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Subject: **Partial Response to NRC Request for Additional Information Letter No. 25 Related to ESBWR Design Certification Application – Accident Analyses – RAI Numbers 15.0-3 through 15.0-15; 15.2-1, 15.2-2, and 15.2-4**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter.

If you have any questions about the information provided here, please let me know.

Sincerely,

Kathy Sedney for

David H. Hinds
Manager, ESBWR

D068

Reference:

1. MFN 06-142, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 25 Related to ESBWR Design Certification Application,* May 9, 2006

Enclosure:

1. MFN 06-173 – Partial Response to NRC Request for Additional Information Letter No. 25 Related to ESBWR Design Certification Application – Accident Analyses – RAI Numbers 15.0-3 through 15.0-15; 15.2-1, 15.2-2, and 15.2-4

cc:	WD Beckner	USNRC (w/o enclosures)
	AE Cubbage	USNRC (with enclosures)
	LA Dudes	USNRC (w/o enclosures)
	GB Stramback	GE/San Jose (with enclosures)
	eDRF	0000-0054-1366

ENCLOSURE 1

MFN 06-173

Partial Response to NRC Request for Additional Information Letter No. 25 Related to ESBWR Design Certification Application – Accident Analyses – RAI Numbers 15.0-3 through 15.0-15 15.2-1, 15.2-2, and 15.2-4

NRC RAI 15.0-3

Provide tables in DCD Tier 2, Sections 15.3, 15.4, and 15.5, similar to DCD Tier 2, Table 15.2-1, listing the input parameters and initial conditions assumed in the respective analyses.

GE Response

DCD Table 15.2-1 is also applicable to the infrequent event analysis in Section 15.3. The title of Table 15.2-1 will be changed to read "Input Parameters And Initial Conditions Used In AOO and Infrequent Event Analyses." A note that reads "The input parameters and initial conditions used to perform the analysis in this table is located in Table 15.2-1" will be added to Table 15.3.1. These changes will be made in the next update of the DCD.

Section 15.4 primarily contains accident dose release analysis. Each accident, for which analysis is performed, contains its own table of input parameters. No DCD Tier 2 changes are proposed to Section 15.4.

Section 15.5 presents the special event evaluations. These events have specific and varied requirements. Each event evaluation includes input parameters and assumptions in the discussion of the event or in a table specific to the event. To include, in one table, the initial conditions for all special events may add confusion due to the varied regulations that govern each event. No DCD Tier 2 changes are proposed to Section 15.5.

NRC RAI 15.0-4

SRP Chapter 15.0, Draft Rev.3, 1996, states that: "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. The Applicant's Safety Analysis Report should discuss each transient and accident analysis to justify that it conforms to GDC 17 requirements. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Evaluation Report for the ABB-CE System 80+ design certification. This position was also applied in the AP1000 design certification.)
Describe in detail how the ESBWR transient and accident analyses were performed to comply with GDC 17.

GE Response

For the ESBWR, no safety-related function requires either offsite AC power or onsite emergency diesel generator AC power for 72 hours, and the onsite battery electric power supplies and the onsite electric distribution system have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure. After 72 hours, credit is taken for nonsafety-related DC and AC power, which can be supplied from two independent nonsafety-related diesel generator divisions and/or two offsite AC power supplies. The Chapter 15 safety analyses are not specifically performed to show compliance with GDC 17, but are performed to

show compliance with GDC 10, GDC 15, GDC 19, 10 CFR 20, 10 CFR 50.34(a) and other applicable acceptance criteria in the 10 CFRs, Technical Specifications' safety limits, and associated design code allowables. Compliance with GDC 17 is addressed within Chapter 8.

No ESBWR accident analysis assumes the availability of offsite power. The ESBWR AOO events do include the loss of offsite power event (i.e., *Loss of Grid Connection*).

The SRP Chapter 15.0, Draft Rev.3, 1996 statement is too broad to be completely applicable to the ESBWR design, and is not consistent with the definition of the AOO in 10 CFR 50, Appendix A. The ESBWR AOOs are not categorized based on past designs and practices, but are categorized based on the actual definition of an AOO in 10 CFR 50, Appendix A, as discussed within DCD Section 15.0 and Appendix 15A. AOO definition in 10 CFR 50, Appendix A is based on event frequency. If any initiating event along with the loss of offsite power is not "expected to occur one or more times during the life of the nuclear power unit," then that event along with the loss of offsite power is, by regulation, not an AOO. Any draft (i.e., not in affect) non-regulation interpretation that contradicts the as-written text of a regulation is not appropriate.
There are many and significant differences between a PWR and an ESBWR, which make it inappropriate to apply this System 80+ specific feature. For example, one ESBWR event, *Runout of One Feedwater Pump*, is not even physically possible if there is a loss of offsite power, and thus, it is impossible to apply the PWR based interpretation of the SRP Chapter 15.0, Draft Rev.3, 1996 statement to that ESBWR event.

No Tier 2 change will be made in response to this RAI.

NRC RAI 15.0-5

On DCD Tier 2, page 15.0-1, second paragraph, it is stated that "[s]ystem response analyses are based upon the core loading shown in Figure 4.3-1, and is used to identify the limiting events for the ESBWR. Other fuel designs and core loading patterns, including loading patterns similar to Figure 4.3-2, do not affect the sensitivities demonstrated by this study."
We do not agree with the statement that "other fuel designs and core loading patterns similar to Figure 4.3-2, do not affect the sensitivities demonstrated by this study." A comprehensive explanation is required to support the conclusion that other fuel designs do not affect the sensitivities.

GE Response

GE does not entirely agree with the stated position. On a specific event basis, there would be small response differences from one core loading to the next, and thus, the statement on page 15.0-1 will be qualified. However, the limiting subset of events for any specific abnormal event category (e.g., AOOs) versus all the events in that category is never expected to change. Plus, the categorization process of all the abnormal events, addressed in Chapter 15, is not fuel/core design dependent.

All limiting events (e.g., AOOs) that could change from one core loading to the next will be analyzed on a fuel cycle specific basis, with resulting cycle-specific fuel thermal limits documented in a Core Operating Limits Report.

The BWR and its responses to abnormal events are far simpler than those of a PWR. The original AOOs and accidents determined in the mid-1960's are still analyzed or evaluated for all operating BWRs today. The ESBWR's core reactivity responses to changes are basically like any other BWR's. Plus, the ESBWR is even simpler than the operating BWRs, and thus, has fewer ways to be perturbed. For example, without forced recirculation, or high pressure ECCS, the ESBWR has fewer ways to increase core reactivity than the current BWRs.

Plus, the ESBWR has far more free volume inside the reactor vessel than the operating BWRs, and thus, ESBWR responses to pressurization events are less severe than those of the operating BWRs. Forty years of operating BWR experience and the Section 15.2 array analyses performed for the ESBWR validate the page 15.0-1 statement.

GE proposes to clarify the second and third sentences to read "Other fuel designs and core loading patterns, developed in compliance with Reference 15.0-2 (GESTAR II), similar to the one shown in Figure 4.3-1, do not significantly affect the sensitivities demonstrated by this study."

Tier 2 Section 15.0 will be revised in the next update as noted above and in the attached markup.

NRC RAI 15.0-6

On DCD Tier 2, page 15.0-1, sixth paragraph, it is stated that "[t]he starting point for the NSOA is the establishment of unacceptable safety results." The goal is to meet the acceptable safety limits. This paragraph should be revised to clarify the establishment of the acceptable safety goals.

GE Response

"Safety limits" only exist in the Technical Specifications and are a subset of the total set of regulatory acceptance criteria that the safety analyses must meet. GE proposes to clarify the sentence to read as follows:

"The starting point for the NSOA is the regulatory acceptance criteria and design code allowables such that the acceptability of safety analysis results can be determined."

Tier 2 Section 15.0 will be revised in the next update as noted above and in the attached markup.

NRC RAI 15.0-7

DCD Tier 2, Section 15.0.1, "Classification and Selection of Events":
10 CFR 50, the SRP and Regulatory Guides were developed at different times and for different purposes. They may not be consistent. Delete the statements "inconsistent" and "are not clearly defined" etc. from this section of the DCD.

GE Response

The statements "inconsistent" and "are not clearly defined" are technically correct, and well documented in NEDO-33175. However, GE proposes to clarify Subsection 15.0.1 by replacing the 2nd through 6th bullets with the following:

- The SRP and Regulatory Guide (RG) 1.70 do not provide specific definitions for all versions of abnormal event categories;
- The SRP and RG 1.70 do not use the same terminology for the non-accident abnormal events, and thus, the non-accident abnormal event classifications within the SRP and RG 1.70 could be misinterpreted;
- The non-accident abnormal event classification terms in the SRP and RG 1.70 are not the same as the abnormal event classifications in the 10 CFR regulations;

Tier 2 Section 15.0 will be revised in the next update as noted above and in the attached markup.

NRC RAI 15.0-8

ANS 52.1 is inactive and is withdrawn. Please refer the current code which is active.

GE Response

GE proposes to revise the 4th paragraph of Subsection 15.0.1 to read:

"Based on Subsection 5.5.2, Item (3) of ANSI/ANS-58.14-1993, DBEs should have annual probabilities ≥ 10-6. Therefore, any event with an annual probability of < 10-6 is not considered credible, and thus, is not classified as a DBE."

GE proposes to change the reference to ANS 58.14 in Subsection 15.0.5.

Tier 2 Section 15.0 will be revised in the next update as noted above and in the attached markup.

NRC RAI 15.0-9

DCD Tier 2, Section 15.0.1.1, "Approach for Determining Event Classifications":
In the fourth paragraph, delete reference to "ABWR DCD portions associated accidents."

GE Response

GE agrees to delete that statement from Subsection 15.0.1.1, Item (4).
Tier 2 Section 15.0 will be revised in the next update as noted above and in the attached markup.

NRC RAI 15.0-10

DCD Tier 2, Section 15.0.1.2, "Results of Event Classification Determination":
Paragraph (1) a: Since item # 3 describes the infrequent event, replace "accidents" with "infrequent events."
Paragraph (1) c: Delete reference to the ABWR.
Paragraph (3): Delete reference to the ABWR.

GE Response

GE agrees to make the requested changes.

Tier 2 Section 15.0 will be revised in the next update as noted above and in the attached markup.

NRC RAI 15.0-11

DCD Tier 2, Section 15.0.2, "Abnormal Events to be Evaluated":
Add "Reactivity and power distribution anomalies" to the nuclear system parameter variations listed.
The potentially limiting anticipated operational occurrences (AOOs) listed based on GESTAR II may not be limiting for ESBWR. Confirm that the ESBWR analyses performed support this conclusion.

GE Response

As stated in Subsection 15.2.3, "There are no reactivity and power distribution anomaly AOOs identified for the ESBWR." These types of events are part of the set of infrequent events, and GE agrees to address reactivity and power distribution anomalies within the infrequent event paragraph in Subsection 15.0.2.

The limiting ESBWR-specific AOOs have been determined, and will be evaluated for the initial core and for subsequent reload cores, as specified Subsection 15.2.7.

Tier 2 Section 15.0 will be revised in the next update as noted above and in the attached markup.

NRC RAI 15.0-12

DCD Tier 2, Sections 15.0.3, "Determination of Safety Analysis Acceptance Criteria," and 15.0.3.4, "Special Events":
Delete reference to the ABWR FSER as the basis for acceptance criteria.

GE Response

GE proposes to change the last sentence of the 1st paragraph of Subsection 15.0.3 to read as follows:

"Where an acceptance criterion is not specified in regulations and the SRP, then the criterion in the Reference 1 LTR shall be used."

GE proposes to delete the 6th paragraph of Subsection 15.0.3.1.

Subsection 15.0.3.4 does not specifically address the ABWR FSER, however, a paragraph in Subsection 15.0.3.4.7 does addresses the ABWR FSER. GE proposes to delete this paragraph.

Tier 2 Section 15.0 will be revised in the next update as noted above and in the attached markup.

NRC RAI 15.0-13

DCD Tier 2, Section 15.0.3.1, "Anticipated Operational Occurrences":
Add a paragraph to the DCD addressing GDC 10, similar to the paragraph addressing GDC 15. GDC 10 is also applicable to AOOs.

GE Response

GE proposes to increase the content of Subsection 15.0.3.1, 1st paragraph to quote the GDC 10 acceptance criterion, similar to how the 4th paragraph addresses GDC 15. The 1st paragraph would read as follows.

"For AOOs, the GDC 10 acceptance criterion is that "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences." To meet the intent of GDC 10, SRP 15.1 and SRP 15.2, detailed acceptance criteria for AOOs both not in combination and in combination with an additional single active component failure (SACF) or single operator error (SOE) are provided. For an AOO, which is not in combination with an additional SACF or SOE, the SRP 15.1 and SRP 15.2 criterion is "Fuel cladding integrity shall be maintained by ensuring that the minimum CPR remains above the MCPR safety limit for BWRs based on acceptable correlations." This is equivalent to the fuel cladding integrity (greater than 99.9% of the fuel rods in the core would be expected to avoid boiling transition) safety limit (SL) included in the ESBWR Technical Specifications (TS)."

Tier 2 Section 15.0 will be revised in the next update as noted above and in the attached markup.

NRC RAI 15.0-14

DCD Tier 2, Section 15.0.3.4.2, "Shutdown Without Control Rods":
Is there a difference between this event and anticipated transient without scram (ATWS)? Why is this event considered as a "special event"? Please clarify.

GE Response

The original design basis (from the 1960s) for the Standby Liquid Control (SLC) system is the Shutdown Without Control Rods event. (The ATWS rule was not published until 1984.) In this event, a controlled shutdown is attempted and it is assumed that multiple equipment failures and/or common mode failures prevent the control rods from being inserted. Upon failure standby liquid control would be initiated manually (see DCD Subsection 9.3.5). This event has a specific set of acceptance criteria, different than ATWS. There is no transient (i.e., AOO) associated with this event, and thus, it is not an ATWS.

No Tier 2 change will be made in response to this RAI.

NRC RAI 15.0-15

Within Chapter 4, the Rod Withdrawal Event (RWE) is classified as an AOO. Based upon the RWE description within Chapter 15 and recent conversations, it appears that GNF is attempting to reclassify the RWE into a lower frequency category.
(a) Please describe the basis for the reclassification of the RWE including initiating actions/events and mitigating strategies from all modes of operation.
(b) Please describe the potential for a "gang" withdrawal error (e.g. multiple control rods).
(c) Please identify the proposed acceptance criteria for the new event classification.

GE Response

As shown in Tables 15.0-2, 15.0-7 and 15A-3 a RWE is an infrequent event and not an AOO. GE will revise Appendix 4B to be consistent with Chapter 15.

The DCD establishes the initial design and licensing for the ESBWR. Many ESBWR design features are significantly different than in the current operating BWRs, and thus, basing the event classifications on past BWR designs would result in some ESBWR events being classified inconsistently with respect to the 10 CFR regulations. To ensure and maximize consistency with the 10 CFR regulations, the ESBWR event classifications are objectively based on the specific wordings in 10 CFR regulations, regardless other plants. 10 CFR 50 Appendix A provides an explicit definition of an AOO. 10 CFR 50 Appendix A states "Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power." The ESBWR design life is 60 years, and thus, any abnormal event with a probability ≥ 1/60 per year must be classified as an AOO, and conversely, any abnormal event with a probability < 1/60 per year should not be classified as an AOO. However, Subsection 15.0.1.2 conservatively defines an AOO "any abnormal event that has an event probability of ≥ 1/100 per year."

From Table 15A-3, the most likely RWE has a probability of 1/1000 per year (1 Event in 1,000 yrs). Therefore, the RWE is correctly classified as an infrequent event in Chapter 15. The analyses that determine the RWE probability are described in Subsections 15A.3.11, 12 and 13. Because a RWE is an infrequent event, (1) Table 15.0-5 provides its acceptance criteria, and (2) "(core-wide AOOs only)" after "No fuel centerline melt," "Energy generation is < 170 cal/gm (RWE during startup only)," and note "*" are being deleted from Table 15.0-3.

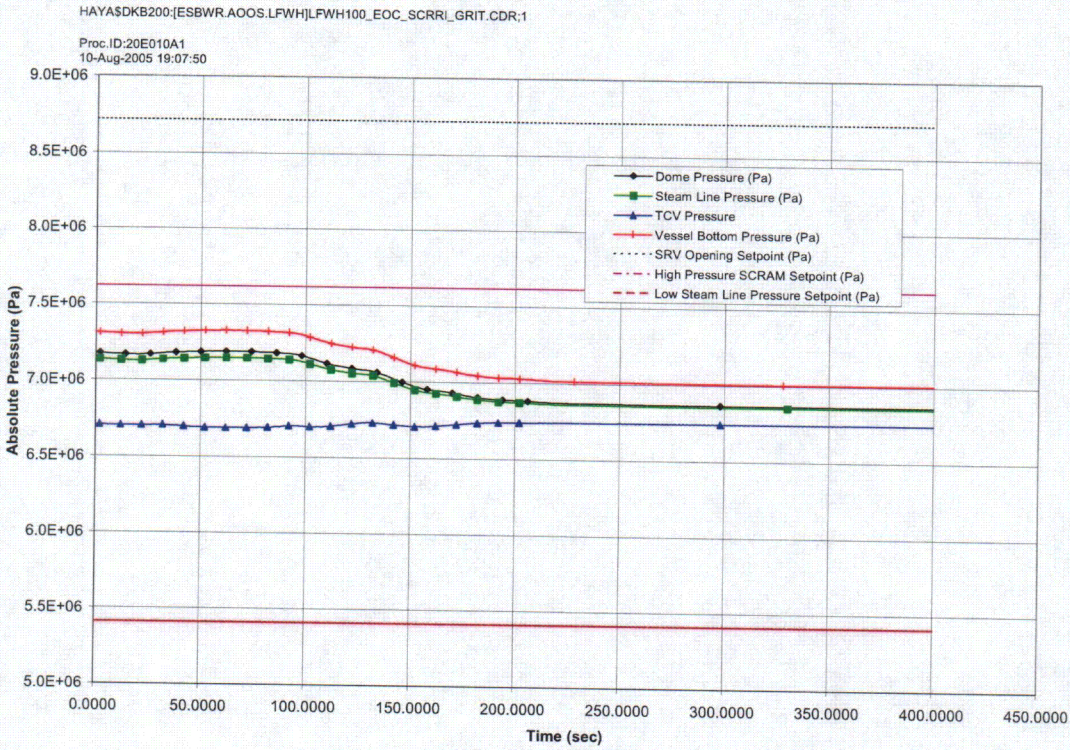
Tier 2 Section 15.0 will be revised in the next update as noted above and in the attached markup.

NRC RAI 15.2-1

DCD Tier 2, Section 15.2.1.1, Fig. 15.2-1d indicates turbine pressure to fail at about 150 seconds, yet the scenario does not indicate turbine trip or associated reactor scram. Please explain.

GE Response

The turbine pressure shown in the plot (15.2-1d) is the pressure downstream of the turbine control valves (TCVs). This is not the pressure that will be monitored to give a low steam line pressure isolation. The low steamline pressure isolation will come from upstream of the TCVs. The pressure upstream of the TCVs will not drop below "Low Steam Line Pressure Setpoint" shown in the figure. Changes to figures (Figures 15.2-xxd and 15.3-xxd) containing "Turbine Pressure" will be made in the next update of the DCD to show the pressure upstream of the TCVs. The figure below shows a Figure 15.2-1d with the plot of "TCV Pressure" taken from upstream of the TCVs.



NRC RAI 15.2-2

DCD Tier 2, Section 15.2.2.6, states that closure of one MSIV will lead to reactor operation at a new steady state. Figure 15.2-8a indicates that the new steady state to be at 100% power. Is this a physically feasible transient or a numerical exercise?

GE Response

It is a physically feasible transient. The closure of one MSIV causes dome pressure to increase. The increase in dome pressure causes the pressure control system to open the Turbine Control Valves (TCVs). This action helps control pressure but is not sufficient therefore the bypass opens. This action controls the dome pressure to a slightly higher pressure than the initial. The small increase in the dome pressure causes a small increase in the new steady state reactor power (~1%). The increase in steady state steam flow (~4%) is due to the increased power and increased liquid content in the steam flow. The result is a new steady state with power about 101% and steam flow about 104%, core flow remains close to the initial value.

The increase in the steam line mass flow rate in the remaining 3 steam lines reaches a value of nearly 140% of rated flow at about 10 seconds. If the flow reached 140% (analytical limit), this could cause an MSIV isolation signal which would close the remaining 3 main steam lines. Considering the uncertainties in the instrumentation it's quite possible the MSIV isolation trip would occur. This event scenario will be studied in more detail and revisions will be made to the DCD as appropriate.

15. SAFETY ANALYSES

This chapter addresses all ESBWR plant safety analyses. The details of most of the safety analyses are contained within this chapter, however, per the Regulatory Guide 1.70 format, some safety analyses are addressed in detail in other DCD Tier 2 chapters (e.g. emergency core cooling system [ECCS] performance is addressed within Section 6.3). For those safety analyses not addressed in detail in Chapter 15, references are provided to their locations within Tier 2.

15.0 ANALYTICAL APPROACH

In this chapter, the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate plant capabilities to control or accommodate such failures and events. System response analyses are based upon the core loading shown in Figure 4.3-1, and is used to identify the limiting events for the ESBWR. Other fuel designs and core loading patterns, developed in compliance with Reference 15.0-2 (GESTAR II), similar to the one shown in Figure 4.3-1, do not significantly affect the sensitivities demonstrated by this study.~~Other fuel designs and core loading patterns, including loading patterns similar to Figure 4.3-2, do not affect the sensitivities demonstrated by this study.~~ Evaluation of these limiting events for each plant fuel cycle ensures that the criteria in Appendix 4B are met.

GE has developed a unique systematic approach to plant safety consistent with the GE boiling water reactor (BWR) technology base. The key to the GE approach to plant safety is the Section 15.1 Nuclear Safety Operational Analysis (NSOA). A NSOA is a system level qualitative failure modes and effects analysis (FMEA) of plant protective functions to show compliance with the events addressed in Chapter 15. Key inputs into the NSOA are derived from the applicable regulations, through industry codes and standards, and the plant safety analyses.

In Section 15.1, all unacceptable safety results and all required safety actions are identified. In addition, an evaluation of the entire spectrum of events is consistently carried out for the ESBWR to demonstrate that a consistent level of safety has been attained.

The NSOA acceptance criteria are based on the Title 10 of the Code of Federal Regulations (10 CFR regulations) and the NUREG-0800 Standard Review Plan (SRP) acceptance criteria.

The starting point for the NSOA is the regulatory acceptance criteria and design code allowables such that the acceptability of safety analysis results can be determined.~~The starting point for the NSOA is the establishment of unacceptable safety results.~~ This concept enables the results of any safety analysis to be compared to applicable criteria. Unacceptable safety results represent an extension of the nuclear design criteria for plant systems and components that are used as the basis for system design. The unacceptable safety results have been selected so that they are consistent with applicable regulations and industry codes and standards.

The focal point of the NSOA is the event analysis, in which all essential protection sequences are evaluated until all required safety actions are successfully completed. The event analysis identifies all required front-line safety systems and their essential auxiliaries.

The full spectrum of initial conditions limited by the constraints placed on planned operation is evaluated. All events are analyzed until a stable condition is obtained. This ensures that the event being evaluated does not have an unevaluated long-term consideration.

In the event analysis, all essential systems, operator actions and limits to satisfy the required safety actions are identified. Limits are derived only for those parameters continuously available to the operator. Credit for operator action is taken only when an operator can be reasonably expected to perform the required action based on the information available to him.

In the NSOA, a complete and consistent set of safety actions (i.e., those required to prevent unacceptable results) has been developed. For all of the events that are evaluated, a single-failure-proof path to plant shutdown is identified. The application of the 10 CFR 50, Appendix A single-failure criterion (SFC) to these events is imposed as an additional measure of conservatism in the NSOA process.

15.0.1 Classification and Selection of Events

From Reference 15.0-1, the classification of events for the ESBWR is primarily based on the classifications and terms used in the 10 CFR regulations because:

- The 10 CFR regulations have authority over all other document types;
- The SRP and Regulatory Guide (RG) 1.70 do not provide specific definitions for all versions of abnormal event categories;
- The SRP and RG 1.70 do not use the same terminology for the non-accident abnormal events, and thus, the non-accident abnormal event classifications within the SRP and RG 1.70 ~~are inconsistently used~~could be misinterpreted;
- ~~The non-accident abnormal event classifications within Regulatory Guide (RG) 1.70 are inconsistently used;~~
- ~~The classifications of non-accident abnormal event classifications between the SRP and RG 1.70 are inconsistent;~~
- ~~Both sets of~~The non-accident abnormal event classification terms in the SRP and RG 1.70 are not ~~consistent~~ the same as ~~with~~ the abnormal event classifications in the 10 CFR regulations;
- ~~All versions of abnormal event categories are not clearly defined in the SRP and RG 1.70;~~
- The 10 CFR regulations do specifically define an Anticipated Operational Occurrence (AOO), Loss-of-Coolant Accident (LOCA), Anticipated Transient Without Scram (ATWS), normal operation, design basis events, and a number of associated terms; and
- The use of terms is more consistent within the 10 CFR regulations than in the SRP or RG 1.70.

The most recently certified BWR (i.e., the ABWR) licensing documents are used for additional guidance.

The design basis events (DBEs) in the 10 CFR regulations assume an initiating event (and any resultant failures) with or without a single active component failure or operator error. The postulating of design basis events that assume a failure beyond the SFC or a common-mode failure is not specifically required by the 10 CFR regulations. However, the 10 CFR regulations do require evaluations of three specific event scenarios, i.e., Safe Shutdown Fire, Station

Blackout (SBO) and ATWS, and some of these event scenarios do assume failures beyond the single failure criterion (SFC) and/or common-mode failures. Therefore, these events should not be classified as DBEs. However, their safety analyses are included in the ESBWR DCD.

Based on Subsection 5.5.2, Item (3) of ANSI/ANS-58.14-1993, DBEs should have annual probabilities $\geq 10^{-6}$. Therefore, any event with an annual probability of $< 10^{-6}$ is not considered credible, and thus, is not classified as a DBE~~Based on Table 3-1 of ANSI/ANS-52.1, DBEs should have annual probabilities $\geq 10^{-6}$. Therefore, any event with an annual probability of $< 10^{-6}$ is not considered credible and is not classified as a DBE~~.

The 10 CFR regulations, SRP and RG 1.70 postulate events that (for the ESBWR with its advanced design features and additional redundancy) require failures beyond the SFC and/or require common-mode failures. Those events are included in the ESBWR DCD, but not as DBEs.

Per the 10 CFR regulations, AOOs are expected to occur once in a plant's lifetime, while accidents are low probability events that are not expected to occur during a plant's lifetime. Because the ESBWR has a design life of 60 years, any abnormal event that has an annual probability of occurrence $\geq 1/60$ could be classified as an AOO. However, historically, a value of $> 1/100$ has been used.

Based on the 10 CFR regulations, the SRP or an NRC reviewed Licensing Topical Report (LTR), the safety analysis acceptance criteria for each of the special events is developed on an event-specific basis.

The 10 CFR regulations consistently refer to any failure of a fission product barrier that results in an offsite radiological consequence as an accident.

15.0.1.1 Approach For Determining Event Classifications

(1) Per the 10 CFR regulations, the 10 CFR 50 App. A definitions, GDC, the 10 CFR 50.49 design basis event definition, SRP 6.1.1, SRP 15.0.1, RG 1.183 and guidance from events addressed in the SRP;

   a. Divide the types of events as DBEs, and by exclusion, all other events as special events,

   b. Provide the basis for which events should be classified as AOOs,

   c. Provide the basis for a (non-AOO and non-accident) event classification for events with lower probabilities than AOOs, but for conservatism have historically not been treated or classified as accidents, and

   d. Generate the criterion for determining which type of accidents shall be classified as design basis accidents (DBAs), and by exclusion, all other accidents are not DBAs.

(2) Per the regulatory definition of an AOO (event probability), historical information and guidance from the SRP determine specific criteria for classifying events as AOOs.

(3) Based on (a) the 10 CFR regulations associating accidents with radiological consequences, (b) application of SFC, (c) SRP and RG 1.70 guidance for the types of events that should be addressed in Chapter 15, (d) SRP acceptance criteria for transient/AOO events that result in fuel failure, and (e) historical consistently used terms, generate a classification term and criteria for determining non-AOO and non-accidents events, which (a) should be treated as

design basis events and (b) result from an initiating event with or without assuming a single active component failure or single operator error. Include this new DBE term in the DBE classifications.

(4) Based on the 10 CFR regulations, SRPs, RG 1.183~~ and the ABWR DCD portions associated accidents,~~ generate a definition for an accident.

(5) Based on (a) reviewing the 10 CFR regulations that have added other abnormal events (e.g., ATWS, SBO, Safe Shutdown Fire), (b) that DBEs do not include common-mode failures and/or additional failure(s) beyond the SFC, (c) reviewing the SRP events that include common-mode failures and/or failure(s) beyond the SFC, and (d) historically evaluated non-DBE events and used associated classification terms, generate classification term for non-DBEs that are addressed in the DCD Chapter 15.

15.0.1.2 Results of Event Classification Determinations

Table 15.0-1 provides the results of the event classifications in the form of a determination criterion vs. event classification matrix. Table 15.0-1 is based on the results from the following evaluation.

(1) a. Per 10 CFR 50.49, and the fact that the SRP treats all postulated abnormal initiating events with or without assuming a single active component failure or single operator error as if they are all design basis events, the following are classified as design basis events:

- Normal operation, including AOOs;
- ~~Accidents~~ Infrequent events [see Item (3) for additional details];
- ~~Design basis a~~Accidents;
- External events; and
- Natural phenomena,

(1) b. AOOs, by definition, are classified as part of normal operations, do not have radiological consequences (except if in combination with an additional single active component failure or single operator error), have more restrictive acceptance criteria (e.g., GDC 10 or 10 CFR 20 vs. 10 CFR 50.34) than accidents, and thus, are not accidents and shall not be treated as accidents.

(1) c. ~~The~~ A classification term for events with lower probabilities than AOOs, but for conservatism should be not treated as accidents should be ~~based on the ABWR~~provided.

(1) d. Except for AOOs, the 10 CFR regulations, SRP and RG 1.70 do not explicitly or implicitly apply any quantitative event frequency criterion for defining any other abnormal event classification. Therefore, event frequencies should not be used to determine accident type event classifications.

SRP 6.1.1, SRP 15.0.1 and RG 1.183 are consistent in categorization of DBAs. A DBA *is an accident postulated and analyzed to confirm the adequacy of a plant engineered safety feature.*

By exclusion, all other accidents are not classified as DBAs.

(2) An AOO *is any abnormal event that has an event probability of ≥ 1/100 per year.*

(3) ~~Based on the ABWR DCD Tier 2 Subsection 15A.2.2, t~~The other (non-AOO and non-DBA) postulated abnormal events are classified as "*infrequent events*." An infrequent event *is defined as a DBE (with or without assuming a single active component failure or single operator error) with probability of occurrence of < 1/100 per year, and a radiological consequence less than an accident.*

(4) The other (non-AOO and non-infrequent incident) DBEs should be classified as accidents with DBAs as a subset. An accident *is defined as a postulated DBE that is not expected to occur during the lifetime of a plant, which equates to either an ASME Code Service Level C or D incident, and results in radioactive material releases with calculated doses comparable to (but not to exceed) the 10 CFR 50.34(a) exposures.*

(5) Historically, non-DBEs that are evaluated in BWR safety analysis reports or DCD have been termed as "special events." As no better term has been specified in a regulatory document, it is judged reasonable to maintain that term in the ESBWR DCD.

Special events

a. Are not included as design basis events in 10 CFR 50.49, and

i. are postulated in the 10 CFR regulations to demonstrate some specified prevention, coping or mitigation capabilities, without specifically requiring a radiological evaluation, and/or

ii. include a common mode equipment failure or additional failure(s) beyond the SFC.

Note: Special events do not include severe accidents or other events that are only evaluated as part of the plant PRA.

Because of the ESBWR's advanced engineering and additional redundant features, some of the abnormal events for earlier plants are classified differently for the ESBWR.

15.0.2 Abnormal Events To Be Evaluated

In selecting the AOOs to be analyzed as part of the plant safety analysis, the nuclear system parameter variations listed below are considered possible initiating causes of challenges to the fuel or the reactor coolant pressure boundary (RCPB).

- Decrease in Core Coolant Temperature
- Increase in Reactor Pressure
- Increase in Reactor Coolant Inventory
- Decrease in Reactor Coolant Inventory

The AOOs considered in the ESBWR safety analyses are listed in Table 15.0-2.

The parameter variations listed above include all the effects within the nuclear system (caused by AOOs) that can challenge the integrity of the reactor fuel or RCPB. The variation of any one parameter may cause a change in another parameter. However, for analysis purposes, challenges to barrier integrity are evaluated by groups according to the parameter variation initiating the plant challenge and which typically dominates the event response.

As discussed in Reference 15.0-2 (GESTAR II), and demonstrated in Section 15.2, the following potentially limiting AOOs are re-evaluated for each fuel reload:

- Loss of Feedwater Heating
- Closure of One Turbine Control Valve
- Turbine Trip with Bypass
- Inadvertent Isolation Condenser Initiation
- Loss of Non-Emergency AC to Station Auxiliaries

The infrequent events considered in the ESBWR safety analyses are listed in Table 15.0-2, and are discussed in detail in Section 15.3. These include reactivity and power distribution anomalies such as the Control Rod Withdrawal Error events and t~~T~~he Loss of Feedwater Heating With Failure of Selected Control Rod Run-In event, which is re-evaluated for each fuel reload.

The accidents considered in the ESBWR safety analyses are listed in Table 15.0-2.

The following accidents pose the most limiting challenge to plant design and radiological exposure limits:

- Loss of Coolant Accident (LOCA) Inside Containment
- Main Steamline Break Outside Containment
- Fuel Handling Accident

The LOCA is re-evaluated each reload as part of the process for establishing the core operating limits for new fuel types.

Each of the accidents listed in Table 15.0-2 is discussed in detail in Section 15.4.

The special events evaluated as part of the ESBWR safety analysis are listed in Table 15.0-2, and discussed in detail in Section 15.5. The Main Steamline Isolation Valve (MSIV) Closure With Flux Scram (Overpressure Protection) event is re-evaluated for each reload, to ensure that the Reactor Coolant System Pressure Safety Limit in the Technical Specifications cannot be exceeded by any Design Basis Event.

15.0.3 Determination of Safety Analysis Acceptance Criteria

Where acceptance criteria are specified in the 10 CFR regulations, those criteria or their equivalent SRP criteria shall be used. However, if an acceptance criterion in the SRP conflicts with the associated acceptance criterion in a regulation, then the criterion specified in the regulation is used. Where an acceptance criterion is not specified in the 10 CFR regulations, then the criterion in the SRP is used. Where an acceptance criterion is not specified in regulations and the SRP, then the criterion in ~~the Final Safety Evaluation Report (FSER) for the ABWR (Reference 15.0-3) or an NRC reviewed~~the Reference 1 LTR shall be used.

A listing of the ESBWR abnormal events and their event classifications and relevant SRP section is provided in Table 15.0-2. Table 15.0-2 is subject to change due to future probabilistic analyses or regulatory considerations, and thus, may be revised in the future.

15.0.3.1 Anticipated Operational Occurrences

For AOOs, the GDC 10 acceptance criterion is that "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences." To meet the intent of GDC 10, SRP 15.1 and SRP 15.2, detailed acceptance criteria for AOOs both not in combination and in combination with an additional single active component failure (SACF) or single operator error (SOE) are provided. For an AOO, which is not in combination with an additional SACF or SOE, the ~~SRP 15.1 and 15.2~~SRP 15.1 and SRP 15.2 criterion is "Fuel cladding integrity shall be maintained by ensuring that the minimum CPR remains above the MCPR safety limit for BWRs based on acceptable correlations." This is equivalent to the fuel cladding integrity (greater than 99.9% of the fuel rods in the core would be expected to avoid boiling transition) safety limit (SL) included in the ESBWR Technical Specifications (TS).

For an AOO in combination with an additional SACF or SOE, the ~~SRP 15.1 and 15.2~~SRP 15.1 and SRP 15.2 criterion is "fuel failure must be assumed for all rods for which the CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding." However, the SRP does not provide a specific radiological acceptance criterion, in the event that fuel cladding failures do occur. As AOOs are part of normal operation, GDC 60 and 10 CFR 20 apply.

The 10 CFR 20.1301(a)(1) 0.1 rem annual dose limit combined with (i.e., subtracting) the 10 CFR 20.1302(b)(2)(ii) 0.05 rem annual limit (for normal airborne releases) is the appropriate radiological acceptance limit for an AOO In Combination With An Additional SACF or SOE (i.e., an AOO with an additional single failure). This position is conservatively based on an assumption that an individual at the exclusion boundary annually receives 100% of the 10 CFR 20.1302(b)(2)(ii) 0.05 rem annual limit from normal operations (which is conservative, when compared to the 10 CFR 50, Appendix I 10 millirad ALARA annual airborne gamma dose guideline), and applying the 10 CFR 20.1301(a)(1) 0.1 rem annual dose limit. Therefore, the radiological acceptance criterion for an AOO with a single failure should generically be (0.1 - 0.05) 0.05 rem total effective dose equivalent (TEDE).

For AOOs, the GDC 15 acceptance criterion is that "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences." The equivalent criterion in ~~SRP 15.1 and 15.2~~SRP 15.1 and SRP 15.2 is "Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values," which corresponds to the ASME Code Service Level B limit. However, for completeness the Reactor Coolant System Pressure Safety Limit in the TS should be addressed.

The SRP provides an AOO related acceptance criterion that is not addressed in GDC 10 or 15, which is "An incident of moderate frequency (i.e., an AOO) should not generate a more serious plant condition without other faults occurring independently."

~~As shown in Subsection 2.4.2, the ABWR FSER (Reference 15.0-3) has a nearly equivalent set of AOO acceptance criteria.~~

Draft Rev. 3 - April 1996 version of ETSB 11-5, applies "a small fraction of 10 CFR 100 limit" (i.e., 2.5 rem total whole body) as the dose acceptance criterion.

~~The ABWR FSER NUREG-1503 Subsection 11.3.2 (page 11-11) gives the acceptance criterion for an "offgas system leak or failure as assumed in BTP ETSB 11-5, Revision 0, July 1981" as a whole body dose "less than 10 percent of the 10 CFR Part 100 limits."~~

Subsection 11.3.1 of the NRC FSER (Reference 15.0-5) for the AP1000 states "The BTP stipulates that the total body dose at the exclusion area boundary (EAB), as a result of the release of radioactivity for two hours from a postulated failure of the WGS, calculated in accordance with BTP assumptions, should not exceed 0.5 rem.... The applicant calculated a 0- to 2-hour total body dose within 0.5 rem, which satisfies BTP ETSB 11-5. Based on the above, the staff finds the analysis acceptable."

In RAI 2 to Revision 1 of Reference 15.0-1, the NRC specified that the acceptance criterion is 0.1 rem TEDE, based on the 10 CFR 20.1301(a)(i), which states " The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year."

In their review of Reference 15.0-1, the NRC stated that the use of the annual average atmospheric dispersion factor would be acceptable for a Waste Gas System Leak or Failure analysis that uses the 0.1 rem TEDE acceptance criterion. GE accepted this NRC position.

15.0.3.4.8 Potential Special Events

The 10 CFR regulations and the SRP do not contain a generic set of safety analysis acceptance criteria for special events. The safety analysis acceptance criteria for these events are on an event-specific basis. It is expected that any (potential) future special event will also have event-specific safety analysis acceptance criteria.

15.0.4 Event Analysis Format

For each event, the following information is provided in Sections 15.2, 15.3, 15.4 and 15.5.

15.0.4.1 Identification of Causes

Situations that lead to the analyzed events are described in their associated event descriptions. The frequency of occurrence of each event is summarized based upon the NSOA, currently available operating plant history for the abnormal event, and the evaluations in Appendix 15A. Events for which inconclusive data exist are discussed separately within each event section.

15.0.4.2 Sequence of Events and Systems Operations

Each event evaluated is discussed and evaluated in terms of:

- A step-by-step sequence of events from initiation to final stabilized condition.
- The extent to which normally operating plant instrumentation controls are assumed to function.
- The extent to which the plant and reactor protection systems are required to function.
- The credit taken for the functioning of normally operating plant systems.

- The operation of engineered safety systems that is required.

Each event's sequence of events is supported by the NSOA. The effect of a single equipment failure or malfunction or an operator error on the event is shown in the NSOA.

15.0.4.3 Evaluation of Results

The results of the design basis events analyses are presented in Sections 15.2, 15.3 and 15.4. The limiting events can be identified, based on those results. Reasons why the other events are not limiting are given in the event documentation.

For the core loading in Figure 4.3-1, a representative MCPR operating limit is determined. Results of the AOO analyses for individual plant-specific core loading patterns will differ slightly from the results shown in this chapter. However, the relative results between core associated events do not change. The MCPR operating limit, for the as-built initial core and each reload core fuel loading pattern, will be provided by the COL licensee to the USNRC for information.

15.0.4.4 Barrier Performance

The significant areas of interest for internal pressure damage are the high-pressure portions of the RCPB (i.e., the reactor vessel and the high pressure pipelines attached to the reactor vessel).

15.0.4.5 Radiological Consequences

This subsection describes the consequences of radioactivity releases for the core loading, during DBEs. For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For non-limiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

15.0.5 Single Failure Criterion

From 10 CFR 50, Appendix A: "A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither

- A single failure of any active component (assuming passive components function properly) nor
- A single failure of a passive component (assuming active components function properly),

results in a loss of the capability of the system to perform its safety function. Single failures of passive components in electric systems should be assumed in designing against a single failure."

The single failure criterion (SFC) requires the plant design to be capable of providing specific functions during any design basis event (DBE) assuming a single failure in addition to the event initiating occurrence and any other coincident failures specified in the required DBE analysis assumptions. The application of the SFC to:

- The total plant is described in ANSI/ANS ~~52~~58.14;
- Fluid systems are described in ANSI/ANS 58.9; and

Table 15.0-3

Safety Analysis Acceptance Criteria for AOOs

- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (i.e., not exceed ASME Code Service Level B), and the reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications.
- Fuel-cladding integrity should be maintained by ensuring that the reactor core is designed with appropriate margin during any conditions of normal operation, including the effects of AOOs. The minimum value of the critical power ratio (CPR) reached during the AOO should be such that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. (This criterion corresponds to the greater than 99.9% of the fuel rods in the core would be expected to avoid boiling transition related safety limit in the Technical Specifications.)
- Uniform cladding strain ≤ 1%~~*~~.
- No fuel centerline melt~~ (core-wide AOOs only)~~.

~~□Energy generation is < 170 cal/g (RWE during startup only).~~

- Reactor water level shall be maintained above the top of the core (i.e., active fuel).
- Containment and suppression pool pressures and temperatures shall be maintained below their design values.
- An AOO should not generate a more serious plant condition unless other faults occur independently.
- There is no loss of function of any fission product barrier (Safety/Relief Valve or Depressurization Valve discharge does not apply).

~~* Based on SRP Sections 15.4.1 and 15.4.2, for the Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition (i.e., control rod withdrawal error [RWE] during startup) event and the Uncontrolled Control Rod Assembly Withdrawal At Power (i.e., RWE during power operation) event.~~