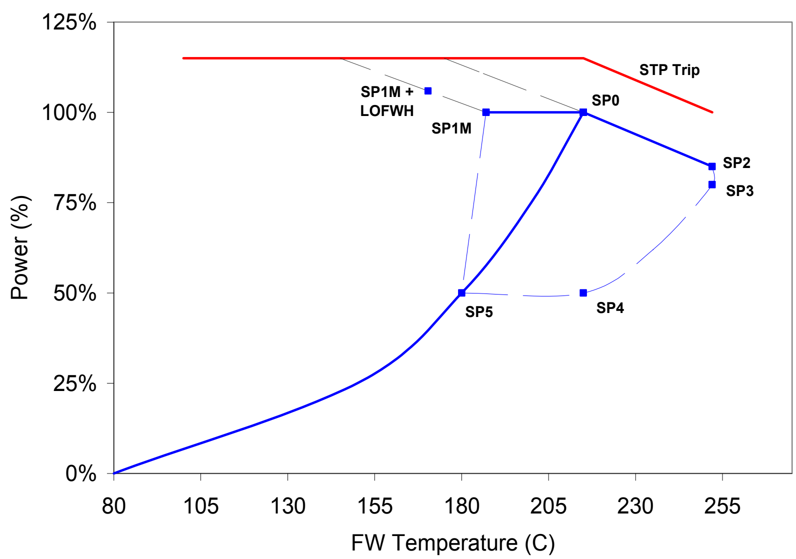

Figure 15.1-1 ESBWR Power-Feedwater Temperature Operating Domain

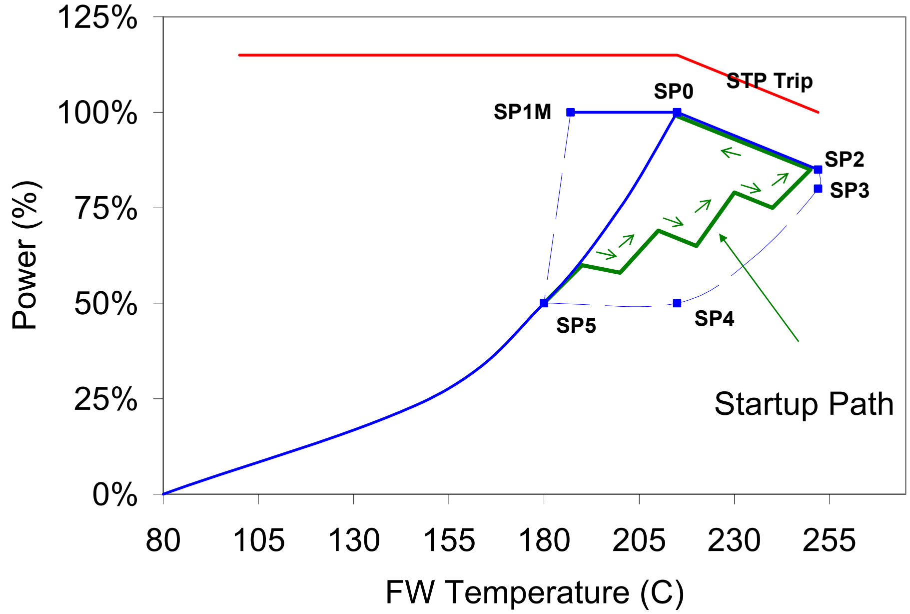

Figure 15.1-2 Typical ESBWR Startup Path

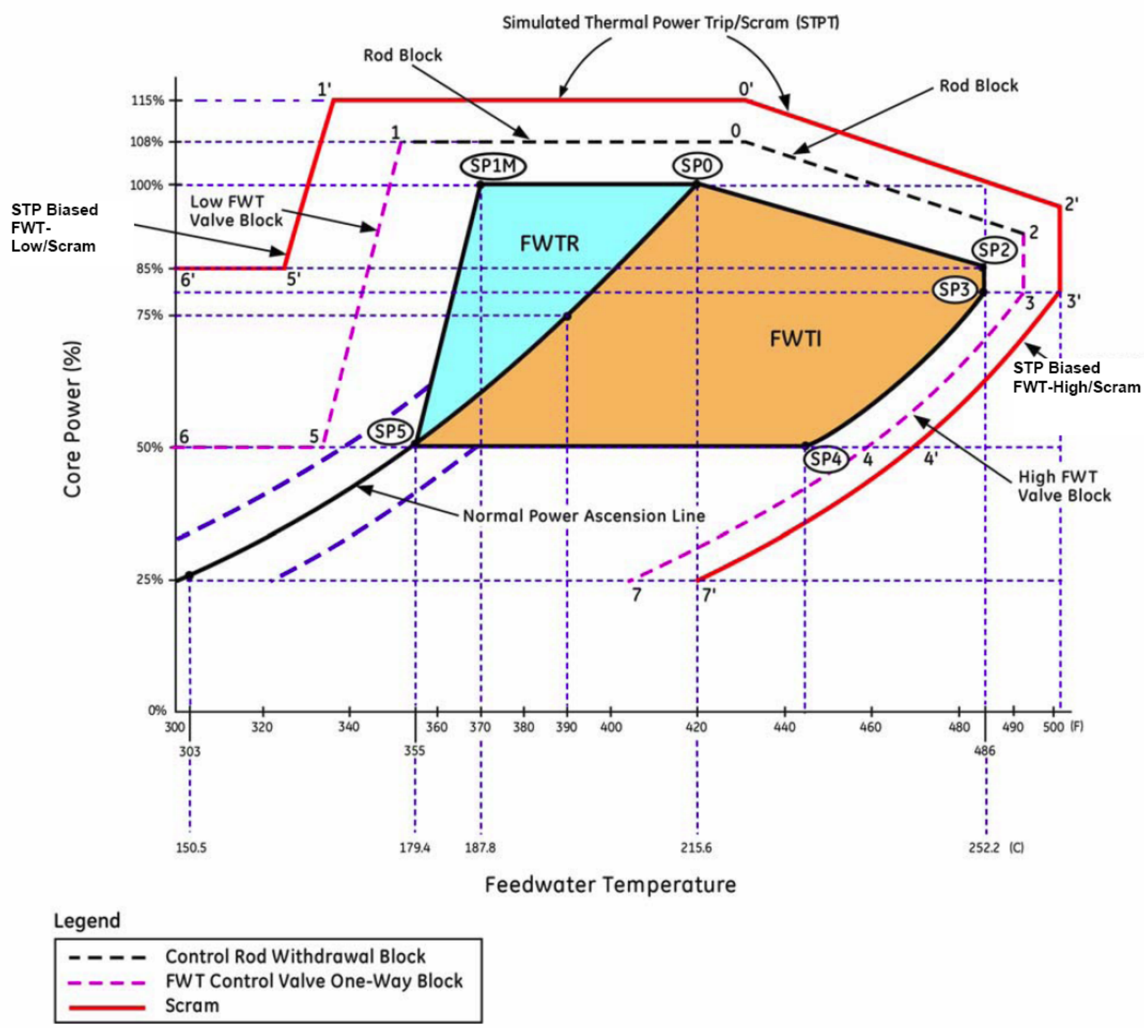

Figure 15.1-3 ESBWR Operating Domain Showing Protection and Control Functions

15.2 <u>Analyses of Anticipated Operational Occurrences</u>

DCD Section 15.2 provides the analyses of AOOs.

The ESBWR design incorporates several features (in addition to natural circulation cooling), such as the four IC units, instrumentation with triplicate digital electronic circuits, more than 100-percent steam bypass capacity, that forestall the evolution of AOOs into a more serious transient and also reduce reactor scram frequency. Another notable feature is the control rod operation in the SCRRI/SRI configuration. SCRRI is a set of sequential- insertion low-speed control rod gang-assemblies to lower power and prevent a scram. However, SCRRI insertion at high power levels could compress power upwards and possibly threaten thermal limits. SRI is a set of fast, hydraulic full-insertion control rods that lower power to prevent possible violation of thermal limits in anticipation of SCRRI insertion. The combination of lowering the average reactor power level would prevent violation of thermal limits that could arise due to SCRRI malfunction. In addition, core flow will increase with lower power which will decrease the oscillation decay ratios. Independently, the "Oscillation Power Range Monitor" will detect and suppress any T-H instability. Therefore, SCRRI augmented with SRI avoids and prevents violation of the thermal limits as well as instabilities. This conclusion resolves RAI 15.2-5 (see Section 15.2.1.1.2 of this report).

DCD Figure 7.7-1 shows the definitions of RPV water-level ranges. Level designation L1 is about 0.5 meters (1.64 Ft) above TAF. L2 is about 8.5 meters (27.89 Ft) above TAF and initiates the IC and CRD pump. L3 is about 12.5 meters (41.01 Ft) above TAF and initiates a reactor low-level scram. L4–6 is the normal operating range. L7 is the high-vessel-level alarm setpoint. L8 is about 14.5 meters (47.57 Ft) above TAF and initiates a high-reactor-water level scram. These level designations are used in Sections 15.2 and 15.3.

GEH analyzed the following categories of AOOs in the DCD sections indicated:

15.2.1 Decrease in Core Coolant Temperature
15.2.2 Increase in Reactor Pressure
15.2.3 Reactivity and Power Distribution Anomalies
15.2.4 Increase in Reactor Coolant Inventory
15.2.5 Decrease in Reactor Coolant Inventory

15.2.1 Decrease in Core Coolant Temperature

The staff used the acceptance criteria in Section 15.1.1.1 in evaluating this AOO.

15.2.1.1 Loss of Feedwater Heating

15.2.1.1.1 Summary of Technical Information

LOFWH decreases the FWT, which in turn decreases core voids and increases moderation and power.

The ESBWR can lose feedwater heating in at least two ways: closing of the heater to the steam extraction line and the feedwater bypassing the heater. The ESBWR design is such that no single failure or operator error will cause LOFWH that would result in a temperature decrease greater than 55.56 degrees C (100 degrees F). The ATLMs will detect a decrease in the FWT, and the diverse protection system (DPS) will initiate SCRRI/SRI insertion to lower power and avert a scram. Although FWT reduction is sensed well before the colder water reaches the core, the analysis conservatively assumes that control rod insertion does not initiate until after core power begins to increase.

When the temperature decrease remains less than or equal to 16.7 degrees C (30 degrees F), the SCRRI/SRI system is not activated, and the power could reach 106 percent of the normal power level, but the ΔCPR/ICPR value is bounded by the inadvertent initiation of an IC, which is analyzed in Section 15.2.4.1 of this report. DCD, Tier 2, Table 15.2-5 and Figure 15.2-1, demonstrate the results of the analysis for this transient. The sequential SCRRI/SRI insertion and the calculated power transient clearly indicate that the reactor attains a lower power level and reactor scram is not needed.

15.2.1.1.2 Technical Evaluation

EC: LOFWH decreases core inlet temperature, resulting in increased moderation due to core void collapse and an increase in core power. The ESBWR has design features that (1) limit feedwater maximum inlet ΔT to 55.56 degrees C (100 degrees F), (2) employs the ATLM, which reduces power to avoid exceeding the thermal limits, and (3) includes the DPS. Either the ATLM or the DPS can activate the SRI to lower power in a fast mode to avoid violation of safety limits and reactor scram. The results are summarized in DCD, Tier 2, Figure 15.2-1, which indicates that the maximum pressure, water level, and MCPR are all well within normal operating limits. If the FWT decrease is less than 16.67 degrees C (30 degrees F), the SCRRI/SRI is not activated, and the power may increase up to 106 percent of normal power. Either way, the resulting transient is bounded by the IICI, which is discussed in Section 15.2.4.1 and has been found acceptable. This AOO does not induce a more serious condition and does not result in a reactor scram.

In RAI 15.2-5, the staff requested the applicant to explain how the reactor, in the event of a partial failure of the SCRRI, would avoid violating local thermal limits or creating 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 reactivity control. This information is derived from the DCD and includes resolution of RAI 15.2-5 regarding operation of the SCRRI/SRI to mitigate the transient. SRI is a fast insertion set of control rods that lower power followed by the sequential low speed SCRRI control rod insertion that avoids reactor scram. Based on the applicant’s response, RAI 15.2-5 is resolved because the results satisfy the applicable acceptance criteria listed in Section 15.1.1.1.

ICL: The LOFWH AOO has been analyzed for the ICL in the same manner using the same assumptions as in the EC. The results of the analysis are close to those for the EC. The feedwater flow increases between 25 and 100 seconds into the transient due to the collapse of the core voids. In both instances, steady operation is achieved in about 200 seconds at about 50 percent of rated power. Pressure remains normal for about 50 seconds and drifts lower at 200 seconds. The MCPR remains well above the OLMCPR. This AOO, does not result in reactor scram; therefore, the LOFWH AOO for the ICL satisfies the acceptance criteria in Section 15.1.1.1.

FWTOD: GEH analyzed the excessive heat removal events and concluded that the LOFWH AOO is limiting. The decreased FWT results in higher power due to increased moderation and thus a decrease in the MCPR. The most limiting LOFWH AOO is the EC at SP0 state point which results in a limiting ΔCPR/ICPR for SP0. Staff review of the analyses indicates that the assumptions are conservative and the MCPR remains above the SLMCPR and hence the results are acceptable because none of the analyzed cases results in reactor scram.

15.2.1.1.3 Conclusion

The comparison of the initial core and the EC analyses results indicate that the equilibrium analysis at SP0 is limiting for the LOFWH AOOs. As stated above, the results of the analyses for the LOFWH AOOs for EC, ICL, and FWTOD satisfy the acceptance criteria.

15.2.1.1.4 Post-COL Activity

This event is potentially limiting with respect to OLMCPR, because of the effect cycle-to-cycle changes to the SCRRI/SRI rod pattern have on ΔCPR/ICPR. This event is analyzed for each fuel cycle with the limiting SCRRI/SRI rod pattern which could be changing from cycle to cycle. The SCRRI/SRI requirements are documented in the COLR in accordance with TSs. The OLMCPR is established for the limiting event and documented in the COLR in accordance with TSs.

In RAI 15.2-5 the staff requested the applicant to explain how the reactor, in the event of a partial failure of the SCRRI, would avoid violating the thermal limits. The resolution of RAI 15.2-5 is documented in Section 15.2.1.2.2 in this SER and the SCRRI/SRI limitations are in the TS Section 5.6.3, COLR (a)(6); TS 3.7.6.

15.2.2 Increase in Reactor Pressure

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

15.2.2.1 Closure of One Turbine Control Valve

15.2.2.1.1 Summary of Technical Information

<u>EC</u>: The steam bypass and pressure control (SB&PC) system includes the TCVs. 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 TCV closes inadvertently (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, depending on the turbine steam admission design (full or partial arc) and the flow through the remaining three TCVs. With one TCV closed, flow through the remaining three valves is 95 percent for full arc and 85 percent for partial arc in the reference core. Therefore, the partial arc case is conservative with regard to void collapse and resulting power and pressure peak. In addition, fast and slow valve closing is assumed. The full-stroke, rated steam flow closure time could be either 0.08 seconds (fast) or 2.5 seconds (slow). Both cases were analyzed (i.e., for fast and slow TCV closure times). The analyses presented in DCD, Tier 2 are based on partial arc (i.e., the most conservative cases). The analytical results for the fast- and slow-closure cases are presented in Tables 15.2-6 and 15.2-7 and Figures 15.2-2 and 15.2-3 respectively. Both instances are in the pressure increase category.

In the fast-closure case, there is a power peak over 120 percent of rated power at around

1.0 seconds, followed by 110-percent feedwater flow peak at about 6 seconds and steady state at about 10 seconds. The flow obstruction, increased pressure which caused partial collapse of the core void that increased moderation and power that farther increased pressure. Peak power from void collapse mirrors core reactivity as a function of time from transient initiation. The MCPR remains above the OLMCPR for both the fast-closure and the slow-closure case.

In the slow-closure case, the transients are similar except that the power peak is about 110 percent at 3 seconds and 112 percent feedwater peak at about 8 seconds. Steady state occurs at about the same time and to the same levels of power and TCV flow. The pressure shows little change in either the fast- or slow-closure case.

ICL and FWTOD: The same transients have been analyzed for the ICL and the FWTOD and the results for both cases are similar. In the ICL fast closure, the power peak exceeds the scram limit. In the case of FWTOD, because of the higher void fraction, the pressure increase will cause larger reactivity insertion and the power peak also exceeds the reactor scram limit. In both cases, the scram is ignored, and the transients are conservatively analyzed.

15.2.2.1.2 Technical Evaluation

The following comments and conclusions apply to both the fast- and the slow-opening TCV.

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

ICL: GEH analyzed both instances for fast and slow TCV closing. The results of the fast-closing case bound those of the slow case. The results for the ICL are bounded by the equilibrium case and, therefore; are acceptable.

FWTOD: This transient has been reviewed in the context of the NEDO-33338 review, and the staff finds that it is bounded by IICI events analyzed at the SP2 state point.

15.2.2.1.3 Conclusion

Based on the above analyses, the NRC staff concludes that inadvertent closure of one TCV at full power for the EC, the ICL, or the FWTOD satisfies the acceptance criteria in Section 15.1.1.1, is bounded by other transients in this report, and, therefore, is acceptable. Because the SCRRI/SRI insertion is part of the transient scenario and the OLMCPR depends on fuel core loading, this transient should be analyzed in each cycle as in Section 15.2.1 of this report

15.2.2.1.3 Post-COL Activity

This event (fast closure) is potentially limiting with respect to OLMCPR which change from cycle to cycle and will be analyzed for each fuel cycle. The OLMCPR is established for the limiting event and documented in the TS Section 5.6.3, COLR (a)(2) in accordance with TS 3.2.2.

15.2.2.2 Generator Load Rejection with Turbine Bypass System

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

15.2.2.2.1 Summary of Technical Information

EC: Grid electrical disturbances could cause loss of generator load. To avoid damage to the turbine generator from overspeed, the TCVs are designed and enabled to close very rapidly. TCV closure would increase vessel pressure, but the opening of the steam bypass valves prevents overpower and overpressurization. If there is no other failure, the steam bypass system has the capacity to discharge the entire steam flow at full power. The SCRRI/SRI rod system will then insert control rods to lower reactor power. 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 between the fast closing of the TCVs and opening of the turbine bypass valves (TBVs), there is a sharp pressure pulse accompanied by a power generation pulse that lasts slightly over 1 second. High neutron flux can cause a scram signal; however, the reactor is conservatively assumed not to scram. At the same time, the SCRRI/SRI system activates and the first SRI occurs at 15 seconds, followed by the second at 30 seconds, and so on until the sixth insertion occurs at 75 seconds. SCRRI insertion is complete at 110 seconds. At 200 seconds, the power level reaches 60 percent of rated power. No specific SCRRI group was assigned and the SCRRI results were not used to show acceptable CPR results. However, SCRRI/SRI rod patterns depend on cycle loading, may affect the ratio ΔCPR/ICPR, and are potentially limiting for the OLMCPR; therefore, their characteristics and requirements are documented in the COLR in accordance with the TS.

ICL: Analysis of the same transient for the ICL yields similar results for the transient characteristics (i.e., peak power, pressure, and feedwater flow). In this case, the transient power stabilizes at about 300 seconds at 40 percent of rated power, with feedwater and steam flows at 30 percent of rated. The peak power exceeds the high thermal flux and the high neutron flux setpoints but the reactor is conservatively assumed not to scram. The MCPR remains above the OLMCPR, and the pressure and core coverage are well above the L2 level

FWTOD: NEDO-33338 presents the transient for the SP2 state point and shows results similar to those described above for the EC and the ICL. The power peak exceeds the high thermal flux scram setpoint (at 100 percent of the ESBWR rated power) but remains below the high neutron flux setpoint. At about 300 seconds into the transient, total power stabilizes at 40 percent (of ESBWR rated power) with feedwater and steam flow at 30 percent of rated steam flow. The reactor power peak (as in the EC and ICL cases) exceeded the high thermal flux scram setpoint.

15.2.2.2.2 Technical Evaluation

EC: The main objective of this AOO is to ensure that the system isolates the turbine generator unit 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/SRI system to reduce power and limit MCPR values. Overpressurization is prevented by the full-flow turbine bypass capacity in the ESBWR.

Assuming no other equipment failure, total power will peak at about 0.6 seconds, and the feedwater flow will peak at about 35 seconds at 140 percent. SCRRI/SRI rod insertion will stabilize power at about 60 percent with 45 percent feedwater and steam flow.

If the TBVs operate as designed, no vessel overpressurization will occur, and RPV pressure will actually decrease. Rod insertion counterbalances the void reactivity increase from void collapse and the small reactivity increase resulting from fuel temperature. The MCPR stays above the OLMCPR, and reactor operation stabilizes at about 170 seconds.

Within a second after initiation of this transient, during the transition from closing of the TCVs to opening of the TBVs, the core experiences a short pressurization and void collapse, which results in a power spike to about 130 percent of rated power for about 1 second. The power peak is higher than the thermal flux and the thermal flux setpoints, but it is assumed that the reactor does not scram.

The staff questioned the energy deposition and the potential of reaching the cladding strain limits (RAI 15.2-2 S01 and RAI 15.3-11 S01), (Note: the staff requested this information for several transients as the review was progressing. GEH responded collectively in MFN 07-641) GEH responded to questions regarding energy deposition in the fuel during high and fast power peaks not addressed in the DCD. Generator load rejection with bypass failure (GLRBF) is the most limiting such transient. In its response, GEH stated that if the normalized nodal power density (NNPD) of any node in the core during a transient is bounded by the generic NNPD used to define the fast transient limits, then the thermal-mechanical limits are not threatened.

In the applicant’s analyses, the NNPD for the GLRBF was increased by a factor of 1.5 and then compared to the generic GE14 fuel NNPD. The GE14 fuel bounds by a considerable margin the results of the NNPD for the GLRBF including the conservative factor of 1.5. In addition, the GE14E fuel which is used for the ESBWR bounds the GE14 NNPD. Therefore, all AOO transients in DCD Section 15.2 and IEs in Section 15.3 are assured for thermal-mechanical integrity. This includes cladding strain, fuel center melt, and maximum linear heat generation rate (MLHGR).

<u>ICL</u>: The core response is very similar to that of the EC (i.e., the power spike and duration are comparable and the reactor resumes stable operation at 40 percent power at 30 percent feedwater and steam flow). The MCPR remains above the OLMCPR. The acceptance criteria are satisfied in that the MCPR, clad strain, and vessel pressure and water level are within the acceptance criteria; therefore, the transient results are acceptable.

<u>FWTOD</u>: The staff reviewed this transient in the context of the NEDO-33338 and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.2.3 Conclusion

The comparison of the initial core and the EC analyses results indicates that the EC analysis at SP0 is limiting in the generator load rejection transients.

As discussed above, the load rejection with turbine bypass transient is well within the acceptance criteria in Section 15.2.1.1.1. The SCRRI/SRI control rod patterns must be recalculated for each reload because they are fuel-loading dependent. Overpressurization and high-energy deposition are avoided because of the bypass and fast response of the SB&PC system; therefore, the results meet the acceptance criteria.

15.2.2.2.4 Post-COL Activity

This event is potentially limiting with respect to the OLMCPR, because of the effect of cycle-to-cycle changes to the SCRRI/SRI rod pattern on ΔCPR/ICPR.

This event is analyzed for each fuel cycle with the limiting SCRRI/SRI rod pattern which changes from cycle to cycle. The SCRRI/SRI requirements are documented in the COLR in accordance with the TS. The OLMCPR is established for the limiting event and documented in the TS Section 5.6.3, COLR (a)(2) in accordance with TS 3.2.2.

15.2.2.3 Generator Load Rejection with a Single Failure in the Turbine Bypass System

The staff used the acceptance criteria in Section 15.1.1.1 in evaluating this AOO.

15.2.2.3.1 Summary of Technical Information

<u>EC</u>: For this AOO, the system and the plant instrumentation responses are similar to that for the generator load rejection with turbine bypass. When the instrumentation senses generator load rejection, the SB&PC system signals closure of the TCVs and opening of the TBVs. However, in this case, the analysis assumes a single failure in the turbine bypass system. For conservatism, the bypass capacity is assumed to be at 50 percent. DCD, Tier 2, Figure 15.2-5, shows core response and core parameter variation as a function of time. Table 15.2-9 lists the sequence of events.

The calculations are based on the assumption that the SB&PC system will signal the bypass valves to initiate opening at 0.02 seconds into the transient and the TCVs will be closed at 0.08 seconds. At 0.15 seconds, the system will sense inadequate bypass, and the plant is scrammed. Control rod insertion initiates at 0.35 seconds. In this AOO, it is assumed that 50 percent of the bypass capacity has failed.

The calculated results are shown in Figure 15.2-5. For a short time (less than 1 second), steam flow decreases because of limited bypass, which increases pressure, neutron moderation, and core power. The power peak lasts less than a second; feedwater flow increases to about 140 percent of normal (due to void collapse) and stabilizes at about 60 percent in about 50 seconds. The vessel dome pressure peaks at about 1,130 pounds per square inch absolute about 2 seconds into the transient. Peak pressure is below the SRV lift setting. The MCPR remains well above the OLMCPR. This event is potentially limiting with respect to the OLMCPR; therefore, it must be analyzed for each cycle loading and included in the COLR. Because of the void collapse and reactor scram, the water level reaches L3 at about 2.82 seconds. No operator action is required to mitigate this transient.

ICL: The results of the ICL (for the same AOO with the same initial assumptions) are almost identical to the EC. However, the MCPR value remains well above the OLMCPR (Figure 2.3-5g in NEDO-33337, Revision 1) peak pressure is below the SRV setpoint. Because of the void collapse, the reactor vessel water level reaches L3 at about 2.7 seconds. Scram initiates at .15 seconds.

FWTOD: This AOO is similar to the EC transient. The value of MCPR is well above the OLMCPR, and the peak pressure is significantly below the SRV setpoint. Scram initiates at 45 seconds into the transient, and the L3 level is reached at 3.17 seconds, but the scram level is not exceeded

15.2.2.3.2 Technical Evaluation

EC: 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, the reactor scrams and control rod insertion begins at about .40 seconds after transient initiation. The resulting pressure and thermal power pulse is less than a second in duration. Should high pressure compress the water to the L2 level for 10 seconds or more, the CRD high-pressure makeup injection will activate. Should the low-level signal remain for 30 seconds, the MSIV and IC will activate.

The vessel pressure remains within acceptable limits, the MCPR remains above the OLMCPR, and no fuel rods are in boiling transition; therefore, the regulatory acceptance criteria are met.

The results provided in DCD Figure 15.2-5(a) show a high and narrow power peak of less than a second's duration. After the initiation of this transient during closing of the TCVs and opening of the TBVs, the core experiences a short pressure pulse resulting in a power spike to about 190 percent of rated power for less than 1.0 second. GEH did not calculate energy deposition to ensure acceptable cladding integrity and fuel cladding interaction. The staff questioned GEH's lack of consideration for energy deposition. In response to RAI 15.2-2 S01, GEH compared the GE14E fuel design limits to the energy deposition in this transient to demonstrate that the energy deposition is almost insignificant. Based on the applicant's response, RAI 15.2-2 S01 is resolved. Staff evaluation of RAI 15.2-2S01 is included in Section 15.2.2.2 of this report.

ICL: The same method (code) and the same assumptions are used as in the EC analysis. The results of the analyses are similar; however, the calculated power peak at the initiation of the transient is about 10 percent higher than in the EC case. The TCVs close at about .08 seconds, and at 2 seconds the system senses inadequate bypass and initiates rod insertion and reactor scram. The ensuing power peak lasts less than 0.2 seconds. The resulting MCPR is higher than the OLMCPR.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.3.3 Conclusion

The comparison of the results of the initial core and the EC analyses indicates that the EC analysis at SP0 is the most limiting for generator load rejection with single failure in the turbine bypass system.

As discussed above, load rejection with a single failure in the turbine bypass system is well within the acceptance criteria in Section 15.2.1.1.1. Overpressurization and high-energy deposition are avoided because of the available (50 percent) steam bypass and the fast response of the SB&PC system. The results are within the bounds of the acceptance criteria.

In response to RAI 15.2-2 S01, GEH compared the GE14E fuel design limits to the energy deposition in this transient to demonstrate that the energy deposition is almost insignificant. Based on the applicant's response, RAI 15.2-2 S01 is resolved. Staff evaluation of RAI 15.2-2S01 is included in Section 15.2.2.2 of this report.

15.2.2.3.4   Post-COL Activity

This event is analyzed for each fuel cycle with the limiting SCRRI/SRI rod pattern which changes from cycle to cycle. The SCRRI/SRI requirements are documented in the COLR in accordance with the TS. The OLMCPR is established for the limiting event and documented in the TS Section 5.6.3, COLR (a)(2) in accordance with TS 3.2.2.

15.2.2.4   Turbine Trip with Turbine Bypass

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

15.2.2.4.1 Summary of Technical Information

EC: A variety of causes, such as vibrations, low condenser vacuum, or loss of turbine control fluid pressure, can initiate turbine trip. After turbine trip activation, the SB&PC system will initiate opening of the bypass valves in 0.02 seconds.

At 0.10 seconds, the turbine stop valves (TSVs) are closed, and at 0.20 seconds, the SRI initiates fast rod insertion to limit core power so as to avoid reactor scram and protect the MCPR limits. At 1.5 seconds, SRI initiates insertion and the first group inserts. The second, third, fourth, fifth, and sixth groups insert at 16.5, 31.5, 46.5, 61.5, and 76.5 seconds, respectively. At 121 seconds, the reactor attains a steady state at about 60 percent of normal power with about 45 percent of normal feedwater flow. Peak feedwater flow occurs at 25–35 seconds at about 135 percent of normal. Vessel pressure shows a very small increase and falls to about 94 percent of the normal operating value. At the new steady state, about 45 percent of steam flow removes about 60 percent of power. The value of the MCPR remains well above the SLMCPR. In DCD, Tier 2, Table 15.2-10 and Figure 15.2-6 summarize the results of the calculation. Reactor scram is not activated, and no operator action is required to mitigate this transient. The calculation results are within the range of the acceptance criteria.

ICL: The results of this AOO are similar to the EC transient results. Both cases are based on the same assumptions regarding bypass availability. SCRRI/SRI insertion lowers reactor power and avoids reactor scram. Power stabilizes at about 40 percent of rated with 30 percent of normal flow. Vessel pressure rises slightly, and the MCPR stays above the SLMCPR. The results are within the range of the acceptance criteria.

FWTOD: NEDO-33338 does not explicitly analyze this transient but explained that this transient is bounded by the generator load rejection with a single failure in the turbine bypass system.

15.2.2.4.2 Technical Evaluation

EC: 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 brief increase in feedwater 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 feedwater flow of about 45 percent. The MCPR remains well above its designated safety limit. This transient is very similar to the load rejection with turbine bypass. The vessel water level remains well above the L3 level, there is no scram, and the reactor attains a stable state. The acceptance criteria are met.

ICL: The transient is similar to the EC transient except that the core power settles at a lower power of about 40 percent of normal with 35 percent of flow. The MCPR value is above the OLMCPR at about 1.0 second. The vessel water level remains well above the L3 level, there is no scram, and the reactor attains a stable state. The acceptance criteria are met.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 and finds that it is bounded by the IICI events analyzed at the SP2 state point.

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 OLMCPR, and the reactor assumes a lower power stable state. Therefore, the results of this transient meet the acceptance criteria, and the transient results are acceptable.

There is no reactor scram. This event is similar to the generator load rejection with turbine bypass and credits the SRI system.

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

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

15.2.2.5.1 Summary of Technical Information

EC: A variety of causes, such as vibrations, low condenser vacuum, and loss of turbine control fluid pressure, can initiate a turbine trip. Upon activation of a turbine trip signal, the SB&PC system will open the TBVs 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 TBV failure to lose 50 percent of the bypass capacity, this is a conservative assumption. The shortfall in bypass capacity creates a pressure pulse. The pressure increase causes void collapse, increased moderation and rapid increase in neutron power that results in a scram signal. The power pulse has a width of about one third of a second at half-maximum and peaks at over 140 percent of rated power. A small vessel pressure peak occurs at about 2.0 seconds into the transient, which remains well below the SRV setpoint. The TSV closure initiates reactor scram at 0.35 seconds. 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 50 seconds. Because the reactor is shut down with the vessel water level near L3, which is 12.5 meters (41 ft) above the TAF, the staff concludes that the reactor is in a stable condition. The MCPR value remains higher than the operating limit OLMCPR. DCD, Tier 2, Figure 15.2-7, and DCD, Tier 2, Table 15.2-11, show the calculated results.

ICL: For the initial core analysis, this transient is similar to the EC transient; for example, the power peaks at the same value of 150 percent of rated power. In both cases, scram initiates at 37 seconds, and the reactor is shut down in about 2 seconds. The MCPR value is well above the OLMCPR. The core remains covered at normal pressure and no rods are in boiling transition. This transient meets the AOO acceptance criteria.

FWTOD: The turbine trip with single bypass failure has not been explicitly analyzed in NEDO-33338 because it is shown to be bounded by the IICI event.

15.2.2.5.2 Technical Evaluation

EC: Following turbine trip due to inadequate bypass flow, the RCS pressure peaks, causing void collapse, increased moderation, and the creation of a power peak. The RPS initiates scram, and the reactor total power falls to about 5 percent in less than 3 seconds with simultaneous closure of the TCVs. The vessel pressure peaks at 2.0 seconds but remains well below the SRV setpoint and is decreasing. The MCPR stays well above the OLMCPR and is increasing very quickly. Therefore; the reactor enters a safe-shutdown state, in terms of pressure and MCPR. The results provided in DCD, Tier 2, Figure 15.2-7, show a high and narrow power peak about one-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 that GEH explain why it did not consider fuel energy deposition. In the response to RAI 15.2-2, GEH demonstrated that the ESBWR fuel (GE14E) has the capacity to accommodate a greater amount of thermal energy than that deposited in this transient; therefore, the issue is resolved. (See also related discussion in Section 15.2.1.1.2)

Because the reactor is shut down, and the vessel water is at 12 meters above TAF (level L3), the staff finds that the reactor is stable. This event is similar to the generator load rejection with a single failure in the turbine bypass system and does not need to be evaluated with each fuel cycle.

ICL: This transient presented in NEDO-33337 is similar to the EC analysis where the feedwater flow is at 150 percent of normal at 30 seconds, the water level is stable, and the vessel pressure is decreasing. The MCPR value is above the OLMCPR.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.5.3 Conclusion

The comparison of the results of the initial core and the EC analyses indicates that the EC at the SP0 state point is the most limiting AOO in the turbine trip with single failure in the turbine bypass system.

The MCPR value for the EC transient is well above the OLMCPR, and pressure and water level are well within normal limits. The staff finds that the FWTOD transient is bounded by other transients. Therefore, the results of this transient meet the acceptance criteria and are acceptable.

15.2.2.6 Closure of One Main Steam Isolation Valve (MSIV)

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

15.2.2.6.1 Summary of Technical Information

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

MSIV closure lasts about 3.0 seconds. At the initiation of the closing process, reactor pressure rises, suppressing the core void, which increases moderation and power, but the turbine bypass opens at 2.8 seconds to limit pressure and power increase. Neutronic power peaks at 2.0 seconds but the thermal neutron and the thermal flux do not reach the scram level. At 3.0 seconds the MSIV is closed. Total power assumes a new steady-state at about 101 percent of normal power, with 101 percent of feedwater flow and turbine steam flow at 93 percent of normal. The new steady state is reached at 40 seconds into the transient. There is a small increase in pressure vessel pressure but the MCPR remains well above the OLMCPR.

ICL: This is similar to the EC transient with an MCPR value of 1.38, a small increase in pressure vessel pressure, power level at 101 percent of the rated level, steam flow at about 93 percent, and a power peak lower than the high thermal flux scram setpoint.

FWTOD: The staff has reviewed this transient in the context of NEDO-33338 and finds that it is bounded by IICI events analyzed at the SP2 state point.

15.2.2.6.2 Technical Evaluation

EC: Under full-power conditions, the MSIV takes 3 seconds to close. During closure, power increases due to void collapse and increased moderation, but fuel temperature reactivity feedback will offset the increase, and total reactivity change returns to zero. Turbine bypass opens at about 2.8 seconds to moderate the pressure and power increase. The calculated results show that the transient has little if any effect on vessel pressure and the MCPR will remain well above the OLMCPR; thus, the acceptance criteria are met. This transient is bounded by the all-MSIV-closure transient, discussed in Section 15.2.2.7 of this report; therefore, it does not need to be reanalyzed for each loading.

ICL: The one MSIV closure transient is almost identical to the EC one MSIV closure transient. This holds true for the power peak and the pressure transient. The MCPR remains above the OLMCPR.

FWTOD: The staff has reviewed this transient in the context of the NEDO-33338 review and finds that it is bounded by IICI events analyzed at the SP2 state point.

15.2.2.6.3 Conclusion

Closure of one MSIV is a minor perturbation in reactor operation without a serious challenge from overpressure, MCPR, the power peak, or the vessel water level. 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 OLMCPR, and the transient assumes a stable steady state with the core fully covered and does not lead to another transient. Therefore, the EC and ICL results of the analysis of this transient meet the acceptance criteria.

Staff review of this transient for FWTOD finds that it is a mild transient bounded by other events that have been found acceptable.

15.2.2.7 Closure of All Main Steam Isolation Valves

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

15.2.2.7.1 Summary of Technical Information

EC: 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, low water level, or manual action will activate closure of all MSIVs. Total time for completion of MSIV closure is 3.0 seconds. In DCD, Tier 2, Figure 15.2-9 and Table 15.2-13 show the calculated results for the evolution of this transient.

MSIV closure initiates a reactor scram on high neutron flux. The same signal also initiates IC operation, which prevents lifting of the SRVs by lowering the RCS temperature and pressure.

The analyses in the DCD conservatively assumed MSIV closure to be completed in 3 seconds, the shortest time in the MSIV closure range, which would thus cause the highest pressure pulse. Vessel pressure reaches a maximum in 4.3 seconds at 7.8 megapascals (MPa) (1,131 psig), while the lowest SRV opening setpoint is at 8.62 MPa. Control rod insertion is completed within 4.0 seconds, and the MCPR reaches the lowest value at 1.25 seconds, which is well above the OLMCPR. The feedwater flow decreases to about 72 percent of normal at about 4.0 seconds because of increased RCS pressure after void collapse, while core flow increases to about 140 percent of normal. At 20.1 seconds, the L2 vessel water level is reached; at 30.1 seconds, the CRD high-pressure injection is activated; and at 31.82 seconds, the IC valves are fully open and liquid flow from the ICs initiates at about 17 seconds. The reactor water level reaches 11 meters above TAF in about 20 seconds and keeps rising.

<u>ICL</u>: This transient is similar to the transient in the EC, but the MCPR value is higher. The reactor becomes subcritical in less than a second, and void collapse does not induce a power peak. Pressure will increase to the high-pressure scram setpoint at about 3 seconds but the reactor is already subcritical. IC liquid flow initiates again at about 17 seconds, and the vessel water level starts recovering at about 22 seconds. At that time, the feedwater flow is still above 140 percent of normal.

<u>FWTOD</u>: The staff review of this transient finds that it is bounded by the IICI event that has been found acceptable.

15.2.2.7.2 Technical Evaluation

<u>EC</u>: A variety of circumstances will result in an MSIV closure signal, which also activates turbine bypass, scram, IC initiation, and CRD injection. Assuming that the RPS operates as designed, rod insertion will dominate core reactivity. Within 1.0 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, operator intervention is not needed. The MCPR remains well above the OLMCPR, and the SRVs are not challenged; therefore, acceptance criteria are met. Because the reactor shuts down within a second and bubble collapse and core pressurization do not create a power spike, this transient need not be analyzed for different core loadings because no power spike is created that would depend on fuel loading.

<u>ICL</u>: The transient analyses results are almost identical to the EC results, except that the MCPR value remains above the OLMCPR.

FWTOD: The staff has reviewed this transient in the context of the NEDO-33338 review, and the staff finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.7.3 Conclusion

The comparison of the initial core and the EC analyses indicates that the results of the analyses are similar.

All MSIV closure is a fast-evolving transient where rod insertion, IC initiation, and CRD activation proceed concurrently. 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 MCPR is well above the OLMCPR, and the reactor is shut down in a stable condition with the core covered. Thus, the results of this event meet the acceptance criteria.

15.2.2.8 Loss of Condenser Vacuum

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

15.2.2.8.1 Summary of Technical Information

EC: Failure or isolation of the steam jet air injectors and loss of one or more condenser circulating water pumps are common causes of loss of condenser vacuum. Sensing the loss of condenser vacuum, the RPS will initiate turbine trip and reactor scram. Turbine bypass will activate (in 0.02 seconds) to regulate pressure and close the TSVs (in 0.10 seconds). Reactor scram initiation will occur at 0.20 seconds, turbine bypass closure at 6.0 seconds, closure of the MSIVs at 8.0 seconds, and IC activation at 9.8 seconds from the MSIV closure signal. When the vessel water level reaches L2, the high pressure control rod drive (CRD) injection initiates (at 24.8 seconds) to control and restore the water level. In DCD, Tier 2, Figure 15.2-10 and Tables 15.2-14 and 15.2-15 present the calculated results.

Control rod insertion dominates reactivity response; thus, pressure increase has no effect on power level via void-collapse and increased moderation. For the first 10 seconds, steam flow increases along with feedwater flow. Reactor water level reaches a minimum in 20 seconds at about 6 feet above TAF that is below the L2 level. 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 MCPR remains at operating or higher level values. For the initial core, this transient analysis result is almost identical to the EC analysis result.

ICL: Because the reactor shuts down, the results of this AOO are independent of fuel loading, and the results are the same as in the EC analysis.

FWTOD: This event is not analyzed explicitly for FWTOD in NEDO-33338, because it is a mild transient and is not a limiting event. In addition, the reactor is shut down and this transient does not depend on fuel loading.

15.2.2.8.2 Technical Evaluation

EC: Failure or isolation of the steam jet air injectors or loss of one or more condenser circulating water pumps will result in loss of condenser vacuum. When loss of condenser vacuum is sensed simultaneously, the TBVs begin to open to regulate RCS pressure. Reactor scram initiates, and the main TBVs open and initiate MSIV closure, which: elevates vessel pressure, collapses voids, and lowers the reactor water level. The HPCRD injection activates to restore the water level. The MCPR remains well above the OLMCPR, and the high RCS pressure does not challenge the SRVs.

ICL: The ICL transient results are almost identical to those of the EC analysis. The basic feature is the fast response of the RPS and rod insertion that shuts the reactor down. The sequence does not depend on fuel loading.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 review and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.8.3 Conclusion

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

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

There are no specific acceptance criteria for this case because the event is not a transient that involves the reactor.

15.2.2.9.1 Summary of Technical Information

EC, ICL, and FWTOD: This is not a specific AOO but the description of a redundant cooling system failure, therefore, the following information and analyses apply to all three modes of reactor operation. In the ESBWR, the RWCU/SDC system is not a safety system. Nevertheless, it can provide high- and low-pressure water cooling for the core. The system consists of two trains with the necessary piping, heat exchangers, power supply, and instrumentation. 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 feedwater line. Each train has an offsite power supply, but if power is lost, each train has its own independent diesel power supply. In this manner, the system is single-failure proof.

In the event that both trains are lost, the ICS is able to maintain the reactor in stable condition for 72 hours. During refueling, the ICs are unavailable. The GDCS is available to provide extended decay heat removal for at least 72 hours. After 72 hours, the suppression pool can drain into the vessel via the equalization valves.

Although the RWCU/SDC system is not safety-related, sufficient redundancy exists that the system can be relied upon to provide decay heat removal (closed or open vessel) for extended time periods.

15.2.2.9.2 Technical Evaluation

EC, ICL, and FWTOD: For the shutdown cooling function, each train has its own suction line from the RPV (unlike the current reactors) and returns to the feedwater line. Thus, each of the two RWCU/SDC trains is completely independent of the other. If the single active failure criterion is applied to the RWCU/SDC system, one of the RWCU/SDC trains could be inoperable. However, the operable RWCU/SDC train could achieve cold-shutdown conditions within 36 hours after reactor shutdown.

The RWCU/SDC system, in combination with the ICs, the GDCS, and the water inventory in the suppression pool, is able to provide cooling water for extended periods of time with a closed vessel or under refueling conditions.

This evaluation does not refer to AOO rather; it demonstrates the availability and redundancy of systems able to supply adequate core cooling water for extended periods of time.

15.2.2.9.3 Conclusion

As indicated in the preceding description, the issue in the loss of the RWCU/SDC system is the availability of redundant systems to provide cooling water for removing decay heat after shutdown and, if required, to bring the reactor to a cold-shutdown condition.

There are multiple redundancies for shutdown cooling with the RPV closed or open. The two RWCU/SDC trains have independent supply, discharge, and redundant power supply. In addition, the ICs could provide core cooling for an additional 72 hours. Under refueling conditions (or with an open vessel) with the RWCU/SDC trains unavailable, the FAPCS could provide cooling. In addition, the GDCS is available to provide core cooling for at least 72 hours. In summary, a fourfold redundancy exists (with either open or closed vessel head) in the ability to supply cooling water for at least 72 hours. Therefore, the possibility of core damage due to RWCU/SDC system malfunction is extremely remote, and the design is acceptable.

15.2.3   Reactivity and Power Distribution Anomalies

15.2.3.1   Control Rod Withdrawal Error during Startup

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

15.2.3.1.1 Summary of Technical Information

(Transients in this Section are independent of fuel loading, and thus there is no distinction between EC, ICL, and FWTOD analyses.)

In Section 15.2.3.1, the DCD assumes that during startup, a gang of control rods (or a single rod) is inadvertently withdrawn continuously due to procedural error or a malfunction in the automated rod movement control system. This assumes that the reactor is critical with power less than the low-power setpoint. The RC&IS has an RWM to prevent any out-of-sequence rod withdrawal. Also RC&IS has restrictions on ganged rod withdrawal sequence such that, if the restrictions are violated, the RC&IS initiates a rod block. The startup range neutron monitor (SRNM) has a period-based scram for periods shorter than 10 seconds.

A typical sequence of events in this AOO begins with the operator withdrawing a rod-gang continuously during startup. No operator action is required to terminate the transient. DCD Section 15.3.8, "RWE During Start-up with Failure of Control Rod Block," presents a bounding analysis which does not credit the rod block action. Review of DCD Section 15.3.8 indicates that if the SRNM rod block is not credited, the power spike that follows rod gang withdrawal (either from zero to 15 percent power) will result in a fuel enthalpy increase that is within the AOO acceptance criteria.

15.2.3.1.2 Technical Evaluation

<u>EC, ICL, and FWTOD</u>: In RAI 15.2-10 and its supplement RAI 15.2-10 S01, the staff requested the applicant's description and the plant's response to a reactivity and power distribution anomaly. In the initial response, the applicant addressed only the electronic part of the system. The staff noted that electrical as well as mechanical causes of control rod malfunction should be included.

In the revised response to RAI 15.2-10 S01, the applicant provided operational information regarding the RC&IS controls FMCRDs employed in the electrical movement of control rods. Mechanical failure of a single relay will not cause an inadvertent RWE. Additionally, failure of the mechanical contact of a switch will not cause RWE because they are single failure proof with respect to RWE. GEH extended the argument to the electronic equipment also being 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 unlikely and can result only from multiple failures. Should an RWE take place a period-based rod block for SRNM occurs for periods shorter than 20 seconds and a scram for periods shorter than 10 seconds. The response references DCD, Tier 2, Sections 15A and 7.7.2. The NRC staff accepts this response because both the mechanical and the electronic elements are single failure proof and there exist additional means to block rod withdrawal or scram the reactor. Based on the applicant's response, RAI 15.2-10 is resolved.

In addition, DCD Section 15.3.8 provides analyses for the case in which the SRNM rod block is not credited with stopping rod or rod-gang withdrawal. The analyses indicate that the energy deposition in the fuel (with conservative adiabatic assumptions) will meet the AOO acceptance criteria; therefore the results of this transient analysis are acceptable. Neither reactor scram nor operator action is required to mitigate this AOO.

ICL and FWTOD: The startup RWE transient and the reactor response are independent of fuel loading; as indicated above, for this AOO, there is no differentiation of the EC analyses.

15.2.3.1.3 Conclusion

The comparison of the initial core and the EC analyses results indicates that the analyses are similar, which is to be expected because the actions involved are independent of fuel loading.

The staff agrees that DCD, Tier 2, Section 15.2.3, supports the conclusion that transients that may result from an inadvertent rod or rod-gang withdrawal from a critical reactor (or reactor power up to 15 percent of nominal power) meet the AOO transient acceptance criteria. Reactor scram could be invoked. Operator action is not required to mitigate this AOO.

15.2.3.2 Control Rod Withdrawal Error during Power Operation

15.2.3.2.1 Summary of Technical Information

(This section is independent of fuel loading; thus, there is no distinction between EC, ICL, and FWTOD.)

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

During power operation, the ATLM performs the rod block monitoring function as a dual channel subsystem of the RC&IS. One channel monitors the MCPR, and the other monitors the MLHGR. The rod block algorithms for both channels are based on actual online core thermal data to protect the MCPR and MLHGR setpoints. Regardless of the origin of the rod withdrawal malfunction, the activation of rod blocks protects the thermal limits. Therefore, an inadvertent rod (or rod gang) withdrawal will be terminated without operator intervention or reactor scram.

The power operation RWE transient and the reactor response are independent of fuel loading; therefore, for this transient, there is no differentiation of the EC from the ICL and FWTOD cores.

15.2.3.2.2 Technical Evaluation

IC, ICL, and FWTOD: The ATLM continuously monitors the MCPR and MLHGR limits and intervenes to prevent violation of either limit due to a rod (or rod-gang) withdrawal error. Because there are two channels, the signal is single-failure proof, and no reactor operator action is required and no scram signal will be generated.

DCD Section 15.3.8 presents a bounding analysis in which the rod block action is not credited. Review of Section 15.3.8 indicates that if the SRNM rod block is not credited, the power spike that follows rod-gang withdrawal (either from zero or 15 percent power) will increase the fuel enthalpy and the increase will be within the AOO acceptance criteria.

15.2.3.2.3 Conclusion

The ATLM system is single-failure proof and protects the fuel. Regardless of the cause of rod withdrawal, the ATLM will intervene to stop rod withdrawal and protect the thermal limits. This action does not require operator intervention or reactor scram therefore, the result is acceptable. In addition, the probability of failure of the ATLM system that would result in an inadvertent rod (or rod-gang) withdrawal is extremely small.

15.2.4 Increase in Reactor Coolant Inventory

15.2.4.1 Inadvertent Isolation Condenser Initiation

15.2.4.1.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

<u>EC and ICL</u>: In its analysis of IICI, the applicant assumed that all IC units are activated; therefore, this is a bounding case. While IC flow initiates at about 10 seconds from activation, the system establishes full IC flow in about 32 seconds. Due to cold-water injection, core power increases due to increased moderation, reaches a maximum at about 50 seconds, and returns to normal level at about 200 seconds. Feedwater flow decreases accordingly to keep the vessel water approximately at the same level. Accounting for the IC liquid flow, the total water injection is about equal to full feedwater flow. MCPR 1.25 is reached at about 150 seconds into the transient. The analysis assumes that the system operates without additional failures. DCD, Tier 2, Figure 15.2-11, shows the calculated results, and DCD, Tier 2, Table 15.2-17, depicts the sequence of events.

With a power peak at about 110 percent and total water injection (feedwater plus IC liquid) at about normal feedwater flow, the MCPR is well above the OLMCPR and occurs at 125 seconds into the transient. Vessel pressure stays at the normal operating level. The power increase stabilizes at normal power level at about 300 seconds. There is no power scram, and no operator intervention is needed to mitigate this transient.

<u>FWTOD</u>: NEDO-33338 presents analytical results for this transient. The calculation is for SP2 conditions, and the transient evolves similarly to that in the EC and ICL analyses. The MCPR value is above the OLMCPR and occurs 130 seconds into the transient.

15.2.4.1.2 Technical Evaluation

<u>EC and ICL</u>: Assuming that all four IC units are activated, the transient represents a bounding case. The only reasonable assumption for the simultaneous initiation of all ICs is inadvertent manual operator action. Cold water injection into the vessel increases water density, core moderation, and core power. The transient proceeds relatively slowly with a gradual increase in the thermal power and corresponding variation in the feedwater flow. The calculated results

indicate that, in about 300 seconds, the reactor attains equilibrium operation at normal power, with feedwater flow at about 90 percent of normal. The MCPR remains well above the OLMCPR. Vessel pressure stabilizes at a slightly lower level than normal. The core remains fully covered and stable. No scram signal or operator action is required in this transient.

<u>FWTOD</u>: NEDO-33338 presents analyses of the inadvertent IC activation for operation SP2 and shows this case to be limiting. Because of the lower power level and the power increase due to cold injection, the transient peak power is less than 100 percent of rated power. At the initiation of the transient, the IC liquid contribution complements the feedwater flow. Vessel pressure shows no significant variation, and water level stays over 12 meters above TAF. The MCPR value remains above the OLMCPR. NEDO-33338, Figure 2.3-5 shows the evolution of the transient. The calculated ΔCPR/ICPR of 0.12 for EC at the SP2 state point is indicated as the most limiting of the AOOs.

DCD Section 15.2.4.1.3 states that this transient is potentially limiting with respect to the OLMCPR; therefore, this transient should be analyzed for each cycle and for FWTOD.

15.2.4.1.3 Conclusion

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 acceptance limits, the MCPR stays well above the OLMCPR, the core remains fully covered, and the reactor returns to a stable state. Therefore, the results of this AOO meet the acceptance criteria.

15.2.4.1.4 Post-COL Activity

This event is analyzed for each fuel cycle with the limiting SCRRI/SRI rod pattern which changes from cycle to cycle. The SCRRI/SRI requirements are documented in the COLR in accordance with the TS. The OLMCPR is established for the limiting event and documented in the TS Section 5.6.3, COLR (a)(2) in accordance with TS 3.2.2.

15.2.4.2 Runout of One Feedwater Pump

15.2.4.2.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.1 in evaluating this AOO.

<u>EC and ICL</u>: The EC and ICL transients are very similar in that the size and timing of the changes after a pump initiates runout are similar.

Three feedwater pumps are running continuously during normal operation. Feedwater pumps are motor driven with variable speed motors. A runout transient consists of one pump increasing speed (and feedwater 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. However, two credible single failures could lead to loss of one actuator for one feedwater pump with increasing flow. The analyses presented in DCD, Tier 2, Figure 15.2-13, and DCD, Tier 2, Tables 15.2-18 and 15.2-19, consider such a case.

When the system senses the increased flow, the feedwater controller will lower feedwater flow to the two operating pumps so that 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 perceptibly. Fuel temperature and void reactivity change in opposite directions, resulting in small changes in total reactivity compensated by small control rod movement. Feedwater flow changes equalize at about 40 seconds, and reactivity variations stabilize at about 100 seconds into the transient. The MCPR value is above the OLMCPR.

This transient has not been explicitly analyzed in the FWTOD because it is bounded by events that have been analyzed and found to be acceptable.

15.2.4.2.2 Technical Evaluation

EC, ICL, and FWTOD: This transient results in increased feedwater flow caused by a single feedwater pump runout. Feedwater controller action to reduce feedwater flow promptly compensates for increased feedwater, and the system achieves normal water level at about 40 seconds into the transient, according to the submitted analytical results.

The increase in feedwater causes brief reactivity changes, which are self-compensating as they produce limited variation in power. The transient does not initiate a scram, the MCPR remains well above the OLMCPR, and there is a barely perceptible variation in pressure. No operator action is required to mitigate this transient, and there is no scram. The analytical results meet the acceptance criteria.

15.2.4.2.3 Conclusion

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

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 staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

EC: Instrumentation failure, such as actuator or voter failure could cause inadvertent opening of a TBV or a TCV. Such failure is highly unlikely because the SB&PC system has a triplicate control configuration (see Section 15.2.4.2 of this report) so that no credible single failure can result in TCV or TBV failure.

Inadvertent operator action could cause a TCV or a TBV to open. DCD, Tier 2, Figure 15.2-14, shows the evolution of the transient, and DCD, Tier 2, Table 15.2-20, lists the sequence of events.

The calculated results indicate that, at the initiation of the transient, steam flow increases very briefly, which increases the void fraction, causing a corresponding dip in power and dome pressure. However, the lower pressure increases the feedwater flow, which promptly increases moderation, and power recovers at about 30 seconds into the transient. At this time, the turbine steam flow reduces to about 82 percent of normal, and the TCV flow remains at 15 percent. Regarding reactivity changes, void reactivity dips sharply within the first second of the transient, which is partially compensated for by fuel temperature and the increase in feedwater flow causing void collapse and total reactivity to return to critical at about 30 seconds into the transient. The vessel pressure remains almost unchanged from the normal operating value, the MCPR stays well above the OLMCPR, and the reactor assumes a stable condition while the fuel remains covered. No operator action is required to mitigate this event, and no scram signal is initiated.

ICL: For the initial core, this transient is very similar to the one for EC, but the reactivity oscillations are somewhat smaller. The vessel pressure is reduced somewhat, the MCPR remains well above the OLMCPR, and the reactor assumes a new stable steady state.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 and finds that it is bounded by IICI events analyzed at the SP2 state point.

15.2.5.1.2 Technical Evaluation

EC, ICL, and FWTOD: 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 without operator intervention or reactor scram. In DCD, Tier 2, Table 15.2-20 and Figure 15.2-14 demonstrate the evolution of the event. There is a small increase in steam flow after transient initiation, with a corresponding oscillation in feedwater flow and steady steam flow from the TCV or bypass flow. At about 30 seconds into the transient, the system establishes a new steady state. The vessel pressure is reduced by a small amount from the normal operating level, 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 TBV 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.

15.2.5.2 Loss of Non-Emergency AC Power to Station Auxiliaries

15.2.5.2.1 Summary of Technical Information

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, it is assumed that concurrent with load rejection, simultaneous loss of power occurs on the four power generation buses, which will cause the feedwater and condenser circulation pumps to be lost. Loss of the circulating water pumps results in loss of the condenser vacuum, which in turn causes turbine trip MSIV closing and reactor scram. The bypass valves will be initially available, but loss of the power buses will produce initiation signals for the ICs and HPCRD injection. With the loss of the station transformer, the station emergency diesel generators will activate to provide power to the CRD pumps. The CRD startup signal is generated when the wide range water level indication falls below the L2 level (for longer than 10 seconds). However, CRD injection is delayed by 145 seconds until the diesels are up in power. Water level is regained above the L2 level at about 800 seconds. The MCPR value remains above the OLMCPR. In summary, loss of the station auxiliary power will lead to reactor scram. In DCD, Tier 2, Table 15.2-21 shows the sequence of events, and Figure 15.2-15 depicts the time dependent variation of the reactor parameters.

The main condenser loss of vacuum signal has a time delay of 50 seconds. Upon loss of load, feedwater flow decreases briefly, followed by a very short power spike from increased moderation with the closing of the MSIVs. The power spike leads to reactor scram. About 100 seconds after initiation, IC water supply increases sharply, then levels off about 20 seconds later.

<u>FWTOD</u>: From the review of the EC and ICL cores, the staff concludes that this transient is not bounding. The critical success parameter is the water level, which for the SP1 is higher than the SP0 level; for SP2, the final water level may be slightly less than SP0.

15.2.5.2.2 Technical Evaluation

<u>EC and ICL</u>: The reactor pressure remains well below the AOO limit of 110 percent of the design value, the MCPR remains well above the OLMCPR, and the reactor is shut down. CRD high pressure injection controls the water level. CRD and IC injection ensure core cooling and core water level recovery. No operator action is required to mitigate this AOO and bring the reactor to a stable shutdown state. Since the core is covered, in a stable state, and cooled, the staff concludes that the acceptance criteria are met.

<u>FWTOD</u>: The staff reviewed this transient in the context of the NEDO-33338 review and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.5.2.3 Conclusion

Loss of all nonemergency ac power to station auxiliaries leads to turbine trip and reactor scram, with concurrent IC and HPCRD 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 MCPR remains well above the SLMCPR; therefore, the results of this transient meet the acceptance criteria.

15.2.5.3 Loss of All Feedwater Flow

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

15.2.5.3.1 Summary of Technical Information

EC: Loss of all feedwater flow could result from operator errors, pump failure, or reactor trip signals. In DCD, Tier 2, Revision 6, Table 15.2-22 lists the sequence of events, and Figure 15.2-16, shows the variation of reactor parameters as a function of time. If the feedwater pumps trip, the ensuing reduction of feedwater flow will initiate IC operation. At about 6 seconds into the transient, the feedwater flow decays to zero, the vessel water level drops to RPV Level 2 (it is assumed that initial water height is at level L), the HPCRD injection initiates at 20 seconds, the ICs reach full flow at 33 seconds and the MSIVs close at 40 seconds. At about 80 seconds, the water level recovers to about 13 meters (43 feet) above TAF, and the core is shut down and stable.

ICL and FWTOD: Neither is dependent on fuel loading; therefore, reactor response is the same as in the EC case. The transient is not bounding and has not been explicitly analyzed in the FWTOD.

15.2.5.3.2 Technical Evaluation

EC, ICL, and FWTOD: The RPS will scram the reactor and initiate ICs to ensure water level recovery. 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 1.31 (the designated OLMCPR value), 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 feedwater flow results in a fast reactor shutdown and simultaneous IC and CRD high-pressure injection activation. This transient does not violate any of the AOO acceptance criteria; therefore, the transient analysis is acceptable.

15.2.6 Conclusion of Anticipated Operational Occurrence Review

The staff concludes that the requirements of GDC 10, 13, 15, 17, 20, and 26 have been met. This conclusion is based on the following:

- The applicant meets the requirements of GDC 10 that the SAFDLs are not exceeded.
- The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation is available and that actuations of protection systems automatically occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.
- The applicant meets the requirements of GDC 15 that the design conditions of the RCPB are not exceeded.
- The applicant meets the requirements of GDC 17 and 26 by demonstrating that SAFDLs are not exceeded.
- The applicant meets the requirements of GDC 20 that the reactivity control systems are automatically initiated so that SAFDLs are not exceeded.
- In addition, the review identified IICI as the limiting AOO with respect to the MCPR.

15.3 Analysis of Infrequent Events

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

Initially, the applicant based some event analyses in DCD, Tier 2, Section 15.3, on a current OLMCPR and a bounding 1,000 damaged fuel rods. Initially, the TS did not include the SLMCPR, so the applicant needed to address these hypotheses. RAI 15.0-16 addresses this issue, and the staff's evaluation of the response appears in Section 15.1.1.1 of this report.

The staff reviewed DCD, Tier 2, Revision 1, Section 15.3, and found that the applicant had not provided a complete source term for the radiological consequence analysis for the IEs 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, by adding applicable information pertaining to radiological consequence analysis for those IEs listed in the DCD. GEH revised DCD Tables 15.3-13 and 15.3-16 and made the requested changes in the DCD. Based on the applicant's response, RAI 15.3-25 is resolved.

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 staff found that the applicant needs to revise DCD, Tier 2, Table 15.3-1, to show the results of all IEs. The applicant also needs to analyze the events described in Sections 15.3.7 to 15.3.12 and 15.3.14. GEH incorporated the requested changes in Revision 5 of the DCD. It modified Sections 15.3.9.3 through 15.3.9.5 and added Table 15.3-1b. The changes are responsive to the staff's request, and the issue is resolved. Therefore, based on the applicant's response, RAI 15.3-26 is resolved.

15.3.1 Loss of Feedwater Heating—Infrequent Event

15.3.1.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

<u>EC</u>: LOFWH with Failure of SCRRI/SRI

LOFWH can occur in at least two ways: (1) the steam extraction line to the heater is closed or (2) the feedwater flow bypasses the heater. LOFWH will decrease FWT and increase core water density, resulting in increased core neutron moderation, and increase in power. The DCD states that the ESBWR design is such that no single failure or operator error will cause LOFWH greater than 55.56 degrees C (100 degrees F). Normally, LOFWH and the associated temperature decrease will be detected by the ATLM and/or the DPS, either of which will activate the SCRRI/SRI to counter the positive reactivity insertion from cold-water injection and partial void reduction and thus avoid reactor scram. In this case, it is assumed that SCRRI/SRI insertion fails. This event is calculated at the simulated thermal power trip (STPT) scram setpoint. The maximum thermal power rises to 115.4 percent of the normal power level. DCD, Tier 2, Revision 6, Figure 15.3-1 and Table 15.3-1a show the calculated results as a function of time. The results indicate that the addition of void reactivity is counterbalanced by fuel temperature reactivity. The power level remains at 115.4 percent of normal power at the end of the calculation at 300 seconds. No operator action is required to mitigate this event. The MCPR value is higher than the OLMCPR.

The MCPR value with failure of SCRRI/SRI indicates that the number of fuel rods to enter boiling transition will be bounded by 1,000; therefore, the expected radiological consequences are within the acceptance limits. The estimated frequency of this event is less than 1.0×10^{-2} pry, which classifies it as an IE, as indicated in DCD, Tier 2, Section 15A.3.6.3.

<u>ICL</u>: This event is very similar to that for EC. A notable difference is that the reactivity control fraction is positive (maximum about 2 cents) for the EC case, while the ICL does not require reactivity compensation.

<u>FWTOD</u>: In the FWTOD at higher FWTs, the heating valves of feedwater heater No. 7 are open. Under those conditions, FWT minimum demand failure will result in closure of the No. 7 heater valves. The resulting temperature decrease could be higher than 55.56 degrees C (100 degrees F). The ATLM and/or the DPS will detect the FWT decrease and initiate

SCRRI/SRI insertion which is credited for this event. In the event of ATLM, DPS or SCRRI/SRI failure to insert the reactor will scram when the STPT setpoint is reached. No operator action is required to mitigate this event; however, at the end of the transient the operators must not permit reactor operation at elevated power levels.

Appendix A.3, "Infrequent Events," to NEDO-33338 presents the calculated results for the SP2 state point. The event is similar to the EC, except that the average fuel temperature reactivity component is higher than the void component and the control component is about 5 cents, which reflects the increased fuel average temperature. This event results in reactor scram.

15.3.1.2 Technical Evaluation

EC and ICL: This case involves two separate events. The first concerns the reference core where the maximum temperature decrease in the feedwater is 55.56 degrees C (100 degrees F), and the second is the FWTOD case where it is possible to have FWT differences greater than 55.56 degrees C (100 degrees F).

In the first case, EC maximum ΔT = 100˚F, the calculated results indicate that coolant pressure will remain within normal limits, power will rise to 115.4 percent of rated power, and the MCPR will stay above the SLMCPR. Operator action is not required to mitigate the event.

In the second case, FWTOD: maximum ΔT > 100˚ F, arises only during the FWTOD operation when heater No. 7 is in service. FWT controller failure (to minimum temperature demand) results in closure of the No. 7 heater steam heating valves and subsequent opening of the high-pressure feedwater heater bypass valves. The resulting decrease in FWT is potentially higher than 55.56 degrees C (100 degrees F). The first credible response is the STPT signal to scram the reactor, but STPT scram is not credited here. This case credits the ATLM and the DPS to scram the reactor when power exceeds 101.0 percent of rated power (85 percent power at SP2 condition). If crediting the scram (initiated by the ATLM or DPS), the analysis of the event has similar results to those obtained for the EC, ICL cases, that are not repeated here. The reactor attains a power level of about 107 percent of rated power. No operator action is required to mitigate this transient.

15.3.1.3 Conclusion

This event at the SP2 state point is the limiting event for LOFWH for the equilibrium and the initial core. In this event, it is assumed that all fuel rods entering transition boiling will fail. The number of rods in boiling transition is confirmed to be less than 1,000, based on the ΔCPR/ICPR results.

The maximum pressure remains within the limits of normal operating pressure, the vessel water level remains above 13 meters (above the L3 setpoint), the MCPR remains above the SLMCPR, and the reactor stabilizes at 101.00 percent rated power at SP2, about 200 seconds into the event. Therefore, the event resulting from LOFWH with SCRRI/SRI failure to insert satisfies the acceptance criteria.

15.3.1.4 Post-COL Activity

This event is potentially limiting with respect to the number of rods in boiling transition. The OLMCPR for each fuel cycle is established for the limiting event and documented in the COLR in accordance with TSs.

The thermal-mechanical analysis for each fuel cycle confirms that the calculated results remain within the assumptions of the radiological analysis. Any resulting limits on MLHGR are documented in the TS Section 5.6.3, COLR (a)(1) in accordance with TS 3.2.1.

15.3.2 Feedwater Controller Failure—Maximum Flow Demand

15.3.2.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

EC: The DCD, referencing the design of the FWCS, states that feedwater controller failure requires several failures of multiple feedwater pumps to result in maximum feedwater demand. The estimated frequency is less than 1.0×10^{-2} pry, which classifies it as an IE (DCD, Tier 2, Section 15A.3.5.1). With excess feedwater flow, the water level rises to the high reference point (Level 8, 14.5 meters above TAF), where feedwater pumps initiate a runback, the main turbine trips, and the reactor scrams. At water Level 8, there is a feedwater isolation signal. However, in this analysis, it is not credited because it does not make a significant difference. In DCD, Tier 2, Figure 15.3-2 depicts the sequence of events as a function of time, Table 15.3-1a summarizes the main points of the event, and Table 15.3-3 lists the complete sequence of events.

The calculated results indicate that the feedwater flow is ramped up to 170 percent of normal flow in about 2.5 seconds. At 12.7 seconds, the TBVs open to control vessel pressure. The vessel water reaches L8 at 14.5 seconds. At 15.4 seconds, turbine trip, reactor scram, and feedwater pump runback are activated. The main TBVs complete their opening at 15.5 seconds to relieve vessel pressure. At 15.6 seconds, a scram initiates with rod insertion. The value of the MCPR remains higher than the designated OLMCPR; therefore, no fuel damage or radioactive releases are anticipated. After 20 seconds, the reactor vessel water level falls to L2, which activates the ICs and CRD high-pressure injection operation to recover water level.

ICL: This event is almost the same as the EC event (i.e., the reactor scrams, and the water level falls to L2, which activates the ICs and the CRD high-pressure injection operation to recover water level).

FWTOD: Analyses of the FWTOD events in NEDO-33338 demonstrates that IICIs is the limiting event. Therefore, this event has not been explicitly analyzed in NEDO-33337.

15.3.2.2 Technical Evaluation

EC, ICL, and FWTOD: The DCD states that runout of all feedwater pumps requires more than one failure to take place. As such, the anticipated frequency is lower than 1.0×10^{-2}, and the event is included in the IE category. The calculated results indicate that the excessive feedwater flow event will cause minimal disturbance to the reactor, in that there is a small and short power peak and a corresponding small pressure peak, but the value of the MCPR will remain well above the designated OLMCPR limit. The reactor scrams, and when the vessel water level falls to L2, the ICs and the CRD high-pressure injection are activated to recover the water level. The reactor is continuously covered above the L2 level, and the vessel pressure remains close to the operating limits. No operator action is required. Therefore, the results of this event meet the Section 15.1.1.2 acceptance criteria.

15.3.2.3 Conclusion

This event does not result in any fuel failures or any release of primary coolant, and hence no radiological consequence analyses were performed.

In this event, the excessive feedwater flow causes an insignificant perturbation to vessel pressure but does not violate the fuel SLMCPR, and at the end of the event, the reactor is in a stable condition. Therefore, the results of the analyses are acceptable.

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

15.3.3.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 in evaluating this IE.

EC and ICL: The SB&PC system controls vessel pressure and steam turbine bypass. In Section 15.2.5 of this report, the staff examined the accidental opening all TCVs and TBVs. The electronic logic aspects of the SB&PC system are such that it would take multiple failures to accidentally open all of the TCVs and TBVs. Therefore, this event is considered as having a very small probability of occurrence and is categorized as an IE.

DCD, Tier 2, Section 15A.3.1.1, estimates that the frequency of this event is less than 1.0×10^{2} pry. In DCD, Tier 2, Table 15.3-4 and Figure 15.3-3 illustrate the calculated results of the time-dependent evolution of the event. At 19.30 seconds into the event, turbine inlet low pressure will initiate MSIV closure, which in turn will initiate reactor scram and IC operation at 20.5 seconds. At 24.1 seconds, MSIV closure will be completed, but bypass valves will remain open. Because of increased steam flow, the water level decreases, reaching the RPV L2 level at 31.60 seconds. At 36.50 seconds, the IC begins to return condensate coolant to the vessel, and at 41.80 seconds, the HPCRD 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, which results in an increase in the void fraction which reduces power. In the first few seconds, the feedwater system attempts to stabilize operation by increasing feedwater flow (due to lower vessel pressure, as shown in DCD Figure 15.3-3a). The MSIV position-switch scrams the reactor. Simultaneously, the IC steam flow increases to about 20 percent of normal steam flow because the MSIVs are closed. At this time, reactor operation stabilizes, with IC cooling having achieved normal water level. The MCPR value stays well above the safety limit value and increases during the event. No operator action is required to mitigate this event. The reactor scrams.

FWTOD: Analyses of the FWTOD events in NEDO-33338 demonstrates that IICIs is the limiting event. Therefore, this event has not been explicitly analyzed in NEDO-33337.

15.3.3.2 Technical Evaluation

EC, ICL, and FWTOD: The important feature is that vessel depressurization leads to decreasing power, reactor scram, and initiation of CRD and IC cooling.

Assuming that the required instrumentation and systems will operate as designed and as expected, the results of this event meet the acceptance criteria, vessel pressure decreases from the operating pressure, and MCPR increases after event initiation. Therefore, no cladding damage occurs, and the event evolves into a stable state.

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 did not agree with this characterization of the event. This event is an IE, as noted 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 the event as an IE rather than as a limiting fault. GEH agreed and changed the DCD to show the event as an IE. The staff confirmed that this change was included in DCD, Tier 2, Revision 5 therefore, RAI 15.3-29 is considered resolved.

15.3.3.3 Conclusion

This event does not result in any fuel failures or any release of primary coolant, and hence, no radiological consequence analysis was performed. Inadvertent opening of the TCVs and TBVs from power results in; fast reactor vessel depressurization, decrease in reactor power, vessel isolation, reactor scram, and IC initiation. The calculated results indicate that the vessel pressure remains below normal operating values, the MCPR is well above the designated OLMCPR, and the reactor is cooled by the ICs and assumes a stable state.

The results of this event satisfy the acceptance criteria; therefore, this event analysis is acceptable.

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

15.3.4.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

EC and ICL: This event assumes failure of the SB&PC system, with closure of all TCVs and TBVs. The DCD states that for this event to occur, more than a single failure is necessary and the probability is exceedingly low. DCD, Tier 2, Revision 6, Section 15A.3.2.1, estimates the event frequency as less than 10^{-2} pry, which classifies it as an IE.

As the TCVs and TBVs begin to close, vessel pressure increases and the core void collapses, which increases moderation and power until reaching the neutron high-flux setpoint at 1.78 seconds, initiating reactor scram. Control rod insertion starts at 2.03 seconds. TCV closure is completed at 2.5 seconds into the event. At about 20 seconds, 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 the 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 throughout the event. IC initiation does not take place, even though Level 2 is reached because neither the high dome pressure nor the low-water-level signals are in effect for more than 10 or more seconds and 6 or more seconds, respectively, required for IC initiation. No operator action is required to mitigate this event and the reactor scrams.

FWTOD: Analyses of the FWTOD events in NEDO-33338 demonstrates that IICIs is the limiting event. Therefore, this event has not been explicitly analyzed in NEDO-33337.

15.3.4.2 Technical Evaluation

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

The applicant based the categorization of this event as an IE on the performance of the SB&PC system. The peak pressure reaches 114 percent of operating pressure (i.e., it remains below the 110 percent of design pressure). The MCPR value remains above the OLMCPR; thus, no fuel damage is expected during this event. 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. In RAI 15.3-11, the staff questioned the TRACG ability to represent the sharp energy peak in pressure increase transients. The staff accepted the response that the TRACG is qualified to represent the energy peak. In RAI 15.3-11 S01, the staff requested the applicant to calculate the total energy deposition and the resulting cladding strain.

GEH responded to this request in a way that encompassed all the events that exhibit similar event behavior (i.e., high power, narrow peak events). GEH stated that the GE14E scheduled to fuel the ESBWR reactor is designed to withstand much greater energy deposition than any of the IEs in Section 15.3 of the ESBWR DCD. As stated in Section 15.2.3.1.2 of this evaluation the staff finds that neither fuel melting nor cladding strain is an issue for the ESBWR with GE14E fuel. Therefore, based on the applicant's response, RAI 15.3-11 is resolved.

FWTOD: This event is bounded by other events that have been found acceptable. This IE has not been explicitly analyzed in NEDO-33338.

15.3.4.3 Conclusion

This event does not result in any fuel failures or any release of primary coolant, and hence no radiological consequence analysis needs to be performed.

Inadvertent closing of the TCV and TBVs from power, results in fast vessel pressurization, power increase, reactor scram due to high power, and HPCRD activation to recover vessel water level. The calculated results indicate that the vessel pressure exceeds normal operating values but remains well below the SRV setpoint and 110 percent of the design pressure. The MCPR is above the designated OLMCPR, and the reactor assumes a stable state, therefore, the calculated results of this event satisfy the acceptance criteria.

15.3.5 Generator Load Rejection with Total Turbine Bypass Failure

15.3.5.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

EC and ICL: 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 analysis examines generator load rejection with total failure of the turbine bypass system. Section 15A.3.4 of the DCD states that the frequency of this event is estimated to be less than 10^{-2} pry which classifies it as an IE.

Upon receiving a load rejection signal, the TCVs will close and upon detection of insufficient turbine bypass, the RPS will initiate reactor scram. The calculated results indicate a sharp and narrow peak in total power, a dip in feedwater flow (5 seconds) due to increased pressure, and a pressure peak that decays slowly after 5 seconds. The neutron flux exceeds the neutron high-point scram signal, and the RPS initiates a reactor scram at about 0.15 seconds, with rod insertion initiating at 0.35 seconds. Peak dome pressure remains lower than the SRV activation pressure. The calculation ends at about 50 seconds, at which time the feedwater flow is at about 70 percent of rated flow. The MCPR value remains above the SLMCPR (1.18) but at 1.07 for ICL, which is below the SLMCPR. Because of reactor void collapse, the vessel level drops below the L2 level longer than 10 seconds which activates CRD high-pressure injection to recover water level. No operator action is required to mitigate this event. The reactor is shut down. The ICL case of a generator load rejection without bypass is a limiting event.

FWTOD: NEDO-33338, Revision 1, includes the results of generator load rejection with total turbine bypass failure for EC at SP2 (85 percent reactor power). This IE also has a high narrow power peak; the RPS issues a scram signal at 0.20 seconds, and control rod insertion initiates at 0.45 seconds.

The ensuing pressure pulse collapses the core void, and the vessel level falls below the L2 for more than 10 seconds, which activates the high-pressure CRD pumps to recover water level. Loss of reactor water level contributes the initial decrease in feedwater flow due to an increase in vessel pressure. However, at about 10 seconds, feedwater flow is above 100 percent of rated flow. The MCPR remains above the OLMCPR (NEDO-33338, Rev. 1, Figure A.3-2g).

At the state point SP1, for EC, the calculated results show similar events, except that in SP1, the resulting power peak is about 335 percent of the rated power (100 percent), while at SP2 the peak power is about 245 percent above the 85-percent rated full power at SP2. The calculated MCPR for SP1 is 1.14 (NEDO-33338, Rev. 1, Figure A.3-3g). For EC, this is a limiting IE event. Because the SP1 MCPR value is lower than the SLMCPR a radiological evaluation has been performed.

15.3.5.2 Technical Evaluation

<u>EC</u>: On sensing loss (or partial loss) of electrical load, the system commands the TCVs to close in the rapid mode, causing a sudden reduction in steam flow, void collapse, and pressure and power spikes. The calculated results (DCD, Tier 2, Revision 6, Figure 15.3-5a) show a very narrow high-power peak.

As in the IE described in Section 15.3.4 of this report, there is a very fast energy deposition for this event. The staff requested the applicant (RAI 15.3-11), to calculate the energy deposition along the pellet-clad mechanical interaction and cladding strain.

GEH provided a summary evaluation demonstrating that the GE14 fuel is capable of absorbing the energy deposition from any transient analyzed in this submittal. The staff reviewed the response to RAI 15.3-11 and finds the response acceptable; therefore, RAI 15.3-11 is closed. (See also Section 15.2.2.2.2 of this evaluation.)

The feedwater dips to about 60 percent of normal at about 3 seconds, and the simulated thermal power peaks well above the high-flux neutron scram setpoint at a fraction of a second after TCV closure, which initiates a scram. Dome pressure peaks also at the minimum of the feedwater flow but remains below the SRV setpoint. The increased pressure, decrease in feedwater 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 to recover the water level in the vessel. The DCD calculation shows that the MCPR remains above the OLMCPR, which indicates that there is no fuel damage. With the HPCRD injection, the core recovers water level and remains stable and shut down. The results of this analysis indicate that the IE satisfies the acceptance criteria.

<u>ICL</u>: The event is similar to the EC case; however, the MCPR value is lower than the SLMCPR and hence a radiological analysis will be performed.

<u>FWTOD</u>: The phenomenology is the same as in the EC event. However, because of the lower power at SP2, the MCPR is higher than for the EC or ICL events. The results of this analysis

indicate that the event satisfies the acceptance criteria.

The SP0 event analysis for the initial core shows that it is the most limiting event. In addition, the results indicate that the pressure increase following the power peak is much lower than the SRV opening setpoint. The reactor vessel water level is above TAF; therefore, the results meet the acceptance criteria.

15.3.5.3 Conclusion

Generator load rejection with total turbine bypass failure has a very low probability of occurrence. However, assuming that the SB&PC and the RPS respond as designed, the reactor will scram promptly.

In this event, it is assumed that all rods entering transition boiling fail. The number of rods in boiling transition is confirmed to be less than 1,000, based on the ΔCPR/ICPR results. The results of this event satisfy the acceptance criteria and hence are acceptable.

15.3.6 Turbine Trip with Total Turbine Bypass Failure

15.3.6.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

EC and ICL: A variety of causes, such as loss of turbine 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 less than 1.0×10^{-2} pry, which classifies it as an IE.

The sequence of events (after the scram signal) is nearly identical to that described in Section 15.3.5 of this report for generator load rejection.

In DCD, Tier 2, Table 15.3-7 and Figure 15.3-6 show the calculated sequence of events as a function of time. The results indicate a sharp and narrow peak in total power due to void collapse from the pressure pulse, a dip in feedwater flow (5 seconds) resulting from increased pressure, and a wide pressure peak that decays slowly after 5 seconds. Vessel water level decreases to lower than L2 at 12 seconds. At 0.10 seconds, the TSVs are closed. At 0.15 seconds, the RPS initiates reactor scram, and at .35 seconds, control rod insertion begins. At about 1.2 seconds, the MCPR is at 1.21. Unlike its behavior in 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. In the long term, CRD injection initiates to recover RPV level. Finally, the dome pressure peaks at about 4 seconds into the event but remains well below the SRV setpoint and well below 110 percent of the design value.

FWTOD: This IE has not been explicitly analyzed because it is bounded by generator load rejection with loss of turbine bypass.

15.3.6.2 Technical Evaluation

EC, ICL, and FWTOD: This event is almost identical to the generator load rejection with turbine bypass failure discussed in Section 15.3.5 of this report. The results of the analyses and the conclusions are the same. There is .06 CPR difference between the ICL and EC for SP0.

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 a staff - GEH information exchange meeting the staff requested that the applicant run the analyses for a longer period of time to show that the reactor has stabilized and long-term cooling has been established.

The applicant addressed the length of time the event was run for 50 seconds and concluded that the reactor was stabilized. DCD, Tier 2, Revision 6, and this longer run showed that the feedwater flow decreased and the pressure was lower and stable.

As in the previous two events, a pulse-like power event occurs. Therefore, the staff requested, in RAI 15.3-11, that the applicant calculate the energy deposition along with the associated pellet-cladding mechanical interaction. In their response, GEH demonstrated that the GE14 fuel has significant margins to cladding strain and fuel melt criteria as listed in the DCD, Tier 2, Revision 4, Table 15.0-3. The staff reviewed the RAI 15.3-11 response finds it responsive to the request and considers this issue closed. The same issue has also been described in Section 15.2.2.2.2 of this report.

No operator action is required to mitigate the event. The reactor is shut down and stable.

15.3.6.3 Conclusion

In this event, it is assumed that all rods entering transition boiling will fail. The number of rods in boiling transition is confirmed to be less than 1,000, based on the ΔCPR/ICPR results.

Turbine trip with total turbine bypass failure has a very low probability of occurrence. However, assuming that the SB&PC and the RPS respond as designed, the reactor will scram promptly without fuel damage or overpressurization. Therefore, the calculated results satisfy the acceptance criteria.

15.3.7 Control Rod Withdrawal Error (RWE) during Refueling

15.3.7.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

EC, ICL, and FWTOD: The DCD states that there is no postulated set of circumstances that results in an inadvertent control RWE while in the refueling mode. The applicant based this conclusion on system interlocks ensuring against inadvertent criticality. In addition, removal of the highest worth rod or two rods (one of which is the highest worth) 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. For example 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 possible 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 $1.0x10^{-2}$ pry, which classifies it as an IE, as indicated in DCD, Tier 2, Section 15A.3.11.3.

15.3.7.2 Technical Evaluation

Analysis of this IE is independent of fuel loading; therefore; the following discussion applies to EC, ICL, and FWTOD.

The design precludes inadvertent criticality because multiple failures must take place to cause criticality under refueling conditions. 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 stated 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 requested the applicant to provide the basis, using applicable information, for reaching this conclusion and the analyses demonstrating the magnitude of the consequences of an RWE under refueling conditions. GEH's response details the results for an RWE under refueling conditions and states that the probability is extremely small, as it would require multiple component failures and/or operator errors. In addition, GEH provided quantitative evaluation of the results of such an event (based on the 1999 event in Japan) to demonstrate that the potential consequences are insignificant.

The GEH response to RAI 15.3-19 S01 (MFN 08-564) presented results of refueling criticality transients and compared to measurements to the Japanese incident (Shika Electric 1999). The calculated values of radiation dose at the top of the vessel water are $1.3x10^{-8}$ mSv and the measurements at the charcoal filters designed to detect this radiation is below detection limits. Likewise pocket dosimeters designed to detect worker exposure to gamma radiation were also below the detection level. The calculated peak power achieved in the Shika reactor was about 15 percent of rated power. The calculated maximum energy deposition was 41-49 cal/gUO_2. These values are much lower than the fuel limit of 150 cal/g of UO_2. The consequences of a RWE event for ESBWR are expected to be less severe than the BWR-4. Neither radiation exposure nor material limits were approached; therefore, the staff agrees with GEH's evaluation and finds the results acceptable. Based on the applicant's response, RAI 15.3-19 is resolved.

15.3.7.3 Conclusion

The RWE during refueling is an extremely unlikely occurrence for the ESBWR because of interlocks, procedures, and reloading practices. The calculated probability for an RWE during refueling is extremely small, but also the estimated consequences are minimal to insignificant, as evidenced by the criticality event that occurred in a Japanese reactor in 1999. The ESBWR design, with the higher amount of water above the core (the high water level shields the refueling crew), indicates that the results would be even less significant. Regarding potential fuel damage, the estimated amount of thermal energy released during such an event challenges neither fuel integrity nor cladding strain. The IE would evolve slowly, thus offering the opportunity to the operator to respond by inserting control rods to absorb the excess reactivity. The IE analyses indicate that the results meet the acceptance criteria and therefore are acceptable.

15.3.8 Control Rod Withdrawal Error During Startup with Failure of Control Rod Block

15.3.8.1 Summary of Technical Information

Analysis of this IE is independent of fuel loading; therefore, the following discussion applies to EC, ICL, and FWTOD.

The staff used the acceptance criteria in Section 15.1.1.2 of this SER 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 SRNM has a period-based reactor trip function that will either initiate a rod block if the period is less than 20 seconds or a reactor scram if the period is shorter than 10 seconds.

Two sets of calculations were performed: (1) rods are continually pulled, and the short-period (20 seconds) rod block fails and the short period trip (10 seconds) is credited and (2) the short-period trip and the rod block logic of both channels of the RWM fail. In either case, the short-period scram (10 seconds) will terminate the event. Should the SRNM based short-period scram fail, the average power range monitor (APRM) high-flux will scram the reactor and terminate the event.

The assumptions for the analysis are that the reactor is critical at near zero power at 271 degrees C (520 degrees F), the rod worth of the withdrawn rods is 3 percent Δk, and the control rod worth speed is 28 millimeters per second (i.e., the nominal FMCRD withdrawal speed and, for rod-gang withdrawal, the reactor period monitored by any SRNM are the same).

The calculated enthalpy change for the 10-second period scram initiated from 1.1 to 4.6 seconds into the event, depending on core exposure and with a conservative addition of 2.23 seconds, is 66.2 joules per gram (J/g) compared to 712 J/g, the SRP fuel cladding failure criteria. (Figure 15.3-7a in the SRP, Appendix B, Section 4.2.) If the SRNM fails and the APRM scram is credited, the results are also within the acceptance criteria. In the second case (APRM at 15 percent power at a high-flux scram initiated between 7.8 to 9.2 seconds by the APRM), the results are 523 J/g, which are again within the acceptance value of 712 J/g. The analysis was performed using the staff-approved PANAC11 code evaluation.

The estimated frequency of this event is less than $1.0\text{x}10^{-2}$ 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 analyses included calculations to assess the impact from about zero power (i.e., less than or equal to the low power setpoint). The energy deposition model assumes adiabatic heating, which is a conservative assumption, and nominal rod (or rod gang) withdrawal rates for the FMCRD mechanism. After event initiation, the reactor promptly scrams on the period trip function. The average at-peak axial location enthalpy increase is well within the acceptance limits. The corresponding pressure and MCPR changes remain negligible. Should the period trip failed, the reactor would scram from the APRM 15 percent (or higher) power scram setting. The adiabatic heatup assumption adds a degree of conservatism. Therefore, the acceptance criteria are satisfied. No operator action is required to mitigate this IE.

15.3.8.3 Conclusion

Because the conservatively calculated fuel enthalpy is within the acceptance criteria of 712 J/g, the transient analyses results are acceptable. 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. In either case the acceptance criteria are satisfied; and the analyses results are acceptable.

15.3.9 Control Rod Withdrawal Error during Power Operation with Automated Thermal Limit Monitor (ATLM) Failure

15.3.9.1 Summary of Technical Information

The staff used the acceptance criteria summarized in Section 15.1.1.2 of this SER in evaluating this IE.

EC, ICL, and FWTOD: Analysis of this IE is independent of fuel loading; therefore, the following discussion applies to EC, ICL, and FWTOD.

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 the MLHGR. Should the reactor reach either of these limits due to control rod withdrawal, the system will remove the rod withdrawal permissive. Potential causes of rod (or rod assembly) withdrawal include procedural operator error and/or 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 less than $1.0x10^{-2}$ 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. The ATLM is based on actual core thermal limit information. In the case of manual control rod withdrawal the ATLM will remove the rod motion permissive when the core reaches any thermal limits.

The DCD states that in either case, the ATLM system will halt the progression of the event before any limits are violated. In the case of operator error or malfunction in the automated rod withdrawal sequence logic, the dual-channel multichannel rod block monitor (MRBM) will stop further rod withdrawal to protect the fuel. The DCD estimates that the potential damage will be limited to fuel failure of fewer than 1,000 rods and no fuel melt; therefore, the offsite dose will be within the acceptance criteria. No operator action is required to mitigate this IE.

15.3.9.2 Technical Evaluation

EC, ICL, and FWTOD: The DCD categorizes this event as highly unlikely. The ESBWR is equipped with the ATLM system, which would be able to arrest rod withdrawal based on actual core data, such as inlet and coolant temperatures, core power, core power distribution, and other parameters. The ATLM is a dual-channel subsystem, not subject to single failure. However, the DCD provides no reference regarding its classification as a safety-grade system. Nor does it refer 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.

In RAI 15.0-15, the staff requested the applicant to describe the bases for the reclassification of the RWE, including the initiating actions/events and mitigating strategies from all modes of operation to address the potential for a gang withdrawal error and to identify the proposed acceptance criteria for the new event classification.

In RAI 15.3-33, the staff requested that the applicant provide additional information regarding this event and particularly evaluation of barrier performance. The ATLM is included in the TS. GEH responded to RAI 15.0-15 and stated that the ESBWR design is such that the probability of RWE during power operation is very low, and hence it is categorized as IE. The basic design feature at issue is the provision of two systems (ATLM and MRBM) to prevent RWE during power operation. Section 15A of this report presents the staff's evaluation of the event frequency. Based on the applicant's response, RAI 15.0-15 is resolved.

GEH responded to RAI 15.3-33 by revising the DCD to include two sets of RWE during power operations: one for the AOO and the other for an IE that includes ATLM failure. In the first case it is assumed that the ATLM responds as designed and no thermal limits are violated. Should the ATLM fail the multiple rod block monitoring subsystem will be activated to control rod bocks in the RC&IS to prevent core thermal limit violation. For the case of ATLM failure GEH performed radiological assessments and concluded that the 1,000 rods failure analyses results are within the acceptance criteria of 2.5 rem. Based on the applicant's response, RAI 15.3-33 is resolved.

15.3.9.3 Conclusion

The control RWE during power operation with ATLM failure has been reclassified in the DCD as an IE. The reclassification is based on the design differences between the ESBWR and conventional BWRs that result in the SRP classification of the event as an AOO. The reclassification is based on the redundancy of systems in the ESBWR that prevent rod withdrawal that could result in violation of thermal limits. Control rod withdrawal could occur because of instrumentation failure or operator error. In the first case, the redundant instruments (with self-diagnosis) have an extremely small probability of failure. In the second case, another system (the MRBM) will intervene to stop control rod withdrawal and prevent (or limit) fuel damage. It is worth noting that the instrumentation will act in anticipation of the reactor exceeding thermal safety limits. Because the staff has accepted the instrumentation and its capabilities, this review finds the reclassification and the fuel damage estimate acceptable.

15.3.10 Fuel Assembly Loading Error, Mislocated Bundle

15.3.10.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

EC, ICL, and FWTOD: Mislocated assemblies involve at least two fuel assemblies at interchanged positions. If one is assumed to operate at a lower power level, the other will operate at a higher power. The plant is instrumented so that the core monitor can recognize the mislocated assemblies, allowing the operator to intervene and minimize the consequences of the mislocated fuel. 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, the core monitor will probably not recognize the mislocation. In this case, the possibility exists that the assembly may operate above the thermal limits. Should a mislocated assembly suffer thermal and/or mechanical damage resulting in leaking fuel rods; the application of existing leak detection techniques and local power suppression methods will minimize the radioactive leakage.

The maximum power at which the mislocated assembly will operate is limited by the detection capability of the core monitoring system. The DCD presented a conservative case, bounding for any mislocated assembly. First, it assumed that all fuel rods in the affected assembly will be damaged and become leakers. Then it assumes 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 the 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 bounding the end-of-cycle fission product inventory.

The estimated frequency of this event is less than 1.0×10^{-2} pry, which classifies it as an IE, as indicated in DCD, Tier 2, Section 15A.3.14.3.

15.3.10.2   Technical Evaluation

<u>EC, ICL, and FWTOD</u>: The staff approved Amendment 28 to GESTAR II which allowed the event category change from AOO to IE, acceptance criteria 10 percent of the radiation dose limit specified in 10 CFR 52.47(a)(2)(iv)(A).

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 design features and loading procedures 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 could be suppressed to minimize fuel leakage. A once-removed mislocated assembly from the detector may not be identified, but the power mismatch is limited. On such occasions, it is possible that the mislocated bundle will operate outside its thermal and/or mechanical limits and damage the cladding. Monitoring will detect fuel leakage (from any cause), and the leakage can be minimized by suppressing power to the segment with the leakers.

The staff established that the GEH estimated potential site boundary dose rate, is conservative as described in GESTAR II, Amendment 28, Revision 1, “Misloaded Fuel Bundle Event Licensing Basis Change to Comply with SRP Section 15.4.7,” dated August 23, 2004. DCD

Section 15.3.11.3 describes core verification requirements and confirmation of assumptions, as summarized in the following:

> The NRC requires licensees to certify that core verification procedures have the following characteristics:
>
> During fuel movement, each move (location, orientation, and seating) is observed and checked at the time of completion by the operator and spotter.
>
> After completion of the core load, the core is verified by a video recording of the core using an underwater camera.
>
> Two independent reviewers perform the verification of the bundle serial number location, orientation, and seating. Each independently records the bundle serial numbers on a core map, which is verified with the core design loading pattern. The licensees are expected to follow the above procedures during refueling.

15.3.10.3 Conclusion

In this section, the applicant analyzed the fuel misloading error event. 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 52.47(a)(2)(iv)(A) and (B) 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 proposed design satisfies the GDC required control instrumentation and conservative estimate of the site boundary exposure to meet the criteria of 10 CFR 52.47(a)(2)(iv)(A) and (B). Therefore, the fuel misloading analysis is acceptable.

15.3.10.4 Post-COL Activity

In accordance with TS 5.6.3, Item C and consistent with the staff's SER related to the Global Nuclear Fuel's (GNF) request for proposed Amendment 28 to GESTAR II, March 8, 2007, "Final Safety Evaluation for GNF Topical Report, Amendment 28, the following conditions are to be met by individual licensees:

1. Plants seeking to apply the infrequent incident must confirm that their core refueling verification procedures meet the requirements defined in Section 5.3, Fuel Loading Error Analysis requirements, of the GESTAR US Supplement. This confirmation will be documented every refueling outage through the reload design documentation and the analysis basis stated in the Supplemental Reload Licensing Report (SRLR).

2. Should a bundle mislocation and seating occur and go undetected, the plant specific acceptance of this categorization for the plant will be revoked, and the classification of this event will revert from "infrequent incident" to an "anticipated operational occurrence classification" immediately.

15.3.11 Fuel Assembly Loading Error, Misoriented Assembly

15.3.11.1 Summary of Technical Information

The staff used the requirements summarized in Section 15.1.1.2 of this SER in evaluating this IE.

<u>EC, ICL, and FWTOD</u>: The probability that a misoriented assembly is placed in a core position and not detected is very small. Proper fuel orientation has five different visual indications: (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 the probability of a misoriented assembly not being detected to be very small.

The letter from the staff, to Lingenfelter, GEH, March 8, 2007, "Final Fafety Evaluation for GNF Topical Report, Amendment 28," is related to the GNF request for proposed Amendment 28 to GESTAR II, 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 mislocation 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 less than $1.0x10^{-2}$ 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-radiation 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 therefore, finds that the fuel misloading analysis is acceptable.

15.3.11.4 Post-COL Activity

In accordance with TS 5.6.3, Item C and consistent with the staff's SER related to the Global Nuclear Fuel's (GNF) request for proposed Amendment 28 to GESTAR II, March 8, 2007, "Final Safety Evaluation for GNF Topical Report, Amendment 28, the following conditions are to be met by individual licensees:

1. Plants seeking to apply the infrequent incident must confirm that their core refueling verification procedures meet the requirements defined in Section 5.3, Fuel Loading Error Analysis requirements, of the GESTAR US Supplement. This confirmation will be documented every refueling outage through the reload design documentation and the analysis basis stated in the Supplemental Reload Licensing Report (SRLR).

2. Should a bundle be misoriented and go undetected, the plant specific acceptance of this categorization for the plant will be revoked, and the classification of this event will revert from "infrequent incident" to an "anticipated operational occurrence classification" immediately.

15.3.12 Inadvertent Shutdown Cooling Function Operation

15.3.12.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

EC, ICL, and FWTOD: This event concerns the power increase resulting from misoperation of the RWCU/SDC system either during power operation or during startup. Malfunction of the SDC leads to lower temperature cooling water entering the core, which results in 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. Any potential event ensuing from this event is bounded by the LOFWH event analyzed previously.

15.3.12.2 Technical Evaluation

ICL, EC, and FWTOD: 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 the statement that with or without operator action the plant thermal limits will not be violated. The GEH response provided a conservative quantification of the expected sequence of events. The response was in terms of results of TRACG analyses from start-up and power operation conditions. Conservative results were assured because of the conservative input assumptions in the analyses. During start-up a reactor scram on high flux or short period may occur or the reactor may reach a higher stable power level. Likewise during power operation no thermal limits are reached nor violated. Appropriate text modification to DCD, Tier 2, Subsection 15.3.12.2 reflects the response to RAI 15.3-34. The results indicate that the peak power and the stable power to be achieved by the event are a very small portion of the rated power, thus having an excess heat transfer capacity to cool the core. Therefore, this event does not represent a threat to core thermal limits, nor pressure vessel integrity, and it is acceptable. Based on the applicant's response, RAI 15.3-34 is resolved.

15.3.12.3 Conclusion

GEH submitted a conservative quantification of the event resulting from an inadvertent activation of the RWCU/SDU during startup or low-power operation. The results indicate a benign event that does not pose a threat to either the core or the pressure vessel and does not require operator intervention. The results satisfy the acceptance criteria and the event analysis is acceptable.

15.3.13 Inadvertent Opening of a Safety/Relief Valve

15.3.13.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

EC, ICL, and FWTOD: Inadvertent 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 on high suppression pool temperature. 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 (DCD, Tier 2, Figure 15.3-8a). The vessel pressure settles at a slightly lower operating pressure, and the MCPR remains well above the OLMCPR. The operator will monitor suppression pool temperature and water level and isolate makeup from external sources as necessary. DCD, Tier 2, Section 15A.3.8, estimates the frequency of this event as less than 1.0×10^{-2} pry, which classifies it as an IE.

In estimating the frequency of inadvertent SRV opening the applicant refers to a factor attributed to the triplicate electronic control system (also in Sections 15.3.1, 15.3.3, and elsewhere) i.e., the applicant based the valve failure probability on the electronic portion of the control system and ignored the mechanical aspects of valve failure. In RAI 15.3-16 the staff requested GEH to justify their choice or revise the probability values in Section 15A.3. GEH response (MFN 07-264) also documents the mechanical aspects of TBVs and SRVs in accident analyses and calculated revised failure probability values. Based on the applicant's response that accounted for the electrical as well for the mechanical causes of valve failure, RAI 15.3-16 is resolved.

15.3.13.2 Technical Evaluation

ICL, EC, and FWTOD: The analytical results indicate that this is an inconsequential event, with or without operator intervention to close a discharging SRV. Neither fuel damage nor overpressurization occurs, and the MCPR remains well above the designated operating limit (DCD, Tier 2, Figure 15.3.8). These results satisfy the acceptance criteria for fuel damage and overpressurization; therefore, the event analysis is acceptable.

15.3.13.3 Conclusion

Assuming no operator action is taken to close the SRV 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 and RAI 15.3-16 resolved.

15.3.14 Inadvertent Opening of a Depressurization Valve

15.3.14.1 Summary of Technical Information

The staff used the requirements summarized in Section 15.1.1.2 of this SER in evaluating this IE.

A depressurization valve (DPV) could open as the result of a valve malfunction or operator error. The difference between the DPV and the SRV (Section 15.3.13 of this report discusses the SRV) is that the DPV is bigger and also discharges into the drywell, where it could raise the drywell pressure (within a few seconds) to the reactor scram setpoint. The opening of a DPV 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 DPVs discharge into the drywell, and the reactor will scram on high drywell pressure. DCD, Tier 2, Section 15A.3.9, estimates the frequency of this event as less than $1.0\text{x}10^{-2}$ 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 and does not need operator action to mitigate the event. The acceptance criteria are satisfied.

15.3.14.3 Conclusion

This event meets the pressure and fuel damage acceptance criteria. Therefore, the results of the event analyses are acceptable.

15.3.15 Stuck-Open Safety/Relief Valve (SRV)

15.3.15.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

ICL, EC, and FWTOD: A stuck-open SRV is attributed to valve malfunction (either electronic or mechanical), regardless of 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 reactor shutdown. The SRV discharges into the suppression pool. The calculated sequence of events (after scram) indicates that depressurization begins at 10 seconds; vessel water level reaches RPV level L2 at 19.3 seconds, activating HPCRD injection; and low steamline pressure activates MSIV closure which, in turn, activates the ICs. 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 well above the normal operating value. The estimated frequency of a stuck-open relief valve is less than 1.0×10^{-2} pry, classifying this event as an IE. (See DCD, Tier 2, Section 15A.3.10.1.) Operator actions consist of monitoring the suppression pool temperature and water level and isolating external sources to the containment as necessary.

15.3.15.2 Technical Evaluation

Mitigation of this event depends on successful removal of decay heat. With successful operation of the HPCRD 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 limiting values; therefore, the acceptance criteria are satisfied, and the results of the analysis are acceptable.

In RAI 15.3-23, the staff requested the applicant to justify the event category and to address the mechanical performance history related to this event.

In its response to this RAI, the applicant noted that the SRVs and safety valves addressed in DCD, Tier 2, Section 15.3.15, event evaluations are designed in accordance with the ASME Code, Section III, Subsection NB (Class 1), as described in DCD, Tier 2, Sections 3.9 and 5.2.2. Per the qualification testing described in DCD, Tier 2, Sections 3.9 and 3.10, for these valves, the design deformation limit criteria for disk-to-seat geometry and disk movement clearances are applied to ensure that the valve performance requirements for pressure response during dynamic load conditions and reclosure leaktightness are met, up to and including exposure to Service Level D loads. The applicant also indicated that it would revise the DCD to describe in more detail its evaluation of the potential for inadvertent opening of a relief valve. As a result, Revision 5 to DCD, Tier 2, Section 15A.3.8, “Inadvertent Opening of a Safety Relief Valve,” describes the controlled operation of these valves using procedures and the design of the human-system interface in the control room to support the determination that operator error resulting in inadvertent SRV actuation is negligible. The NRC staff considers the applicant's clarification of the SRV and SV design as consistent with the ASME Code, and the evaluation of a potential inadvertent SRV opening described in the DCD to be acceptable.

In RAI 15.3-23 S01, the staff requested clarification of the statement that allowable limits are beyond the elastic region yet do not exhibit deformation, and the assigned zero for operator error regarding SRVs and/or SVs and to update the DCD with the information. In their response GEH stated that deformation does not take place until exposure to service level D. In such cases the exposed valve does not re-enter service until after rigorous qualification testing takes place. Regarding operator error, GEH modified the DCD Section 15A.3.8 to demonstrate that operator error is negligible. In addition to the response to RAI 15.3-23 S01 GEH added a DCD markup for sections 15A.3.8.1, to 15A.3.8.3. The staff reviewed the revised Section 15A.3.8.1/2/3 and finds they are responsive to the staff's request and GEH justified the frequency values used for the SRV and SV response therefore; the staff considers RAI 15.3-23 to be resolved.

15.3.15.3 Conclusion

As stated above the NRC staff concludes that the design of the ESBWR with respect to potential malfunction of SRVs satisfies the NRC regulations and, therefore, is acceptable.

Summary of Staff Review Findings for Sections 15.3.1 to 15

There are five criteria for this review: reactor vessel water level, RCPB pressure, radiological consequences, containment and suppression pool pressure and temperature, and control room radiation exposure. Control room radiation exposure is discussed elsewhere in this report; therefore, the remaining four criteria are applicable in the review of Section 15.3.

(1) Vessel Water Level

There is no core uncovery during any of the IEs evaluated above, the reactor water level is always above the TAF and therefore, meets the acceptance criteria.

(2) RCPB Pressure

An increase in vessel dome pressure occurred in several events, but none reached the SRV setpoint. The ESBWR vessel is large and difficult to overpressurize in the context of the events considered in Section 15.3, therefore, RCPB meets the acceptance criteria.

(3) Radiological Consequences

Radiological consequences of IEs are relatively less severe than the DBAs evaluated in Section 15.4 of this report and are bounded by the radiological consequences of the DBAs.

Evaluation of the actual radiological values is not part of this review. In this context, the limiting IE is the ICL generator load rejection with total turbine bypass failure. Several events that depend on fuel loading and are characterized as Post COL activity items could become as limiting (or more limiting) with different future fuel loadings.

(4) Containment and Suppression Pool Pressure and Temperature

Inadvertent opening of an SRV (or a stuck-open SRV) will discharge steam into the suppression pool, and if not corrected, the reactor will scram on high suppression pool temperature. Similarly, with inadvertent opening of a DPV that discharges into the drywell, the reactor will scram on high drywell pressure. Both types of scram are credited with terminating the events.

No required operator actions are identified to mitigate the events. However, if as a result of an event, the reactor is not at normal operating parameters, the operator is expected to intervene to bring the reactor within normal operating conditions.

15.3.16 Liquid-Containing Tank Failure

15.3.16.1 Regulatory Criteria

The staff used the requirements summarized in Section 15.1.1.2 of this SER in evaluating this IE.

The staff reviewed DCD, Tier 2, Revision 5, 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 those 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.

- GDC 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to the ability of structures housing the LWMS to control releases of liquid radioactive wastes.

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

- SRP Section 11.2, “Liquid Waste Management System”
- SRP Section 11.2, BTP 11-6
- SRP Section 15.7.3, “Postulated Radioactive Releases Due to Liquid-Containing Tank Failures,” issued July 1981 (and with the SRP updated in March 2007) (The requirements of SRP Section 15.7.3 have been relocated to BTP 11-6.)
- 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 to the provisions used to control leakages.

15.3.16.2 Summary of Technical Information

DCD, Tier 2, Revision 5, 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. The LWMS includes the equipment drain subsystem, the floor drain subsystem, the chemical drain subsystem, and the detergent drain subsystem. DCD, Tier 2, Revision 5, Figure 11.2-1, provides an overview of the LWMS process diagram depicting all subsystems, while Figures 11.2-1a, 11.2-1b, 11.2-3, and 11.2-4 present specific design details for each subsystem. Figure 11.2-2 of the DCD provides an LWMS process stream information directory and simplified flow diagram. DCD Sections 9.3, 9.2, and 10.4 describe the equipment and floor drain drainage systems and origins and discharges of nonradioactive effluents. DCD, Tier 2, Revision 5, Figures 1.2-21 to 1.2-25, present the general arrangements of the LWMS within the radwaste building.

The LWMS and its components are housed in the radwaste building and located in radiologically controlled access areas. DCD, Tier 2, Revision 5, Figures 11.2-1a–b, show 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. Subsystem tanks and components are vented to the radwaste building ventilation system, as described in DCD, Tier 2, Revision 5, and Section 9.4. The cubicles where tanks are located are lined with steel liners to avoid releases of radioactive materials in the environment. Concrete walls are coated with sealants for additional protection and minimization of radioactive waste (e.g., in the form of contaminated concrete). 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 any of the systems can segregate liquid wastes from various sources and process them separately. The subsystems maintain the segregation of process streams to support the most appropriate treatment of wastes by the LWMS.

Cross-connections between subsystems provide additional flexibility in processing wastes by alternate methods and provide redundancy if one subsystem were to become 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 Technical Evaluation

The staff evaluated 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 5, Section 15.3.16, with information drawn from DCD, Tier 2, Revision 5, Sections 11.2 and 12.2. Section 12.2 of DCD, Tier 2, Revision 5, 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 and 61; 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 prior DCD, Tier 2 revisions, the staff could not confirm that the approach used in assessing the impact of tank failure was consistent with the 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 considers 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.

A review of DCD, Tier 2, Revision 3, Section 15.3.16, indicated that the technical approach was not consistent with that described in SRP Section 11.2 (BTP 11-6). The analysis assumed 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 that 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." This implies 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 skid-mounted radwaste processing systems in treatment bays would provide 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 2.4-29 S01, the staff requested the applicant to address these inconsistencies with the NRC's guidance and acceptance criteria in 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 asked the applicant to provide additional details on "special design features" to support the approach, to 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 asked 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, and 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 the use of 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 appropriate radwaste subsystem for processing. The applicant has updated Section 11.2.2.3 of DCD, Tier 2, Revision 5, to indicate that rooms where tanks are located will be lined with steel to prevent accidental releases of radioactivity in the environment. Similar revisions were made in DCD, Tier 2, Revision 5, 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 and 61 for the control of releases of radioactivity in the environment during normal operations and AOOs. The staff evaluated the corresponding revision of DCD, Tier 2, Revision 5, and finds the changes acceptable as the inclusion of steel liners in cubicles housing radwaste system tanks would mitigate the release of liquid radwaste to the environment. This approach is also consistent with the guidance of SRP Section 11.2 and BTP 11-6 in including mitigating engineering design features. Based on the applicant's response, RAIs 15.3-4, 15.3-5, and 2.4-29 S01 are resolved.

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 5, Tables 15.3-17 and 15.3-18. The amounts of radioiodines assumed for this analysis 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 treatment. The analysis assumes an atmospheric dispersion factor of 2.0×10^{-3} seconds (s)/m^3 for a receptor located at the exclusion area boundary (EAB). The applicant's results, presented in DCD, Tier 2, Revision 5, Table 15.3-19, indicates an inhalation dose TEDE of 0.072 rem (0.72 millisievert (mSv)), for an offsite receptor. The staff confirmed the result and concludes that the dose complies with the 10 CFR 20.1301 dose limit of 0.1 rem (1 mSv) for members of the public.

The staff finds that the inclusion of such design features to mitigate the consequences of the 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 and 61 for the control of 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 comply with the dose limit of 10 CFR 20.1301 and 20.1302. 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.

Under the provisions of 10 CFR 52.47(b)(1), a DCD application is required to propose inspections, tests, analyses, and acceptance criteria (ITAAC) for the LWMS. The ITAAC are described in ESBWR DCD, Tier 1, Revision 5, Section 2.10.1 and Tables 2.10.1-1 and 2.10.1-2. In summary, the relevant ITAAC include the following:

- Confirming the description and functional arrangement of the LWMS

- Assessing the pressure and leakage integrity of the LWMS when subjected to hydrostatic testing pressures expected during operation

- Confirming the installation of steel liners in cubicles housing LWMS tanks and vessels for the purpose of ensuring that, in the event of a tank rupture, the effluent concentration limits of Table 2 (Column 2) of Appendix B to 10 CFR Part 20 will not be exceeded at offsite locations.

15.3.16.4 Conclusion

The staff finds that the analyses and impact of the postulated failure of a tank and its components, located outside of containment are consistent with NRC's regulatory requirements and guidance. The applicant has met the requirements of GDC 60 and 61 with respect to the control of releases of radioactive materials to the environment by providing design features 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, and Table 2 (Column 2), in the nearest source of 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 and 61 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.

15.4 Analysis of Accidents

15.4.0 Design-Basis Accidents

In DCD, Tier 2, Section 15.4, "Analysis of Accidents," the applicant performed radiological consequence assessments of the following five reactor 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. Both tables have the same title, "Envelope of ESBWR Standard Plant Site Parameters." 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 to provide reasonable assurance that the radiological consequences of a DBA will be within the radiological dose limits specified in 10 CFR 52.47(a)(2) and 10 CFR 100.21, "Non-Seismic Siting Criteria." No specific reactor site is associated with the ESBWR design. The DBAs analyzed in DCD, Tier 2 include the following:

- FHA (DCD Section 15.4.1)
- LOCA (DCD Section 15.4.4)
- MSLB outside containment (DCD Section 15.4.5)
- failure of small lines carrying primary coolant outside containment (DCD Section 15.4.8)
- failure of reactor water cleanup system line outside containment (DCD Section 15.4.9)

In addition, in DCD, Tier 2, Section 15.4.7, the applicant performed a radiological consequence assessment for the feedwater line break outside containment. This event is neither listed as a DBA nor required to be analyzed for its radiological consequences in 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."

Both SRP Section 15.0.3 and RG 1.183 list the BWR CRDA 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 CRDA need not be considered because such an accident is extremely unlikely with the improved design of the ESBWR, and furthermore, there is no credible basis for the control rod drop to occur.

The ESBWR design employs the FMCRD, which has several new features that are unique and not found in current BWR locking piston control rod drives. DCD, Tier 2, Section 4.6.1, "Information for Control Rod Drive System," describes the FMCRD system, and the staff evaluated and accepted the system in Section 4.6 of this SER. For the CRDA to occur in the ESBWR design, it is necessary for both Class 1E separation and detection devices or failure of the rod block interlock and of the latch mechanism to occur simultaneously with the occurrence of a stuck rod on the same FMCRD.

In 1996, the NRC certified the advanced boiling-water reactor (ABWR) design in Appendix A, "Design Certification Rule for U.S. Advanced Boiling Water Reactor," to 10 CFR Part 52, then titled "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," with the same fine motion control rod drive (FMCRD) design as that provided in the ESBWR. The staff accepted the FMCRD design in the ABWR without the applicant analyzing its potential radiological consequences of a CRDA because such an accident was considered to be extremely unlikely.

However, the staff did evaluate the radiological consequences for this event in the ABWR SER (NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, Main Report," issued July 1994), Section 15.4.1, "Control Rod Drop Accident," and found that the radiological consequences of a postulated CRDA using the FMCRD design at the EAB, LPZ, and control room were well within the dose acceptance criteria.

In Section 15.4.6.5 the applicant stated that conservative analyses were performed for the ESBWR using adiabatic heat retention in the fuel and maximum expected control blade worth. In addition, the ESBWR design proposed a higher hypothetical set of X/Q values than those certified for the ABWR design. The analyses included the initial and EC loadings. The applicant reported that during a postulated CRDA the fuel enthalpy rise remains well below the lower bound clad failure limits in Appendix B of Revision 3 to SRP Section 4.2. Based on the acceptance of the ABWR FMCRD and the results of the conservative calculations reported by the applicant the staff accepted the FMCRD design in the ESBWR. Because the clad failure limits are not violated there is no need for radiological analyses. However, because CRDA analysis is required (SRP Section 15.4.9, "Spectrum of Rod Drop Accidents (BWR)," and Section 4.2, Appendix B, Revision 3, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents" to provide the interim acceptance criteria and guidance for the reactivity-initiated accident (RIA)) the applicant performed such analysis that is reported in Section 15.4.6 of this evaluation.

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 design is such that a spent fuel cask drop height of 9.2 meters (30 feet) cannot be exceeded. In RAI 15.4-5 the staff requested information on the fuel building design and configuration to preclude a postulated spent fuel cask drop. SRP Section 15.7.5, "Spent Fuel Cask Drop Accidents," requires a design-basis radiological consequence analysis only if a cask drop exceeding 9.2 meters. The applicant's response to RAI 15.4-5, provided fuel building figures showing spent fuel cask movements and lifting heights as security-related sensitive information in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding." The staff finds that the cask drop distance is within the 9.2-meter (30-foot) height limit specified in SRP Section 15.7.5. Therefore, neither the staff nor the applicant analyzed the radiological consequences for a spent fuel cask drop accident. Based on the applicant's response, 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 of this event to be bounded by that resulting from the MSLB accident outside containment for all light-water BWRs; therefore, this event is neither listed as a DBA nor required to be analyzed for radiological consequences 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. Neither SRP Section 15.0.3 nor RG 1.183 lists this event as a DBA, nor is required to be analyzed for its radiological consequences. During promulgation of Appendix A to 10 CFR Part 52, the Advisory Committee on Reactor Safeguards (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 five selected 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 the selection 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 52.47(a)(2)) of 0.25 sievert (Sv) (25 rem) TEDE and the control room 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 provided in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and in RG 1.183, and a set of hypothetical χ/Q values (discussed in Section 2.3.4 of this report). No specific reactor 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 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 1 and Tier 2 in order to demonstrate that the COL application meets the offsite dose criteria specified in 10 CFR 52.47(a)(2) and the control room operator dose criterion specified in GDC 19 of Appendix A to 10 CFR Part 50. This is identified as COL Information Item 2.0-1-A.

15.4.1 Fuel-Handling Accident

15.4.1.1 *Regulatory Criteria*

The staff reviewed DCD, Tier 2, Section 15.4.1, "Fuel Handling Accident," in accordance with the guidance provided in SRP Section 15.0.3 and RG 1.183. The staff evaluated the radiological consequences of an 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 also used a criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences from a postulated FHA in the control room 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, "Accident Source Term," which has the same regulatory dose criteria specified in 10 CFR 52.47(a)(2) (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.

15.4.1.2 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 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 fuel building. The staff requested that the applicant provide the source term assumptions used in its FHA radiological consequence analysis (RAI 15.4-1). The applicant provided the information requested in DCD, Tier 2, Revision 2.

In RAI 15.4-1 S01, the staff requested that the applicant provide the FHA radiological consequence analyses for both a fuel assembly drop onto the reactor core and into the spent fuel storage pool. In its response to RAI 15.4-1 S01, the applicant provided the requested analyses in DCD, Tier 2, Revision 4. The results indicate that the fuel building release is bounding, due to the higher control room X/Q values. In its radiological analyses, 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. Therefore, the staff finds this portion of applicant's response to be acceptable.

In RAI 15.4-1 S02, the staff requested that the applicant provide administrative controls to mitigate radiological consequences of an FHA in accordance with the guideline provided in Section 5.3 of Appendix B to RG 1.183. This section states that if the containment (e.g., the ESBWR reactor building or fuel building) is open during fuel-handling operations, the TS allowing such operations should include administrative controls to close the open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure, should an FHA occur.

In response to RAI 15.4-1 S02, dated December 8, 2008, the applicant stated that the reactor building or fuel building doors that potentially open following the implementation of COL Information Item 2A.2-2-A, "Confirmation of the Reactor Building X/Q Values," will have X/Q values that are less than the X/Q values used in the ESBWR DCD, Revision 5, to meet the dose limit. COL Information Item 2A.2-2-A, in this SER states the following:

> If the X/Q values (for a release from any door or personnel air lock on the east sides of the Reactor Building or Fuel Building have X/Q values that would result in doses greater than the bounding dose consequence reported for the FHA in the ESBWR DCD, Revision 5) are not bounded by the X/Q values in the ESBWR DCD, Revision 5, for a release in the Reactor Building, the affected doors or personnel air locks must be administratively controlled prior to and during movement of irradiated fuel bundles.

Based on the applicant's response, RAI 15.4-1 is resolved.

15.4.1.3 Technical Evaluation

The staff has reviewed the applicant's analyses 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 experience damage to the cladding on all fuel rods. One fuel assembly is dropped either into the spent fuel storage pool or onto the reactor core, which impacts fuel assemblies (equivalent to one fuel assembly) in the pool or in the reactor core. In its evaluation the staff considered the wet weight of a dropped fuel assembly, a drop height, and a factor of 2 reductions to obtain the kinetic energy in a fuel assembly drop through water. The staff finds the total number of failed fuel rods is less than the total fuel rods in two fuel assemblies. 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 in accordance with the guideline provided in RG 1.183. The kinetic energy developed in this drop is conservatively assumed to be dissipated in damage to the cladding on all fuel rods in two fuel assemblies. All fission product inventories in the fuel rod gap of the damaged fuel rods are 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 in the reactor core are assumed to be in all fuel rod gaps) 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 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.

For the control room, the applicant assumed that the room 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 ventilation system will be operational during this event with no credit for fission product removal. The applicant used a normal control room ventilation system flow rate of 270 liters per second (L/s) (572 cubic feet per minute (cfm)) as an unfiltered air in-leakage rate into the control room 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 control room. 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 that the staff's results agree with the applicant's values. Both the applicant's and the staff's results meet the relevant dose acceptance criteria at the EAB, LPZ, and control room.

The staff performed independent radiological consequence calculations for the FHA occurring 24 hours after reactor shutdown, coincident with a loss of the spent fuel pool cooling capacity. The ESBWR design does not rely on safety-related equipment to cool the pools that contain spent fuel. The reactor building is provided with passively acting relief devices that allow the building to vent to the environment if the spent fuel pool cooling lost during refueling operation. The staff finds that the radiological consequence resulting from the FHA coincident with a loss of the spent fuel pool cooling capacity still meets the relevant dose acceptance criteria as stated above at the EAB, LPZ, and control room.

15.4.1.4 Conclusion

The staff concludes that the ESBWR design, as bounded by the hypothetical X/Q values proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated FHA at the EAB and LPZ will be well within the dose criteria in 10 CFR 52.47(a)(2) (i.e., 25 percent or 6.3 rem (.063 Sv) TEDE) and that the radiological consequences to an individual in the control room as a result of a postulated fuel FHA will be within the dose criterion in GDC 19 5 rem (.05 Sv)) TEDE. Therefore, the staff finds the radiological consequence analysis provided by the applicant to be acceptable.

15.4.2 Loss-of Coolant Accident Containment Analysis

Staff evaluation of this Section is included in Section 6.2 of this report

15.4.3 Loss-of-Coolant Accident ECCS Performance Analysis

Staff evaluation of this Section is included in Section 6.3 of this report

15.4.4 Loss-of-Coolant Accident Inside Containment Radiological Analysis

15.4.4.1 Regulatory Criteria

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 52.47(a)(2), 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 a LOCA in the control room 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 radiological consequence analyses.

15.4.4.2 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 assumed 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 (1) the passive containment cooling system (PCCS) in the containment, (2) the natural deposition of fission product aerosol in the containment, (3) an essentially leak tight containment barrier, and (4) 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 52.47 and GDC 19.

To support its conclusion, the applicant submitted the following LTR and three research reports:

- Licensing Topical Report, NEDE-33279P, "ESBWR Containment Fission Product Removal Evaluation Model" (GE Licensing Topical Report), Revision 2, dated July 9, 2008. This report provides the methodology used by the applicant to evaluate the radiological consequences of a postulated design-basis LOCA.

- Research Report, VTT-R-04413-06, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment—Part 1," issued October 2006 (VTT Report No. 1).

- Research Report, VTT-R-04413-06, “Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment—Part 2,” issued December 2006 (VTT Report No. 2).

- Research Report, VTT-R-06771-07, “Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment—Part 3,” Revision 2, issued March 2008 (VTT Report No. 3).

The staff found that the most relevant 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, in RAI 15.4-6, the staff requested that the applicant incorporate the radiological consequence analyses provided in these reports into DCD Section 15.4 or incorporate the reports into DCD Chapter 15 as appendices. In its response, the applicant revised DCD, Tier 2, Section 15.4, Revision 5, and incorporated the radiological consequence analyses from the LTR and the VTT reports in Revision 6 as the staff requested in RAI 15.4-6. The staff finds that the applicant’s response is acceptable and therefore, this open item is resolved.

The applicant postulated the following three LOCA scenarios:

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

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

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

The applicant originally proposed accident scenarios (1) and (2) above, stating that the reactor core uncovers and fission product release timing is shortest for these scenarios. For accident scenarios (1) and (2), 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. Both 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 that the applicant add one additional accident sequence, “loss of preferred power with ADS operating and with degraded low-pressure makeup system,” since it is the most 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. In its response, the applicant revised DCD, Tier 2, Section 15.4, in its Revision 5 and incorporated 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. The staff finds that the applicant's response is acceptable and therefore, RAI 15.4-17 is resolved.

The applicant performed and provided the radiological consequence analysis only for accident scenario (1) above in DCD, Revision 3. In RAI 15.4-7, the staff requested that the applicant provide the same radiological consequence analyses for accident scenarios (2) and (3) above as for accident scenario (1). The staff requested the applicant to incorporate these two remaining radiological consequence analyses into the LTR NEDE-33279P, "ESBWR Containment Fission Product Removal Evaluation Model" and DCD Section 15.4. 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.

In its response, the applicant stated that they will revise the LTR and DCD, Tier 2, Section 15.4, accordingly. Subsequently, Revision 1 of the LTR and Revision 5 of the DCD, the applicant provided the same radiological consequence analyses for accident scenarios 2 and 3 in addition to the accident scenario 1 and discussed the results of the radiological consequences and fission product removal rates in the containment for all three accident scenarios. The staff finds the applicant's response is acceptable and therefore, this open item is resolved.

Proposed DCD, Tier 2, Revision 3, Section 15.4, states that the applicant's radiological consequence analyses are based on the NUREG-1465 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, was 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 requested the applicant to review the entire LTR and Section 15.4.4 to ensure that no further discrepancies exist. In its response, the applicant stated that it would revise the LTR and DCD, Tier 2, Section 15.4.4, accordingly. In its LTR Revision 2 and DCD Revision 5, the applicant revised the LTR and DCD, Tier 2, Section 15.4.4, to be consistent with NUREG-1465 and RG 1.183. The staff finds that the applicant's response is acceptable and therefore, RAI 15.4-8 is resolved.

In RAI 15.3-25, the staff requested that the applicant provide complete source term information for the radiological consequence analysis for IEs. In response to RAI 15.3-25, the applicant provided the requested source term information in tabular form, including a complete fission product inventory of the core at 4,590 megawatts thermal, 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 source term information (fission product inventory) in DCD, Tier 2, Section 15.4.4. The applicant incorporated this information in Revision 5 of the DCD. Therefore, based on the applicant's responses, RAIs 15.3-25 and RAI 15.4-9 are resolved.

All of the fission product releases caused by a postulated LOCA are the result of either a containment atmosphere leak through the reactor building (reactor building leakage), a containment atmosphere leak bypassing the reactor building (PCCS leakage), or a main steamline isolation valve leakage bypassing the turbine building (MSIV 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 SRP and RG 1.183 require a radiological consequence analysis for ESF system leakage).

The ESBWR containment consists of a drywell, a wetwell, a PCCS, and supporting systems to remove and control fission product leakage to the environment 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 0.35 percent by weight per day (wt%/d) at the calculated peak containment pressure associated with a LOCA for the entire duration of the accident recovery (30 days). Both the applicant and the staff used this leak rate in their respective radiological consequence analyses. The ESBWR design provides neither an ESF filtration (e.g., charcoal adsorbers) nor a safety-related containment spray system in the containment.

All containment leaks are released into the reactor building except for two potential leak paths that bypass the reactor building. The applicant assumed that a small fraction of 0.35 wt%/d containment leak rate through the PCCS (less than 0.01 wt%/d) into the air space directly above the PCCS and subsequently leak directly to the environment without mixing with the reactor building atmosphere (reactor building bypass). The applicant further assumed that the MSIV leak rate is less than 200 cubic feet per hour, and that leakage is released directly into the environment without mixing with the turbine building atmosphere. These assumed leak rates are used by the applicant and by the staff for the radiological consequence analyses. The feedwater isolation valve lines are located in the main steam tunnel that is open to the turbine building.

In RAI 15.4-11, the staff requested that the applicant include the PCCS leak rate test in a preoperational test program as an ITAAC item and in the TS as surveillance requirements. In response to RAI 15.4-11, the applicant included the PCCS leak rate test in ESBWR Chapter 16, TS Section 5.5.9, and in DCD, Tier 1, Table 2.15.4-1, "ITAAC for the PCCS." Therefore, RAI 15.4-11 is resolved. The ESBWR TS specify the maximum allowable containment and MSIV leak rates and the surveillance requirements.

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

The applicant originally assumed in the ESBWR DCD, Tier 2, Revision 4, that the effective mixing volume of 5.65×10^5 ft^3 will be available for mixing for holdup and decay of fission products before leaking from the reactor building to the environment and that an overall reactor building leakage rate will be less than 50 percent per day. The applicant stated that the reactor building envelope is not intended to provide a leak tight barrier against radiological fission product release; however, the reactor building is capable of periodic testing to ensure that the leakage rates assumed in the radiological consequence analyses is met. The staff requested in

RAIs 15.4-26 and 6.2-165, that the applicant (1) identify the flow paths 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 RAIs 15.4-26 and 6.2-165, the applicant identified the flow paths 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 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.

Subsequently, in DCD Revision 5, the applicant revised the reactor building mixing volume and its leakage rate. The revised effective mixing volume is now 4.11×10^5 ft^3, and its leakage rate is an exfiltration rate of 300 cfm. To justify these changes, the applicant provided an analysis of the reactor building mixing and leakage using the GOTHIC computer code. The staff reviewed the applicant's analysis and accepted the revised effective mixing volume and leakage rate. Section 6.2.3, "Reactor Building Functional Design," of this SER presents the basis for the staff's acceptance. Based on the applicant's responses, RAIs 15.4-26 and 6.2-165 are 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 with airborne fission products from the drywell atmosphere, condense the steam, and return the condensate with condensed fission products to the RPV though the gravity driven cooling system (GDCS) pools. The non-condensables, 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 non-condensables will again become airborne into the wetwell air space and flow back into the drywell during vacuum breaker openings.

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 drainline 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 containment pressure.

The staff assumed, as specified and stipulated in 10 CFR 52.79(a)(1), the postulated LOCA to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities 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. RAI 15.4-7 requested that the applicant provide fission product removal rates in the containment for the entire period of the accident. In its response to RAI15.4-7, the applicant did not provide the information on 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 in Revision 4 of the DCD. Subsequently, in DCD, Tier 2, Revision 5, the applicant provided this information in LTR NEDE-33279P, "ESBWR Containment Fission Product

Removal Evaluation Model," Revision 1. The information provided by the applicant is consistent with the guidance provided in RG 1.183 and therefore, RAI 15.4-7 is resolved.

The applicant assumed leakage of the MSIVs at the TS limit of 0.0623 m^3 per minute total (200 cubic feet per hour).

In RAI 15.4-10, the staff requested whether the MSIV leakage in the turbine building is included in the total containment leakage rate of 0.5 wt%/d. The applicant's response stated that it is not included in the total containment leakage rate of 0.5 wt%/d. Based on the applicant's response, RAI 15.4-10 is resolved. Subsequently, the total containment leakage rate of 0.5 wt%/d was revised to 0.4 wt%/d in DCD Revision 5, and to 0.35 wt%/d in DCD Revision 6.

In RAI 15.4-19 the Staff requested that the applicant include the main steam drain lines along with the main steam lines in the analyses of the loading conditions of the safe-shutdown earthquake (SSE) in the DCD. The main steamlines are classified as seismic Category 1 from the RPV interface to the outboard seismic restraint of the downstream MSIV. The steamlines and their associated branch lines outboard of the last reactor building seismic restraint, including the main steam drainlines, are dynamically analyzed to 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 1, Table 2.11.1-1, now requires the turbine main steam system piping and MSIV fission product leakage path to be able to withstand an SSE without loss of structural integrity. Based on the applicant's response, RAI 15.4-19 is resolved.

15.4.4.3 Staff Evaluation

15.4.4.3.1 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 and exclude "ex-vessel" and "late in-vessel" releases. 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 52.47(a)(2), 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 result only 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 52.47(a)(2). 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 ASTs 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.

15.4.4.3.2 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 PCCS in the containment, the natural deposition of fission product aerosol in the containment, an essentially leak tight 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 52.47(a)(2) and GDC 19.

15.4.4.3.2.1 Primary Containment Atmosphere Leakage

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 1, 2, and 3 discuss the removal of airborne activity from the containment atmosphere. The applicant provided a nonsafety-related containment spray system for accident management following a severe accident as part of the ESBWR fire protection system design. The containment spray system design is not safety-related and is not intended to be used during or following the postulated LOCA. Therefore, radiological consequence assessments give no credit for removing fission products by the containment spray system.

(1) Iodine Removal

The ESBWR passive containment design utilizes a unique PCCS 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 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 and staff assumed substantial meltdown of the core as a result of the postulated LOCA, with subsequent release into the containment of appreciable quantities of fission products as stipulated in 10 CFR 52.47(a)(2) and in 10 CFR 100.21.

Condensation occurring in the PCCS tubes driven by the boiling of water in the reactor vessel provides a very effective means of scrubbing radioactive iodine in the drywell, and in time, most of the drywell iodine will be captured in the PCCS condensate. Since the PCCS condensate drains back into the reactor vessel, most of the iodine 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 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 CsI 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, may 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 reintroduced to the reactor vessel. Therefore, it may be postulated that 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 revolatilization (source) from the reactor vessel and iodine removal by the PCCS condenser and by natural deposition (sink). In its response, the applicant provided an analysis report titled "Iodine Re-Volatilization from the Reactor Pressure Vessel During Late-Stage ESBWR LOCA." The applicant's response addressed the evolution of iodine in the volatile elemental iodine form in the event of a change in pH of the water pool in the RPV from alkaline to acidic conditions in the course of a LOCA. Detailed evaluation of the applicant's response to this RAI by the staff follows in Section (3) below, "Containment Pool Water Chemistry."

(2) Aerosol Removal

Applying credit for aerosol removal through the PCCS requires input from T-H analyses in the containment. 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 have evaluated aerosol removal through well-established models of spray removal or condensation. The ESBWR design 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 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 12.5 hours into a DBA. The values ranged from 0 to 6.5 per hour.

In its independent evaluation of aerosol removal coefficients, 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 Laboratories to evaluate aerosol removal coefficients and to perform quantitative analyses of uncertainties in predicting the aerosol removal rates, both in the containment and the main steamlines. Sandia used a MELCOR ESBWR containment-only model, incorporating the three accident scenarios described in Section 15.4.3.2 above. 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 MELCOR 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.

The staff's contractor used a Monte Carlo method which randomly samples the uncertain parameters. The uncertain parameter distributions were randomly sampled for 150 times. The sampled values were then incorporated into 150 realizations of the containment-only ESBWR MELCOR model. The model results were used to calculate the distribution of aerosol removal rates in the ESBWR containment and the main steamlines.

In its evaluation of aerosol removal rates, the staff at Sandia National Laboratories 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
- multipliers on heat and mass transfer to containment shell

After several discussions between the staff and the contractor, engineering judgment was used in choosing the parameters, as well as identifying the range and distribution of their values.

(3) Containment Pool Water Chemistry

Iodine in the form of CsI is soluble in the containment pool water. Some of it may be converted into the elemental form (I_2), which can be released into the containment atmosphere. The released radioactive elemental iodine may leak out of the containment atmosphere to the reactor building and, subsequently, to the environment. To minimize formation of elemental iodine, the pH of the containment pool water should be kept basic.

The ESBWR design includes three water pools: the PCCS pool in the reactor building 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). In response to this RAI, the applicant provided pH values in each pool following the postulated LOCA for the duration of the entire accident period in VTT Report No. 3. The applicant determined the pH for the various pools inside containment for 30 days after the postulated LOCA for the three accident scenarios described in Section 15.4.3.2 above using input from VTT Technical Research Centre of Finland. VTT used the commercially available Chemsheet code to calculate, among other things, pH values in the containment water pools following a DBA. The staff reviewed the report and finds that the applicant addressed and provided the pH values in each pool following the postulated LOCA for the duration of the entire accident period in VTT Report No. 3. The applicant concluded from this report that the containment pool water pH remains above 7, and therefore, iodine trapped in the pools does not re-evolve into the containment atmosphere. For this reason the staff finds the pH and iodine transport analyses in the report to be acceptable. Based on the applicant's responses, RAIs 15.4-28 and 15.4-29 are resolved. The bases for the staff's acceptance are described below.

The pH of the containment pool 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. The applicant estimated the generation of hydrochloric acid by radiolytic decomposition of cable jacketing using the methodology described in NUREG/CR-5950, "Iodine Evolution and pH Control," issued December 1992. The applicant assumed that 92 percent of the cables reside in the low drywell and the remaining 8 percent of the cables are in the upper drywell. The applicant made a conservative assumption by scaling the hydrochloric acid formation rates by 125 percent.

In RAI 15.4-14, the staff requested that the applicant identify the amount of cable insulation material used in the ESBWR containment and include it in DCD, Tier 1 as an ITAAC item. In response to this RAI, the applicant revised DCD, Tier 1, Revision 4, Section 2.15.1 and Table 2.15.1-1, to include exposed cable mass. Therefore, based on the applicant's response, RAI 15.4-14 is resolved.

Nitric acid is produced by the irradiation of air and water. The applicant used the methodology described in NUREG/CR-5950 to determine the amount of nitric acid in the containment pools. This methodology considers the production of nitric acid to be proportional to the time-integrated radiation dose rate for gamma and beta radiation. The applicant made a conservative assumption by scaling the nitric acid (HNO_3) rates by 125 percent. The applicant made an additional conservative assumption by including the formation of HNO_3 in the water vapor in the containment atmosphere in addition to its formation in the water pools. HNO_3 is a strong acid and will lower the pH.

NUREG-1465 specifies that 5 percent of the total core cesium inventory is discharged to the suppression pool during the gap release phase, and an additional 20 percent is discharged during the early in-vessel phase. In both cases, cesium is released as cesium hydroxide (CsOH) and CsI. The cesium that is not in the form of CsI is assumed to exit the RCS in the form of CsOH. The applicant performed a sensitivity analysis of pH as a function of the amount of CsOH formation (at 100, 50, 25, 10, and 0 percent) to study the effect of uncertainty in cesium formation. The applicant assumed 50-percent cesium formation. CsOH is a strong base and will increase the pH.

Sodium pentaborate is a buffering solution primarily used as a backup means for criticality control within a post accident RPV. Sodium pentaborate is injected directly into the RPV by the SLCS.

$$Na_2B_{10}O_{16} + 16\ H_2O \leftrightarrow 2\ Na^+ + 2\ H_2BO_3^- + 8\ H_3BO_3$$

Since boric acid is a relatively weak acid and sodium hydroxide (formed by the union of a sodium ion and hydroxyl ions) is a strong base, their solution has a buffering effect and will control pH in the containment pools at values higher than 7. The staff considers the buffering action of sodium pentaborate an important factor in enhancing the pH control of containment pools.

To minimize formation of elemental iodine, and thus to prevent its release into the containment atmosphere (and subsequent leakage to the reactor building at a design-basis containment leak rate and then to the environment from reactor building), the pH of the containment pool water must be kept near 7 (neutral) or preferably to basic. VTT Report No. 3 shows that the pH in the RPV becomes acidic at 704 hours, in the low drywell at 603 hours, and in the GDCS at 12 hours. The pH in the wetwell remains permanently at basic. The applicant used these pH values in its determination of aerosol removal rates in the containment in performing the radiological consequence analysis.

For the staff's independent evaluation of the containment pool water pH, the staff contracted with Sandia National Laboratories. Sandia used the iodine pool model developed by their laboratory for the NRC in the MELCOR code to evaluate pH in the containment pools. Based on the findings and conclusions, the staff concluded that the containment pool water pH remains above 7, and therefore, iodine trapped in the water pools does not re-evolve into the containment atmosphere, which confirms the applicant's analysis.

15.4.4.3.2.2 Main Steamline Isolation Valve Leakage

The MSIVs automatically isolate the four main steamlines that penetrate the drywell in the postulated LOCA. Two MSIVs are 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 reactor building 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 standard cubic feet per hour (scfh) specified in the ESBWR DCD TS. The applicant modeled one main steamline with the leak as a single main steamline and combined the three remaining nonleaking main steamlines 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 DCD 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," September 1993. 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 1962.

However, the ESBWR design uses the ASTs to meet the radiological consequence evaluation factors as expressed in TEDE as required by 10 CFR 52.49(a)(1) and 10 CFR 100.21. 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. In response to this RAI, the applicant referenced VTT Report No. 3, which demonstrates that the aerosol removal rates using the MELCOR analysis were higher than those rates using the BWROG methodology. In addition, the applicant stated that it did not claim any credit for aerosol deposition in main steamlines and drainlines being more conservative. The staff finds the BWROG methodology used by the applicant for determining aerosol removal rates is more conservative than the MELCOR analysis used by the staff, and therefore, the applicant's response to be acceptable. Based on the applicant's response, RAI 15.4-22 is resolved.

15.4.4.3.2.3 Reactor Building Leakage

Section 6.2.3, "Reactor Building functional Design," of the ESBWR DCD Tier 2 describes the reactor building functional design, including reactor building leakage.

GDC 16, "Containment Design," states that reactor containment and associated systems shall be provided to establish an essentially leak tight 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 reactor building is not considered to be a leak tight barrier.

The staff considered the applicant's statement with respect to the applicability of GDC 16. The applicant assumed that the reactor building leakage to the environment is no greater than 300 cfm. This assumption directly affects the results of the design-basis radiological consequence analyses required by 10 CFR 52.47 and the control room operator dose stated in GDC 19. The staff requested in RAI 15.4-26 that the applicant provide the method to be used to verify the reactor building leak rate and include the leakage rate as a TS and ITAAC.

In response to RAI 15.4-26, the applicant provided (1) the maximum leak rate that could occur from the reactor building under design-basis conditions (300 cfm) and (2) the method to be used to test reactor building leakage. The leakage rate test is specified in ESBWR TS 3.6.3.1.4 and identified in ESBWR DCD, Tier 1, Section 2.16.5, "Reactor Building," as an ITAAC item.

To justify the 300-cfm leak rate, the applicant provided an analysis of the reactor building mixing and leakage using the GOTHIC computer code. The staff reviewed the applicant's analysis and accepted the revised effective mixing volume and leakage rate. Section 6.2.3 of this report presents the bases for the staff's acceptance. Based on the applicant's response, RAI 15.4-26 is resolved.

15.4.4.3.2.4 Control Room Radiological Consequence Analysis

In DCD, Tier 2, Section 15.4, 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 of this SER, "Habitability Systems," describes the staff's review and evaluation of the CREFU in more detail.

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 control room. Subsequently, in DCD, Tier 2, Revision 3, the applicant changed its ESBWR control room design to provide the CREFU as an active containment ESF atmosphere cleanup filtration unit, designed to remove fission products from the control room habitability area and to pressurize the control room with nonradioactive air from outside following postulated DBAs. The CREFU is a safety-related system and a subsystem of the control building HVAC system located in the control building; it 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. In RAI 15.4-27, the staff indicated that it is aware of possible design changes that include the EBAS and requested that the applicant state whether the design changes are complete. The applicant responded that the EBAS is no longer applicable to the ESBWR design. Based on the applicant's response, RAI 15.4-27 is resolved.

Section 6.4, "Habitability Systems," of ESBWR DCD, Tier 2, Revision 6, describes the CREFU design, and Section 6.4 of this report provides the staff's evaluation. The applicant assumed an unfiltered air in-leakage rate of 5.66 L/s (12 cfm) in DCD, Tier 2, Table 15.4-5, in its control room radiological consequence analysis. In RAI 15.4-30, the staff requested the applicant include the preoperational testing of assumed control room unfiltered air in-leakage rate in DCD, Tier 1, Table 2.16.2-1 as an ITAAC item and in DCD, Tier 2 Chapter 16, Section 3.7.2 as a TS surveillance requirement in accordance with guidance provided in Technical Specification Task Force (TSTF)-448, "Control Room Habitability," dated July 1, 2003.

In Revision 4 of the ESBWR DCD Tier 1, the applicant specified the testing of assumed control room unfiltered air in-leakage rate in Table 2.16.2-6 as an ITAAC item and included its surveillance requirement in ESBWR DCD, Tier 2, Revision 4, Chapter 16, Section 5.5.12. The staff finds the response to RAI 15.4-30 to be acceptable and 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 control room χ/Q values at the COL stage. In RAI 2.3-9, the staff asked the applicant to provide figures showing control room 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 control room necessary to evaluate the radiological consequences. In response to this RAI, the applicant included the requested information in DCD Revision 5. Based on the applicant's response, RAI 2.3-9 is resolved.

In Revision 3 to the DCD, the applicant revised the control room χ/Q values in DCD, Tier 1, Table 5.1-1, and Tier 2, Table 2.0-1, by listing them as standard plant site design parameters. Two sets of control room χ/Q values are provided for the reactor building, PCCS/reactor building roof, and turbine building release pathways; the first 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 control room χ/Q values it used for the control room radiological consequence analysis and why. In response to this RAI, the applicant provided the requested information in DCD, Tier 2, Revision 4. Based on the applicant's response, RAI 15.4-31 is resolved.

In DCD, Tier 2, Section 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. After performing an independent radiological consequence dose calculation, the staff finds that the ESBWR control room design meets the 0.05 Sv (5 rem) TEDE criterion in GDC 19 for the postulated LOCA.

15.4.4.3.2.5 Technical Support Center Radiological Consequence Analysis

The technical support center (TSC) provides an area and resources for use by the applicant to provide plant management and technical support to the reactor operating personnel located in the control room in the event of an emergency. The TSC relieves the reactor operator peripheral duties and communications not directly related to reactor operations and prevents congestion in the MCR.

The TSC is a required facility specified by the NRC regulation, 10 CFR Part 50, Appendix E, Section IV.E.8, as it relates to providing emergency facilities and equipment for use in an emergency. 10 CFR Part 50, Appendix A, GDC 19 requires the applicant to provide equipment at appropriate locations outside the control room with a design capability for prompt hot shutdown of the reactor and with a potential capability for subsequent cold shutdown of the reactor. Its functional criteria are specified in NUREG-0696, "Functional Criteria for Emergency Response Facilities," and the radiological acceptance criterion is specified in NUREG-0737, Supplement No. 1, "Clarification of TMI Action Plan Requirements."

NUREG-0737 requires, among other things, radiological protection to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole-body, or its equivalent to any part of the body, for the duration of an accident. The SRP Section 15.0.3 states that the radiation protection design of the TSC is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criteria specified for the control room of 5 rem TEDE for the duration of an accident.

In Section 13.3, "Emergency Planning," of the ESBWR DCD Revision 6, the applicant describes the TSC design requirements and the staff evaluated it in Section 13.3, "Emergency Planning," of this SER. The applicant stated among other things, that the TSC is provided with radiological protection and monitoring equipment necessary to ensure that the radiation exposure to any person working in the TSC would not exceed 5 rem TEDE for the duration of the accident. The staff audited the applicant's dose calculations and performed an independent TSC dose calculation generating the same results. Therefore, the staff finds that the TSC radiological consequence analysis provided in the ESBWR DCD is acceptable

15.4.4.3.2.6 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 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 SER provides the staff's evaluation of the hypothetical reference χ/Q values used for the control room radiological consequence evaluation. The staff will review site-specific χ/Q values for a COL application that references the ESBWR design. If site-specific χ/Q values exceed the referenced χ/Q values used in this evaluation (e.g., poorer dispersion characteristics), a COL applicant may need to consider compensatory measures, such as increasing the size of the site or providing additional ESF systems to meet the relevant dose limits given in 10 CFR 52.47 and GDC 19.

15.4.4.4 Conclusion

The staff performed an independent confirmatory dose calculation and found that the staff's results agree with the applicant's values. Table 15.4-2 provides the major parameters and assumptions used by the staff for the LOCA radiological consequence analysis. Both the applicant's and the staff's results meet the relevant dose acceptance criteria for the EAB, LPZ, and control room. Therefore, the staff concludes that the ESBWR design, is bounded by the hypothetical χ/Q values proposed by the applicant, will provide reasonable assurance that the radiological consequences of a LOCA at the EAB and LPZ will be within the dose criteria set forth in 10 CFR 52.47(a)(2) (i.e., 0.25 Sv (25 rem) TEDE) and that the radiological consequences to an individual in the control room as a result of a postulated LOCA will be within the dose criterion established in GDC 19 (0.05 Sv (5 rem) TEDE). Therefore, the staff finds the radiological consequence analysis provided by the applicant to be acceptable.

15.4.5 Main Steamline Break Outside Containment

15.4.5.1 Regulatory Criteria

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

The staff evaluated 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 preaccident iodine spike at the EAB for any 2-hour period following the onset of the postulated fission product release. The staff used 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, in accordance with GDC 19 of Appendix A to 10 CFR Part 50.

15.4.5.2 Summary of Technical Information

The applicant postulated that one of the four main steamlines will rupture between the containment outer isolation valve and the TCV. 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 outermost MSIV. 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 15.4-13. The main MSIVs are assumed to isolate the break within 5 seconds, as specified in the ESBWR DCD, Tier 2, TS 3.6.1.3. The staff assumed the duration of this event to be 5.5 seconds, which includes an additional 0.5 seconds for MSIV response time. No other release mitigation (i.e., plateout, holdup, dilution) is assumed, and no fuel damage is projected to occur. The only radioactivity available for release from this event is the activity which was in the reactor coolant and steamlines during the normal plant operation before the break.

Following isolation of the main steam supply system (i.e., MSIV closure) ADS initiates depressurization. Once the reactor system has been depressurized, the GDCS automatically begins reflooding the reactor vessel, and therefore, no fuel damage is projected to occur. The radioactivity in the released coolant is assumed to be released to the environment instantaneously from the turbine building as a ground-level release.

The applicant concluded in DCD, Tier 2, Revision 3, that no more than 8.2328×10^4 kilograms (kg) (181,339 pounds mass (lbm)) of reactor coolant will be lost through the break before automatic isolation and that less than 4.705×10^3 kg (103,634 lbm) of that will be lost as steam. In RAI 15.4-2, the staff requested the source term information used in the MSLB accident analysis. In its response to RAI 15.4-2, the applicant stated that it is revising the MSLB event to determine exact mass release values. In RAI 15.4-2S01, the staff requested that the applicant provide, among other things, revised steam and water mass releases for the MSLB accident. In its response to RAI 15.4-2 S01, the applicant provided the revised steam and water mass releases stating that it will include the revised radiological consequence analysis of this event in its forthcoming Revision 5 to the ESBWR DCD Tier 2.

In DCD, Tier 2, Revision 5, the applicant concluded that no more than 45,593 kg (101,513 lbm) of reactor coolant will be lost through the break before automatic isolation and that less than 21,084 kg (46,482 lbm) of that will be lost as steam. In DCD, Tier 2, Revision 5, the applicant evaluated the dose to operators in the control room. 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 be isolated during this event and the CREFU is credited for removing fission products. The applicant assumed an in-leakage rate of 5.66 L/s (12 cfm) of unfiltered air into the control room envelope. Based on the applicant’s response, RAI 15.4-2 is resolved.

15.4.5.3 Staff Evaluation

The staff performed an independent radiological consequence dose calculation for the two scenario cases for the MSLB accident described below.

For Case 1, the staff assumed 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 staff assumed that the ESBWR reactor was operating at 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 will result in an increased concentration in the primary coolant during the course of the accident.

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

For both cases, the staff's independent radiological consequence dose calculation confirmed the applicant's assertion that a postulated MSLB accident meets the dose criterion provided in SRP Section 15.0.3 and in RG 1.183 at the EAB and LPZ, as well as the GDC 19 criterion of 0.05 Sv (5 rem) TEDE for the control room.

15.4.5.4 Conclusion

The staff performed an independent confirmatory dose calculation and found that its results agree with the applicant's values. Both the applicant and the staff's results meet the relevant dose acceptance criteria at the EAB, LPZ, and control room.

Therefore, the staff concludes that the ESBWR design, as bounded by the applicant's proposed hypothetical χ/Q values, will provide reasonable assurance that the radiological consequences of an MSLB accident at the EAB and LPZ will be within the dose 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 preaccident iodine spike. Furthermore, the radiological consequences to an individual in the control room as a result of a postulated MSLB accident will be within the dose criterion set forth in GDC 19 of 0.05 Sv (5 rem) TEDE. Therefore, the staff finds the applicant's radiological consequence analysis to be acceptable.

15.4.6 Control Rod Drop Accident

As stated in Section 4.6 of this report the staff accepted the FMCRD as a system for which a CRDA is a very unlikely event; therefore, radiological analysis is not required.

15.4.6.1 Regulatory Criteria

As indicated in Section 15.4.0 of this evaluation the staff used SRP Section 15.4.9, "Spectrum of Rod Drop Accidents (conventional BWRs)," and Section 4.2, Appendix B, Revision 3, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents" to provide the interim acceptance criteria and guidance for the RIA. RIAs consist of postulated accidents that involve a sudden and rapid insertion of positive reactivity. This accident scenario includes a CRDA for BWRs. The uncontrolled movement of a single control rod out of the core results in a positive reactivity insertion which promptly increases local core power. Fuel temperatures increase rapidly, prompting fuel pellet thermal expansion. The reactivity excursion is initially mitigated by Doppler feedback and delayed neutron effects followed by reactor trip.

Fuel Cladding Failure Criteria

(1) The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 calories per gram (cal/g) for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g per fuel rod with an

internal rod pressure exceeding system pressure. For intermediate (greater than 5-percent rated thermal power) and full-power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., CPR).

(2) The pellet/cladding mechanical interaction failure criteria are a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in SRP Section 4.2, Appendix B, Figure B-2 (BWR).

Core Coolability Criteria

Fuel rod thermal-mechanical calculations, employed to demonstrate compliance with Criteria 1 and 2 below, must be based on design-specific information accounting for manufacturing ranges and modeling uncertainties using NRC-approved methods including burnup enhanced effects on pellet power distribution, fuel thermal conductivity, and fuel melting temperature.

(1) Peak radial average fuel enthalpy remain below 230 cal/g.

(2) Peak fuel temperature must remain below incipient fuel melting conditions.

(3) Mechanical energy generated as a result of (a) nonmolten fuel-to-coolant interaction and (b) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.

(4) There must be no loss of coolable geometry as a result of (a) fuel pellet and cladding fragmentation and dispersal and (b) fuel rod ballooning.

Fission Product Inventory

The total fission-product gap fraction available for release following any RIA would include the steady-state gap inventory (present before the event) plus any fission gas released during the event. The steady-state gap inventory would be consistent with the non-LOCA gap fractions cited in RG 1.183 (Table 3) and RG 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors” (Table 2) and would depend on operating power history. Whereas diffusion governs the fission gas release into the rod plenum during normal operation, pellet fracturing and grain boundary separation are the primary mechanisms for fission gas release during the transient.

15.4.6.2 Technical Information

Section 4.6 of this report discusses the FMCRD system design features provided to reduce the occurrence of CRDAs.

In DCD Section 15.4.6.2, the applicant listed the following highly unlikely events for postulating a CRDA:

- The reactor is at less than 5 percent power.

- Failures of both safety-related separation detection devices or failure of the rod block interlock occurs.

- The latch mechanism fails.

- A simultaneous additional failure causes the occurrence of a stuck rod on the same FMCRD.

- The control rod is withdrawn without the operators noticing that the control rod withdrawal did not result in a neutron flux increase.

- The stuck rod has to become unstuck.

15.4.6.3 Staff Evaluation

Based on the design features, the applicant believes that the ESBWR design incorporates sufficient safeguards to negate its susceptibility to excess reactivity events. Initially, the ESBWR DCD did not include design requirements or a CRDA analysis. The staff was concerned that several scenarios might lead to an excess reactivity event and that each scenario would require exploration to ensure that it was not beyond design basis. If any scenario were to be credible, acceptance criteria (e.g., coolability, radiological consequences) would need to be developed and an acceptable accident analysis performed to demonstrate that these criteria were satisfied. The inclusion of this family of accidents may involve changes to the proposed ESBWR TS (e.g., LCOs, ESF actuation system setpoints) and the ESBWR DCD (e.g., Sections 4.2, 4.6, and 15).

In RAI 4.6-23 and RAI 4.6-23 S01, the staff requested the applicant to describe any enhanced features (with respect the ABWR design) or design requirements developed for the ESBWR to minimize the probability of an excess reactivity addition event. The staff also requested the applicant perform a Failure Modes and Effects Analysis to discuss the probability and potential consequences for each scenario leading to an excess reactivity event. The staff reviewed the control rod drop event frequency estimates provided by GEH in response to RAI 4.6-23 and RAI 4.6-23 S01. The design and testing of the control rod and CRD mechanism include a number of diverse and redundant features for preventing a rod drop event, which is an indicator of high reliability in the design. Based on its review of key design and operational features and the applicant's fault-tree analysis, the staff concludes that GEH has provided a reasonable estimate of the rod drop frequency.

In RAI 4.6-23 S02, the staff requested the applicant demonstrate compliance with GDC-28 and guidance provided in SRP Section 4,2, Appendix 4B. The staff also considered the applicant's control rod drop event frequency evaluation and regulatory requirements provided in response to RAI 4.6-23 S02. Based on the potential consequences of an unrestricted reactivity excursion and to ensure compliance with GDC 28, "Reactivity Limits," the staff concludes that the ESBWR design must demonstrate RCPB integrity and acceptable radiological consequences for the CRDA, irrespective of the probability of a CRDA.

GEH provided the CRDA analyses in the response to RAI 4.6-23 S02 and RAI 4.6-38. GEH utilized a combination of the nuclear core simulator, PANACEA, and the T-H code, TRACG, for the analyses.

Compliance with GDC 28 is demonstrated by analysis of the consequences of a postulated CRDA. The staff notes significant conservatism in the analysis. In particular, the adiabatic assumption precludes any void formation (which would insert negative reactivity during the accident). Also, the calculations assumed that the worth of the dropped rod, regardless of its position during the startup withdrawal sequence, is added to a critical reactor.

The analysis appropriately assumed that the control rod is dropped from its full inserted position to the position of the drive and explicitly accounted for the effects of exposure.

The staff notes that the calculation did not include either operator error or calculational biases and uncertainties. The staff, however, has reviewed the applicability of PANAC11 to evaluating nuclear characteristics for the ESBWR. The staff found that PANAC11 is suitable for calculations of blade worth for the ESBWR. The staff has approved previous versions of PANACEA to provide control blade worth and control rod drop shape information to downstream transient evaluations. Therefore, the staff is reasonably assured that the calculations are indicative of the expected ESBWR behavior.

The staff found that the low enthalpy rises are a result of low blade worth (less than 80 cents in all cases). Therefore, the staff finds that the calculational results indicating large margin are expected. The staff is reasonably assured that consideration of modeling biases, uncertainty, and operator error would not result in changes to the analytic result on the order of magnitude of the available margin. The large margins to cladding failure for the ESBWR initial core provide the staff with reasonable assurance that, for the core design described in the DCD, the radiological consequences are bounded by the DCD analyses and that barrier integrity has been demonstrated.

NUREG-0471, “Generic Task Problem Descriptions (Categories B, C, and D),” issued June 1978 (ADAMS Legacy Library Accession No. 7811270090, Microfiche Address 94439:153, 94439:289), describes Unresolved Safety Issue (USI) D-3, “Control Rod Drop Accident.” This issue is an ACRS generic concern which involves assessing the uncertainties in calculations of the CRDA, including the choice of a negative reactivity insertion rate due to a scram and the potential difference between a two-dimensional calculation and a three-dimensional calculation. The response to RAI 4.6-38 refers to the analysis performed in response to RAI 4.6-23 S02. The response briefly describes a reload licensing screening approach, analysis procedures, and analytical results. The applicant performed the analyses using the PANAC11 (PANACEA version 11) three-dimensional simulator in a transient mode with six delayed neutron groups. PANAC11 calculates the fuel enthalpy rise according to an adiabatic model (by integrating transient power) and explicitly accounts for blade worth, nominal blade pull during startup, and radial power shapes. Section 4.3 of this report provides a detailed evaluation of the reactivity aspects.

The staff concluded that GEH followed the SRP Section 4.2, Appendix B, interim acceptance criteria and analyzed the CRDA. Based on the applicant’s response, RAIs 4.6-23 and 4.6-38 are resolved.

Since this accident does not result in any fuel failures or release of any primary coolant to the environment, core coolability and fission product criteria do not apply.

15.4.6.4 Conclusion

The staff concludes that the rod drop accident analysis is acceptable and meets the requirements of GDC 13 and 28. This conclusion is based on the following findings:

- The applicant met GDC 13 requirements by demonstrating that all credited instrumentation was available and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

- The applicant met GDC 28 requirements by providing reactivity control systems features which mitigate postulated reactivity accidents that could result in damage to the RCPB greater than limited local yielding or damage that impairs core cooling capability significantly.

The staff has evaluated the applicant's analysis of the assumed CRDA and finds the assumptions, calculation techniques, and consequences acceptable. Because the calculations predict peak fuel temperatures below melting conditions, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten U02 presumably did not occur. The pressure surge results in a pressure increase below Service Limit C as defined in Section III of the ASME Boiler and Pressure Vessel Code (SRP Section 15.4.9-6, Revision 3—March 2007) for the maximum control rod worths assumed. The staff believes that the calculations are sufficiently conservative in both the initial assumptions and analytical models to maintain primary system integrity. Section 21.6 of this report provides additional information for the use of PANAC11.

15.4.6.5 Post-COL Activity

For use in assessing ESBWR reload cores, GEH has developed a conservative criterion for the maximum static control blade worth below which the enthalpy rise curve in Appendix B of Revision 3 to SRP Section 4.2 would not be exceeded. This criterion will be applied to future ESBWR reload cores to determine whether additional calculations are needed. Only if necessary, will the enthalpy rises be calculated using a conservative adiabatic methodology or a best-estimate methodology that has been approved by the NRC.

In accordance with TS 5.6.3, Item C and as discussed in response to RAI 4.6-23 S02, licensees will perform cycle-specific confirmatory evaluations based on an NRC-approved or NRC accepted method for reload cores to ensure that all requirements pertaining to a postulated CRDA are met.

15.4.7 Feedwater Line Break Outside Containment

Staff evaluation of this Section is included in Section 15.4.0 of this report.

15.4.8 Failure of Small Lines Carrying Primary Coolant outside Containment

15.4.8.1 Regulatory Criteria

The staff reviewed DCD, Tier 2, Section 15.4.8, "Failure of Small Lines Carrying Primary Coolant Outside Containment," in accordance with guidance provided in 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 neither provides guidance nor lists this event as a DBA. The staff considers the radiological consequence resulting from this event to be bounded by that resulting from the MSLB accident outside containment for all light-water BWRs.

The staff evaluated 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 used a criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences in the control room of the ESBWR design, in accordance with GDC 19.

GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," contains a provision to ensure isolation of all pipes that are part of the RCPB 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.

15.4.8.2 Summary of Technical Information

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 of the containment, but inside of the reactor building at a location where it may not be isolated automatically for 30 minutes at normal reactor operating temperature and pressure. The applicant assumed that, 30 minutes after initiation of this event, the operator will detect the pipe break, scram the reactor, and initiates reactor depressurization. The applicant assumed the duration of this event to be 5.9 hours (0.5 hours to detect and 5.4 hours to depressurize the reactor). The applicant presented its analyses of the radiological consequences of a postulated small line break (SLB) accident outside containment but inside the reactor building in DCD, Tier 2, Section 15.4.8 and Tables 15.4-17 through 15.4-19.

The applicant estimated that 1.48×10^4 kg (32,595 lbm) of primary coolant will be released through the break before it is isolated until the reactor is depressurized and that 4.0×10^3 kg (8,834 lbm) 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 reactor building to the environment without any mitigation. Furthermore, the applicant assumed that the iodine in the primary coolant was at the maximum equilibrium limit of 0.148 MBq/g (4 µCi/g) for DEI-131, as specified in the ESBWR TS.

The applicant evaluated the radiological consequence doses at the EAB, LPZ, and reactor operators in the control room. The applicant assumed that the control room will be isolated during this event and the CREFU will be operational to remove fission products. The applicant used an in-leakage rate of 5.66 L/s (12 cfm) of unfiltered air into the control room envelope. The applicant analyzed the control room dose over a 30-day period. The resulting 30-day TEDE to an individual in the control room is less than the GDC 19 dose criterion.

In RAI 15.4-3, the staff requested that the applicant (1) state if applicant has taken any exceptions to the guidance provided in SRP Section 15.6.2, (2) provide steam and water break flow rates and reactor building leak rate used in dose calculation, and (3) provide a copy of dose calculation performed. In response to RAI 15.4-3, the applicant stated that it did not take any exceptions to the guidance provided in SRP Section 15.6.2 and provided information requested in items (2) and (3) above. The staff performed an independent dose calculation using the information provided by the applicant and confirmed the applicant's results meeting the dose acceptance criteria specified in SRP Section 15.0.3.

In RAI 15.4-3 S01, the staff requested that the applicant add the duration of the event, fission product release point, and site boundary and control room atmospheric dispersion values used in Table 15.4-17. In Table 15.4-17 of DCD Revision 5, the applicant provided the information requested in RAI 15.4-3 S01. The staff finds that the estimated duration of this event is consistent with the guidance provided in SRP Section 15.6.2 and the control room atmospheric dispersion values used are the same as those provided in DCD, Tier 1, Table 5.1-1, and DCD, Tier 2, Table 2.0-1. The applicant added the fission product release point in Table 15.4-17 of the DCD Revision 5. Therefore, based on the applicant's response, RAI 15.4-3 is resolved.

15.4.8.3 Staff Evaluation

While performing past licensing reviews, such as those for the AP600, AP1000, and ABWR, the staff determined that an SLB 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) and 10 CFR 52.47(a)(1). Furthermore, the staff believes that the radiological consequences resulting from this event are bounded by those resulting from the MSLB and the RWCU line failure outside containment. However, the staff performed an independent radiological consequence dose calculation for this event and confirmed the applicant's assertion that a postulated SLB accident indeed meets the dose criteria in SRP Section 15.0.3 and RG 1.183 at the EAB and LPZ, as well as the GDC 19 criterion of 0.05 Sv (5 rem) TEDE for the control room.

15.4.8.4 Conclusion

The staff performed an independent confirmatory dose calculation and found that its results agree with the applicant's values. Both the applicant's and the staff's results meet the relevant dose acceptance criteria for the EAB, LPZ, and control room.

Therefore, the staff concludes that the ESBWR design, as bounded by the applicant's proposed hypothetical χ/Q values, will provide reasonable assurance that the radiological consequences of an SLB accident at the EAB and LPZ will be within the dose criterion specified in SRP Section 15.0.3 and RG 1.183 of 0.025 Sv (2.5 rem) TEDE. Furthermore, the radiological consequences to an individual in the control room as a result of a postulated MSLB accident will be within the dose criterion set forth in GDC 19 of 0.05 Sv (5 rem) TEDE. Therefore, the staff finds the applicant's radiological consequence analysis to be acceptable.

15.4.9 RWCU/SDC Line Failure Outside Containment

15.4.9.1 Regulatory Criteria

Neither SRP Section 15.0.3 nor RG 1.183 lists this event as a DBA; therefore, the NRC does not require that it 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 failure of the RWCU system line outside of containment for the radiological consequences as a DBA for the ABWR. Therefore, the applicant analyzed, and the staff reviewed, the radiological consequences of this event for the ESBWR using the guidance provided in SRP Section 15.0.3 and RG 1.183 for the MSLB accident as a substitute for the failure of the RWCU system line outside of containment event.

The staff evaluated the radiological consequences of this event 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 preaccident iodine spike at the EAB for any 2-hour period following the onset of the postulated fission product release. The staff also used a criterion of 0.05 Sv (5 rem) TEDE to evaluate the radiological consequences in the control room of the ESBWR design, in accordance with GDC 19.

15.4.9.2 Summary of Technical Information

In DCD, Tier 2, Section 15.4.9 and Tables 15.4-20 through 15.4-23, the applicant presented its analyses of the radiological consequences of a postulated RWCU 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 (a 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 would result as a consequence of this event. The only radioactivity available for release from this event is the activity which was in the reactor coolant and the RWCU system during the normal plant operation before the break. The applicant limited the initial break flow rate to 2.218×10^3 kg/s, assuming two-phase critical flow for limiting diameter piping inside containment. The applicant further assumed that no more than 1.33×10^5 kg of reactor coolant would be lost through the break before automatic isolation and that less than 5.0×10^4 kg of that would be lost as steam.

In RAI 15.4-4, the staff requested that the applicant (1) state if any operator actions are credited in the event of a RWCU/SDC system line failure, (2) provide the break flow rate and break flow duration used in the radiological consequence dose calculation, and (3) provide a copy of dose calculation performed. In response to RAI 15.4-4, the applicant stated that it did not credit operator actions and provided the break flow rate and break flow duration used in the radiological consequence dose calculation and a copy of dose calculation performed. The staff performed an independent dose calculation using the information provided by the applicant and confirmed the applicant’s results meeting the dose acceptance criteria specified in SRP Section 15.0.3.

In RAI 15.4-4 S01, the staff requested that the applicant add (1) the duration of this event, (2) fission product release point, and (3) site boundary and control room atmospheric dispersion values used by the applicant to Table 15.4-21. In response to RAI 15.4-4 S01, the applicant revised Table 15.4-4 and added the requested information in Table 15.4-21 of DCD Revision 5. Therefore, based on the applicant’s response, RAI 15.4-4 is resolved.

The applicant evaluated the radiological consequence doses at the EAB, LPZ, and to operators in the control room. The applicant assumed that the control room will be isolated during this event and the CREFU will be operational. The applicant used an in-leakage rate of 5.66 L/s (12 cfm) of unfiltered air into the control room envelope. The applicant analyzed the control room dose over a 30-day period. The resulting 30-day TEDE to an individual in the control room is less than the GDC 19.

15.4.9.3 Staff Evaluation

The staff provided no specific regulatory guidance for evaluating the radiological consequences for this event in RG 1.183 or in SRP Section 15.0.3. Therefore, the staff reviewed this event using the guidance provided for the MSLB accident in SRP Section 15.0.3 and RG 1.183.

The staff performed an independent radiological consequence dose calculation for the following two scenario cases for this event. In Case 1, the staff assumed 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 staff assumed that the ESBWR reactor was operating at 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 was assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate will lead to an increasing concentration of DEI-131 in the primary coolant during the course of the accident. In Case 2, the staff assumed 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 staff's independent radiological consequence dose calculation for this event confirmed the applicant's assertion that a postulated SLB accident meets the dose criteria provided in SRP Section 15.0.3 and RG 1.183 at the EAB and LPZ, as well as the GDC 19 limit of 0.05 Sv (5 rem) TEDE for the control room.

15.4.9.4 Conclusion

The staff performed an independent confirmatory dose calculation and found that its results agree with the applicant's values. Both the applicant's and the staff's results meet the relevant dose acceptance criteria for the EAB, LPZ, and control room.

Therefore, the staff concludes that the ESBWR design, as bounded by the applicant's assumed χ/Q values, will provide reasonable assurance that the radiological consequences of this event at the EAB and LPZ will be within the dose 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 preaccident iodine spike. Furthermore, the radiological consequences to an individual in the control room as a result of this event will be within the dose criterion set forth in GDC 19 of 0.05 Sv (5 rem) TEDE. Therefore, the staff finds the applicant's radiological consequence analysis to be acceptable.

15.4.10 Spent Fuel Cask Drop Accident

Staff evaluation of this Section is included in Section 15.4.0 of this report.

15.5 <u>Special Events</u>

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:

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

- events that include a common-mode equipment failure or additional failures beyond the single-failure criterion

The applicant analyzed special events in DCD, Tier 2, Chapter 15. 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 evaluation of RPV overpressure protection.

15.5.2 Shutdown without Control Rods

The 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

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 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, or containment conditions. For a conventional BWR, AOOs with failure to scram could lead to unacceptable conditions, such as closure of the 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 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 USI A-9. In 1986, the agency resolved USI A-9 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 (Volume 49, page 26036) present the bases for current regulatory requirements related to ATWS, including the associated regulatory evaluation.

15.5.4.1 Acceptance Criteria

The provisions of 10 CFR 50.62 specify the prescriptive requirements for ATWS. This regulation requires BWRs to have the following mitigating features for an ATWS event:

- an 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

- an 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 the applicant had provided comparable actions.

The staff also compared BWR performance during an ATWS 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 1,500 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 Section 15.8 provided guidance for the staff's review of BWR ATWS. SRP Section 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," which requires 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, “Containment Design,” 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, dated 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 and 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 feedwater 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, feedwater runback, ADS inhibits, and the SLCS. The following are the initiation signals and setpoints for the above responses:

- For ARI and FMCRD run-in, the following apply:

  – high pressure, or

- Level 2, or
- either RPS scram command or SCRRI/SRI command and elevated power levels exist after time delay
- manual

- For SLCS initiation, the following apply:
  - high pressure and SRNM ATWS permissive for 3 minutes, or
  - Level 2 and SRNM ATWS permissive for 3 minutes, or
  - manual ARI/FMCRD run-in signals and SRNM ATWS permissive for 3 minutes

- For feedwater runback, the following apply:
  - 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, the following apply:
  - high pressure and APRM not downscale for 1 minute, or
  - Level 2 and APRM not downscale
  - MCR controls manually inhibit the ADS under ATWS condition

- For HPCRD, the following apply:
  - Level 2 with maximum 10-second delay
  - Level 2 with maximum 145-second delay during loss of offsite power

- For IC, the following apply:
  - closure of MSIV
  - high pressure for 10 seconds
  - Level 2 with 30-second 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 for the 251-inch RPV (i.e., 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 the ATWS rule, as described in Section 9.3.5 of this report, and concluded that the applicant had satisfied these requirements.

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 CP after July 26, 1984. Section 9.3.5 of this report provides a detailed evaluation of the SLCS.

GDC 26, 27, and 28 require the SLCS, which is described in Section 9.3.5 of this report. Because the new CRD design eliminates the previous common-mode failure potential and there is a very low probability of simultaneous common-mode failures of a large number of FMCRDs, the NRC staff considers a failure to achieve shutdown to be unlikely. The staff believes that the provisions of the ATWS rule continue to require the SLCS. In addition, the ESBWR incorporates automatic initiation of the SLCS under conditions indicative of an ATWS to meet the rule specified at 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 for 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 reflects the BWR use of forced core flow circulation. Because the ESBWR uses natural circulation, there are no recirculation pumps to be tripped. Hence, the ESBWR cannot implement recirculation pump trip (RPT) logic.

The ESBWR does implement an ATWS automatic feedwater runback feature, 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 feedwater 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 CRDS.

The use of this design reduces the common-mode failure potential 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, thus satisfying the intent of the ATWS rule.

The staff issued RAI 15.5-5 regarding the manner in which the applicant credited operator actions. 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 noted that the TRACG analysis of ATWS MSIV closure transient response evaluation assumes operator action to achieve the following:

(1) maintenance of level at TAF + 5 feet (1.524 meters) after the initial automatic feedwater runback

(2) depressurization of the reactor, if the heat capacity temperature limit curve is reached

Since the above operator actions are consistent with the operator actions specified in Emergency Procedure Guidelines (EPG), the staff agrees with this response. Based on the applicant's 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 the ASME Code Service Level C limits of 120 percent of the RPV design peak pressure of 1,500 pounds per square inch. The applicant performed this analysis using the TRACG systems code. The NRC staff reviewed 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." Section 21.6 of this report, as well as the staff's SER with open items, provides the staff's evaluation of NEDE-33083P, Supplement 2.

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 of this report, the applicant analyzed 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 Feedwater 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 Feedwater 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-This event is not limiting but still analyzed for completeness

- MSIV Closure without Scram in Combination with the ARI, FMCRD Run-in-Failure, and Automatic SLCS—This is the most limiting event. The results indicate that the peak calculated reactor pressure of 1,390 psig, containment pressure of 30 psig, suppression pool temperature of 163 degrees F, and peak cladding temperature of 1,702 degrees F are all within the acceptance criteria and hence are acceptable.

The staff completed 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 used the Monte Carlo N-Particle Transport Code and a flux-squared, adjoined weighting factor to determine core criticality in situations with limited boron distribution. Section 21.6 of this report also discusses these calculations.

Feedwater Water Temperature Operating Domain

The applicant has analyzed the impact of the proposed P-FWTOD extension on the limiting ATWS events. NEDO-33338, Revision 1, specifically calculates the following ATWS events at off-nominal FWT:

(1) MSIV closure
(2) loss of condenser vacuum

The applicant analyzed these two ATWS events at the reduced FWT point (SP1) and increased FWT point (SP2) for the initial core design. In addition, GEH analyzed MSIV closure ATWS at both SP1 and SP2 conditions for the equilibrium cycle. These analyses used standard ATWS assumptions, with a combination of nominal and bounding inputs. The results show sufficient margin to ATWS criteria, including peak reactor pressure, peak clad temperature, suppression pool temperature, and containment pressure.

15.5.4.5 USI A-9—Anticipated Transient without Scram

The staff published its technical findings regarding this USI in Volume 4 of NUREG-0460, and the publication of the ATWS rule resolved the issue. The ESBWR design meets the ATWS rule, and hence this issue is resolved.

15.5.4.6 Conclusion

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 risk assessments that demonstrate that it has considered limiting ATWS transient and event sequences. Based on this information, the applicant has determined that features included in the design, in accordance with the ATWS rule, result in reasonable assurance, based on a low estimated frequency of occurrence, and 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.

15.5.4.7 Post-COL Activity

SRP Section 15.8, Section III.4.C states that: on a cycle specific basis, the licensee must confirm that the ATWS analysis of record, based on new fuel design or power-density change, bounds the plant-specific core configuration. This is covered by TS 5.6.3, Item C.

15.5.5 Station Blackout

As required by 10 CFR 50.63, “Loss of All Alternating Current Power,” each light-water-cooled nuclear power plant must be able to withstand and recover from an 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 must be capable of maintaining core cooling and appropriate containment integrity. The rule also identifies the factors that must be considered in specifying the SBO duration.

15.5.5.1 Acceptance Criteria

The provisions of 10 CFR 50.63(a) (2) requires the following:

> [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. In Section 15.5 of the ESBWR DCD the applicant addressed containment integrity in terms of design limits on pressures and temperatures. In RAI 15.5.-8, the staff requested that the applicant add a discussion to this section 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.

In response to RAI 15.5-8, the applicant revised DCD Section 15.5.5.1 in Revision 5 to state that, “SBO requirements related to the required power for valve position indication and containment isolation closure verification are met.” All containment isolation valves are safety-related. All containment isolation valve position indications are supplied from the safety-related dc system through the uninterruptible power supply, and hence the position indications are available to the operator at the onset of an SBO. Based on the applicant’s response, RAI 15.5-8 is resolved.

The staff performed its systems review of SBO using Regulatory Position 3 of RG 1.155. SRP Section 8.4, “Station Blackout,” issued March 2007, incorporates the guidance of Regulatory Position 3, which states the following:

- Section 3.2.1: Assume a 100-percent rated thermal power for 100 days.
- Section 3.2.2: Determine core cooling and decay heat removal capability.
- Section 3.2.3: Ensure adequate inventory.
- Section 3.2.4: Evaluate design adequacy and capability, including potential failures of equipment necessary to cope.
- Section 3.2.5: Consider use of nonsafety-related equipment.
- Section 3.2.6: Consider timely operator actions.
- Section 3.2.7: Address the ability to maintain appropriate containment integrity.

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 Power In Light Water Reactors,” 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, feedwater, 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 feedwater pumps because loss of feedwater 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 or to control the reactor pressure when it begins to drop because

of the reactor scram or both.

- 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 because of a loss of ac power. No safety systems are credited, with the exception of three ICs.

- ICs are automatically initiated upon loss of feedwater 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 Level 3.

- Vessel depressurization occurs, and the 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 employing 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. The staff determined that the conditions predicted during the SBO scenario are within the limits of a LOCA and the 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 PANTHERS facility. The staff verified TRACG in this respect by conducting a comparison to the staff's confirmatory TRACE calculations. In the SER approving NEDE-33083-P-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 providing this information in a revision to the DCD. The applicant has updated DCD, Tier 2, Revision 3, with this change. The staff has reviewed and accepted the change. Based on the applicant's response, 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 so 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 applicant carried out the TRACG analysis for 20,000 seconds to demonstrate that the ICS is capable of maintaining a 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 demonstrated the adequacy of the core cooling, decay heat removal capability, and coolant inventory. The SBO analysis indicates that appropriate containment integrity is maintained throughout the duration of the event.

Because an IC is assumed to be out of service, the staff concludes that the applicant considered the potential for failure of equipment necessary to cope with an SBO. The use of nonsafety-related equipment is not assumed, and no operator actions are required.

Section 8.4 of this report provides additional information about the staff's evaluation of SBO.

15.5.5.4 Conclusion

The ESBWR reactor core and associated coolant, control, and protection systems, including station batteries and other necessary support systems, provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity in the event of an SBO for 72 hours. The applicant conducted an appropriate coping analysis to demonstrate the capability for coping with an SBO with a 72 hour duration, and hence, the acceptance criteria are satisfied.

15.5.5.5 Post-COL Activity

DCD Section 15.5.5.3 states, "Re-analysis of this event is performed for each fuel cycle."

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 control room fire. From a reactor systems standpoint, the SBO review demonstrates system response adequacy as applied to a safe-shutdown fire. Section 9.5.1 of this report provides the staff's evaluation of safe-shutdown fire from a fire protection perspective.

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 - A.2 Scope and Methodology

The NRC staff reviewed the methodology used in the determination of the event frequency. The applicant 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 which was lower than that used in the ESBWR PRA.

- The event frequency is determined from actual BWR operating experience, modified to reflect the ESBWR improved design features. For cases in which the analysis depended on specific assumed design features or testing, these features and tests are identified as ESBWR design requirements. The staff verified that the applicant had described these designed features and tests in the appropriate sections.

- 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 the applicant used this approach for the events of turbine trip with total bypass failure, generator load rejection with total bypass failure, LOFWH with failure of SCRRI/SRI, and inadvertent shutdown cooling function operation.

15A.3 Staff Evaluation Results

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, “Rates of Initiating Events at U.S Nuclear Power Plants 1987-1995, “ issued February 1999. 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 determine whether such increases produced results that exceeded the staff’s acceptance criteria.

The staff also verified that the final event frequencies are at least a factor of 3 above the criterion for the infrequent event (i.e., less than 3.33×10^{-3}/yr) to account for the modeling uncertainty.

A discussion of the staff’s evaluation of the frequency of each specific event follows.

15A.3.1 Pressure Regulator Failure—Opening 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, fault-tolerant digital controller (FTDC). The vendor will confirm the reliability of the FTDC as part of the COL applicant commitment. Upon vendor confirmation, the reliability of the SB&PC controller will meet the requirement that the mean time to failure (MTTF) be greater than 1,000 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 2,000 years (i.e., 5.0×10^{-4}/yr).

In the initial submittal, the applicant did not address adequatetly the contribution of mechanical failure to the frequency of the events. In RAI 15.0-25, the staff requested the applicant to discuss the reasons for not considering mechanical failures and clearly state any assumptions made in the analysis regarding mechanical failures. RAI 15.0-25 has been divided into multiple questions to the applicant. Each response is discussed in the applicable paragraph below.

In response to RAI 15.0-25, Item A, GEH assessed 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 5.0×10^{-4}/yr. The staff reviewed the applicant's calculations and found them to be within acceptable ranges. Therefore, based on the applicant's response, RAI 15.0-25 is resolved.

Based on the design requirement of the SB&PC described in Section 7.7.5 of this report, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.2 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 vendor will confirm the reliability of the FTDC as part of the COL applicant commitment. Upon vendor confirmation, the reliability of the SB&PC controller will meet the requirement that the MTTF be greater than 1,000 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 2,000 years (i.e., 5.0×10^{-4}/yr).

In response to RAI 15.0-25, Item A, GEH assessed 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 5.0×10^{-4}/yr. The staff reviewed the applicant's calculations and found them to be within acceptable ranges. Therefore, based on the applicant's response, RAI 15.0-25 is resolved.

Based on the design requirement of the SB&PC described in Section 7.7.5 of this report, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.3 Turbine Trip with Total Bypass Failure

In RAI 15.0-20, the staff requested the applicant to provide additional information to justify and/or clarify assumptions and statements made in DCD, Tier 2, Revision 1, Sections 15A.3.3 (Turbine trip with total bypass failure) and 15 A.3.4 (Generator load rejection with total bypass failure). In response to RAI 15.0-20, Items (A) through (F), GEH modified the model of the turbine bypass failure using the linked fault-tree approach. The modeling of the bypass valves failures includes the electric-hydraulic control (EHC) system, related mechanical components, and supporting power supplies. The staff reviewed the applicant's calculations and found them to be within acceptable ranges. Therefore, based on the applicant's response, RAI 15.0-20, Items (A) through (F), are 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 $5.17x10^{-4}$/yr. The staff agrees that this event frequency meets the criterion of less than $1.0x10^{-2}$/yr.

15A.3.4 Generator Load Rejection with Total Turbine Bypass Failure

In response to RAI 15.0-20, Items (A) through (F), GEH modified the model of the turbine bypass failure by using the linked fault-tree approach. The modeling of the bypass valve failures included the EHC system, related mechanical components, and supporting power supplies. The staff reviewed the applicant's calculations and found them to be within acceptable ranges. The staff agrees with the applicant's 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. The staff reviewed the applicant's calculations and found them to be within acceptable ranges. Therefore, based on the applicant's response, RAI 15.0-20 Item (G) 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 $1.98x10^{-4}$/yr. The staff agrees that this event frequency meets the criterion of less than $1.0x10^{-2}$/yr.

15A.3.5 Feedwater Controller Failure

The FWCS accomplishes both RPV water level control and FWT control. The two functions are performed by two sets of triple-redundant controllers located in separate cabinets, using independent and diverse inputs. Two events of concern may result from failures of the FWCS. One event consists of the FWCS erroneously generating a maximum flow demand, and the other event consists of the FWCS erroneously generating a minimum temperature demand. The simultaneous occurrence of a maximum flow demand and a minimum temperature demand is considered incredible because of the independence of the two control schemes. The random probability of the second controller (e.g., temperature) failing while the first controller (e.g., flow) is failed, and before the effects of its failure are mitigated, is insignificant.

15A.3.5.1 Feedwater Controller Failure—Maximum Flow Demand

One function of the FWCS is to regulate the flow of feedwater into the RPV to maintain predetermined water level limits during transients and normal plant operating modes. The FWCS is equipped with a dedicated, triple-redundant FTDC, including power supplies, and input and output signals. The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms. The FTDC is designed to a high degree of reliability, and the MTTF of the FTDC is at least 1,000 years. It is assumed that the feedwater flow controller can fail high or fail low with equal probability. Therefore, the frequency of the controller failing in a manner to cause maximum demand is less than once in 2,000 years.

Based on the design requirement of the FWCS described in Section 7.7.3 of this report, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.5.2 Feedwater Controller Failure—Minimum Temperature Demand

One function of the FWCS controls FWT to allow reactor power control without moving control rods. The FWCS is equipped with a dedicated, triple-redundant FTDC, including power supplies, and input and output signals. The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms. The FTDC is designed to a high degree of reliability, and the MTTF of the FTDC is at least 1,000 years. It is assumed that the FWT controller can fail high or fail low with equal probability. Therefore, the frequency of the controller failing in a manner to cause minimum demand is less than once in 2,000 years.

Based on the design requirement of the FWCS described in Section 7.7.3 of this report, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.6 Loss of Feedwater Heating with Failure of Selected Control Rod Run-In and Selected Rod Insertion

In RAI 15.0-21, the staff requested the applicant to justify assumptions in the frequency estimate for "Loss of Feedwater Heating with Failure of SCRRI". Based on the responses to RAIs 15.0-21 and 15.0-25, Item (B.1), GEH modified the initiating event frequency estimate by adding the SRI 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 1.51×10^{-3}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr. Therefore, based on the applicant's response, RAIs 15.0-21 and 15.0-25 are resolved.

15A.3.7 Inadvertent Shutdown Cooling Function Operation

In RAI 15.0-22, the staff requested the applicant to justify the assumed interlock frequency and operator error probability for "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 SDC actuation. The analysis modeled valve functions, testing, and operator errors. Based on this approach, GEH estimated 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 less than 1.0×10^{-2}/yr. Therefore, based on the applicant's response, RAI 15.0-22 is resolved.

15A.3.8 Inadvertent Opening of a Safety/Relief Valve

In RAI 15.0-23, the staff requested the applicant to justify the assumptions in event frequency estimate for "Inadvertent Opening of a Safety/Relief Valve".

In response to RAI 15.0-23, GEH included detailed failure modes of inadvertent opening of an SRV leading to vessel depressurization. Modeled failure modes include 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 2.81x10^{-3}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0x10^{-2}/yr. Therefore, based on the applicant's response, RAI 15.0-23 is resolved.

15A.3.9 Inadvertent Opening of a Depressurization Valve

In RAI 15.0-24, the staff requested the applicant to address PRA modeling of the I&C system including common cause failures.

In response to RAI 15.0-24, GEH 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 5.75x10^{-4}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0x10^{-2}/yr. Therefore, based on the applicant's response, RAI 15.0-24 is resolved.

15A.3.10 Stuck-Open Relief Valve

In RAI 15.0-28, the staff requested the applicant to provide technical basis of unavailability of the IC.

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 ICS for the ESBWR. The applicant assumed that the probability of the IC being unavailable is less than 0.1. However, the applicant provided no justification for this number in this section. The staff 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 ICS and explained that the assumed value of 0.1 is conservative because of the simplicity, redundancy, and diversity of the ICS system. The staff agrees with this response. Therefore, based on the applicant's response, RAI 15.0-28 is resolved.

The staff issued RAI 15.0-29 to clarify the inconsistency of the data used for the stuck open relief valve initiating event in ESBWR

In RAI 15.0-29, Section 15A.3.10, GEH provided a best estimate value for the expected frequency of a stuck-open SRV in an ESBWR of 3.28x10^{-4}/yr. However, the traditional number used for existing BWR plants is about 4.6x10^{-2}/yr (see NUREG/CR-5750). In addition, the number used in the ESBWR PRA is 2.23x10^{-2}/yr (see NEDO-33201, Revision 2, Section 2, "ESBWR Design Certification Probabilistic Risk Assessment"). The staff asked the applicant to explain why the ESBWR PRA did not use the best estimate of ESBWR frequency (i.e., 3.28x10^{4}/yr).

The applicant addressed RAI 15.0-29 by providing the initiating event frequency of a stuck-open SRV and explaining that the value used in the ESBWR is lower than that of the existing BWRs by crediting the ICS and eliminating the surveillance testing requirement of SRVs during power operation for the ESBWR. The staff agrees with this response. Therefore, based on the applicant's response, RAI 15.0-29 is resolved.

Based on the RAI responses and improved modeling, the estimated event frequency of this event is 2.24×10^{-4}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.11 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 an RWE during refueling is significantly less than once in 1,000 years, based on the multiple failures required for this event to occur.

In response to RAI 15.0-25, Item B.2, GEH assessed the mechanical failure of the FMCRD and concluded that the likelihood of mechanical failure of the FMCRD is negligible compared to the estimated overall failure frequency. The staff reviewed the applicant's calculations and found them to be within acceptable ranges.

Based on the RAI responses and estimated failure frequency of 1×10^{-3}/yr, the staff agrees that this event frequency meets the criterion of less than 1×10^{-2}/yr. Therefore, based on the applicant's response, RAI 15.0-25, Item B.2 is resolved.

15A.3.12 Control Rod Withdrawal Error during Startup with Failure of Control Rod Block

The applicant 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 1.20×10^{-6}/yr, and the frequency of manual rod withdrawal is about 1.5×10^{-7}/yr. With the consideration of uncertainty, these values are less than 1.0×10^{-2}/yr.

In response to RAI 15.0-25, Item B.2, GEH assessed 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 reviewed the applicant's calculations and found them to be within acceptable ranges and agrees with the RAI responses and the failure assessment of this event. Based on the estimated failure frequency of control RWE during startup, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr. Therefore, based on the applicant's response, RAI 15.0-25 is resolved.

15A.3.13 Control Rod Withdrawal Error during Power Operation

The causes of a potential RWE 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 estimated that the frequency of an automatic control rod withdrawal is about 1.20×10^{-9}/yr, and the frequency of manual rod withdrawal is about 2.5×10^{-5}/yr. With the consideration of uncertainty, these values are less than 1.0×10^{-2}/yr.

In response to RAI 15.0-25, Item B.2, GEH assessed 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 reviewed the applicant's calculations and found them to be within acceptable ranges and agrees with the applicant's RAI responses and the failure assessment of this event. Based on the estimated failure frequency of control RWE during power operation, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr. Based on the applicant's response, Therefore, based on the applicant's response, RAI 15.0-25 is resolved.

15A.3.14 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 in 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 estimated that the mislocated bundle frequency is 9.6×10^{-4}/yr.

The staff reviewed the applicant's calculations and found them to be within acceptable ranges. 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 less than 1.0×10^{-2}/yr.

15A.3.15 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 estimated that the misoriented bundle frequency is 2.4×10^{-3}/yr.

The staff reviewed the applicant's calculations and found them to be within acceptable ranges. 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 not being detected and the estimate of the frequency of a misoriented bundle, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.16 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 less than 1.0×10^{-2}/yr.

15A.3 Conclusion

Based on the above discussions, the staff agrees that the events reviewed have frequencies less than 0.01/yr, with the consideration of the uncertainty.

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