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Subject: **Submittal of Licensing Topical Report, "Classification of ESBWR Abnormal Events and Determination of Their Safety Analysis Acceptance Criteria," NEDO-33175, December 2004**

The subject Licensing Topical Report (LTR) is hereby submitted for NRC review and approval. GE developed this LTR to establish a clear basis for the ESBWR new plant design with regard to event classification and acceptance criteria. The two reasons that prompted this LTR are: 1) the new passive design features in the ESBWR required classification; and 2) the perceived need for clarification of existing regulatory inconsistencies with regard to classification of events that are less probable than an anticipated operational occurrence (AOO) but more probable than a design basis accident (DBA), and their associated acceptance criteria.

The LTR describes GE's review of the following documents, in order of their regulatory hierarchy, as part of the process to determine consistent abnormal event classifications and their associated acceptance criteria for the ESBWR:

- 10 CFR Regulations
- Standard Review Plan, Chapter 15
- Regulatory Guide 1.70, Chapter 15
- The Final Safety Evaluation Report for the Advanced Boiling Water Reactor

Where there was a conflict/inconsistency between multiple guidance documents, the highest hierarchy document was used as the basis for the final determination contained in the LTR.

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The result of this evaluation is a consistent set of event classifications and acceptance criteria for the ESBWR, as described in the LTR. The LTR also contains a preliminary list of ESBWR Abnormal Event Classifications (Table 2). This list only is provided for information, and is subject to change as the ESBWR design and PRA development progresses. Some unique ESBWR design features that may influence the abnormal event classifications include:

- Elimination of recirculation system
- Increase in RPV volume
- Incorporation of sealless FMCRDs
- Dedicated IC lines
- Option to bypass 110% steam

GE is requesting approval of this LTR to establish the ESBWR design and licensing basis for the classification of abnormal events and their acceptance criteria, but not for the final determination of the specific events, which fit into each event classification.

If you have any (informal) questions, please contact Kurt Schaefer at (408) 925-2426. Please let us know if we can provide any additional information to assist in your review. If needed, we request a meeting with the NRC at the earliest possible opportunity to discuss this LTR in more detail.

Sincerely,

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Enclosure:

MFN 04-139 – NEDO-33175, *Classification of ESBWR Abnormal Events and Determination of Their Safety Analysis Acceptance Criteria*, December 2004

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ENCLOSURE

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Classification of ESBWR Abnormal Events
and Determination of Their Safety Analysis Acceptance Criteria

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CLASSIFICATION OF ESBWR ABNORMAL EVENTS AND DETERMINATION OF THEIR SAFETY ANALYSIS ACCEPTANCE CRITERIA

K. T. Schaefer

GE Nuclear Energy

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Licensing Topical Report

CLASSIFICATION OF ESBWR ABNORMAL EVENTS AND DETERMINATION OF THEIR SAFETY ANALYSIS ACCEPTANCE CRITERIA

Approved by: George Stramback, Manager
Regulatory Services

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EXECUTIVE SUMMARY

For the ESBWR passive plant design, the regulatory criteria for event classification were reviewed to determine the appropriate abnormal event classifications and their associated safety analysis acceptance criteria. This review included, in order of their regulatory priority, the 10 CFR regulations, USNRC Standard Review Plan (SRP) Section 15, Regulatory Guide 1.70 (RG 1.70) Chapter 15, and the Final Safety Evaluation Report (FSER) for the Advanced Boiling Water Reactor (ABWR) Design Chapter 15.

The SRP and RG 1.70 together use the following terms to classify the non-accident design basis events.

- *anticipated operational occurrences*
- *transients*
- *anticipated transients combined with the worst single failure*
- *anticipated transients*
- *moderate-frequency transients*
- *most limiting transients*
- *incident of moderate frequency in combination with any single active failure, or operator error*
- *initiating events which are expected to occur with moderate frequency*
- *initiating events and associated transients*
- *moderate-frequency events*
- *incident of moderate frequency*
- *incident of moderate frequency in combination with any single active failure, or operator error*
- *events having a moderate-frequency of occurrence*
- *incident of moderate frequency with a single active component failure*
- *anticipated frequency classification*
- *off-design transients*
- *normal operational occurrences*
- *infrequent incidents*

Only the term *anticipated operational occurrences* (AOOs) appears in the 10 CFR regulations. In total, the SRP and RG 1.70 use eighteen different terms to classify the non-accident design basis events, and (except for AOOs) none of those classification terms is defined or quantified in either the 10 CFR regulations or SRP 15.

The SRP and the 10 CFR regulations do not have a concise classification term to identify events that are less probable than an AOO but more probable than a design basis accident (DBA). Therefore, one is developed herein.

The 10 CFR regulations, SRP Section 15 and RG 1.70 Chapter 15 have no classification terms for beyond design basis events such as Anticipated Transient Without Scram (ATWS) and other

events that assume failures beyond the single failure criterion. However, SECY-94-084, Section A.I (Scope and Criteria) specifically addresses ATWS and Station Blackout as "beyond design basis."

Because (a) the SRP and RG 1.70 are not consistent, lack definitions and quantifications, and lack consistency with the 10 CFR regulations, and (b) the fact that the 10 CFR regulations have the highest regulatory priority, only the 10 CFR regulations are used to determine the abnormal event classifications for the ESBWR.

The 10 CFR regulations, SRP and ABWR FSER do provide useful guidance with respect to abnormal event safety analysis acceptance criteria. Reviews of the acceptance criteria in the 10 CFR regulations and SRP, and their application discussed in the ABWR FSER were used to develop consistent sets of safety analysis acceptance criteria for all of the ESBWR event classifications.

1. INTRODUCTION

For the ESBWR passive plant design, the abnormal event classifications have to be established for the first time. In addition, the associated safety analysis acceptance criteria for the event classifications have to be determined. There is no regulation that requires or implies that the licensing basis of a new plant design (like the ESBWR) should or shall be based on past plant licensing bases. Therefore, to establish the ESBWR plant-specific licensing basis for the classification of abnormal events and the determination of their safety analysis acceptance criteria, first the 10 CFR regulations and second the associated NRC published review and guidance documents are reviewed.

The following, in order of their regulatory priority, the 10 CFR regulations, Standard Review Plan (SRP), Regulatory Guide 1.70 (RG 1.70) and the Final Safety Evaluation Report (FSER) for the Advanced Boiling Water (ABWR) are reviewed, as part of the process to determine consistent sets of abnormal event classifications and their associated acceptance criteria for the ESBWR. Where there is a conflict/inconsistency between multiple documents, the highest priority document is used as the basis for the final determination.

Event classifications for earlier BWRs have been inconsistent, and thus, a secondary goal here is to not carry such inconsistencies forward with regard to the ESBWR. For example, RG 1.70, Safety Analysis Report (SAR) Chapter 15 has been titled "*Accident Analyses*." However, most of safety analyses within Chapter 15 are not classified as accidents. The 10 CFR regulations, SRP Section 15, RG 1.70 Chapter 15, and the FSER for the ABWR Design Chapter 15 (References 1 through 4, respectively) use multiple, conflicting and inconsistent terms to classify/categorize the non-accident abnormal events that are reported in a SAR or a Design Control Document (DCD). These types of inconsistencies can also be found in the assignment of acceptance criteria for the different classifications of events.

2. REVIEW OF REGULATORY DOCUMENTS

The following only summarizes the findings from searching the 10 CFR regulations, SRP, RG 1.70 and the FSER for the ABWR for event classification terms and safety analysis acceptance criteria. These findings are limited to the different event classification terms and the NRC acceptance criteria from each reference source. The conclusions based on these findings, including the final abnormal event classifications and their acceptance criteria, are presented in Sections 3 and 4. Also, a statement from SECY-94-084 (Reference 5), with regard to ATWS and Station Blackout, is provided.

2.1 10 CFR Regulations

2.1.1 *Classification Terms*

From 10 CFR 50.2, "*Design bases* means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or

experiments) of the effects of a *postulated accident* for which a structure, system, or component must meet its functional goals."

10 CFR 50.2 states "*Safety-related structures, systems and components* means those structures, systems and components that are relied upon to remain functional during and following *design basis events* to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) The capability to prevent or mitigate the consequences of *accidents* which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable."

Like criterion (3) above, an event that is termed as an *accident* is usually associated with a consequence in the form of a radiological dose.

10 CFR 50.2 states "*Safe shutdown* (non-design basis accident (non-DBA)) for *station blackout* means bringing the plant to those shutdown conditions specified in plant technical specifications as Hot Standby or Hot Shutdown, as appropriate (plants have the option of maintaining the RCS at normal operating temperatures or at reduced temperatures)." Plus, 10 CFR 50.2 states "*Station blackout* means the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). Station blackout does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources as defined in this section, nor does it assume a concurrent single failure or design basis accident. At single unit sites, any emergency ac power source(s) in excess of the number required to meet minimum redundancy requirements (i.e., single failure) for *safe shutdown (non-DBA)* is assumed to be available and may be designated as an alternate power source(s) provided the applicable requirements are met." Therefore, *station blackout* (SBO), which has an annual probability of occurrence < 1/60, is not classified as either a DBA or an AOO.

10 CFR 50.49(b)(1)(ii) states "*Design basis events* are defined as conditions of *normal operation*, including *anticipated operational occurrences*, *design basis accidents*, *external events*, and *natural phenomena* for which the plant must be designed to ensure" the safety-related functions. Therefore, all other abnormal events that are not *anticipated operational occurrences* (AOOs), *design basis accidents* (DBAs), *external events*, and *natural phenomena* (by regulatory definition) must be classified as *beyond design basis events*.

10 CFR 50.62 states an "*Anticipated Transient Without Scram* (ATWS) means an *anticipated operational occurrence* as defined in appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of appendix A of this part." For an ATWS event to happen, a common-mode failure and multiple other independent failures have to simultaneously occur. The scenario assumes failures far beyond the single failure criterion (SFC) in 10 CFR 50 App. A, and has a probability of << 1/60 per year, and thus, is not classified as either an AOO or a DBA.

10 CFR 50 App. A provides an explicit definition of an *anticipated operational occurrence* (AOO). 10 CFR 50 App. 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 shall be classified as an AOO, and conversely, any abnormal event with a probability < 1/60 per year shall not be classified as an AOO.

10 CFR 50 App. A provides an explicit definition of a *loss of coolant accident* (LOCA). 10 CFR 50 App. A states "*Loss of coolant accidents* mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system."

10 CFR 50 App. A provides an explicit definition of a *single failure*. 10 CFR 50 App. A states "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 (1) a *single failure* of any active component (assuming passive components function properly) nor (2) 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 functions."

10 CFR 50 App. A, General Design Criteria (GDC) 10 and 15 apply to "any condition of *normal operation*, including the effects of *anticipated operational occurrences*." Per GDC 10, GDC 15 and the definition of an AOO, an AOO is considered as part of *normal operation*, and thus, an AOO can not be classified as an *accident*, and has more conservative acceptance criteria than an *accident*.

GDC 17, 20, 22, 26, 27, 29, 31, 41, 55, 60 and 64 each address *anticipated operational occurrences* and/or *postulated accidents*, with no additional clarification. GDC 28 addresses *postulated reactivity accidents*, with no additional clarification. GDC 36, 46, 50 and 64 address *LOCA*s, with no additional clarification. Because AOOs and LOCAs are defined in 10 CFR 50 App. A, no additional clarification is needed.

10 CFR 50 App. B "establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components." "Nuclear power plants and fuel reprocessing plants include structures, systems, and components that prevent or mitigate the *consequences of postulated accidents* that could cause undue risk to the health and safety of the public." "The pertinent requirements of this appendix apply to all activities affecting the safety-related functions." The (10 CFR 50.2) safety-related function related to accidents applies to "prevent or mitigate the consequences of *accidents* which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable."

10 CFR 100.1(c), "Purpose," states "Siting factors and criteria are important in assuring that radiological doses from normal operation and postulated accidents will be acceptably low."

Therefore, 10 CFR 100 only recognizes two event classifications, *normal operation* and *postulated accidents.*

10 CFR 100.10(a)(4), "Factors to be considered when evaluating sites," states "The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur." Therefore, a *breach* of the fission product barrier that results in *a release of radioactive material* constitutes an *accident.*

10 CFR 100 App. A establishes *Safe Shutdown Earthquake* (SSE), which has commonly been referred to as the *Design Basis Earthquake,*" criteria. These criteria are applied to those "structures, systems, and components are those necessary to assure:

The integrity of the reactor coolant pressure boundary,

The capability to shut down the reactor and maintain it in a safe shutdown condition, or

The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part."

The 10 CFR regulations have no classification terms for *beyond design basis events* such as ATWS, SBO and other events that assume failures beyond the single fail criterion. For the SBO event, 10 CFR 50.2 specifies that an SBO as "*non-design basis accident.*"

2.1.2 *Acceptance Criteria*

GDC 10 - Reactor design states "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."

GDC 15 - Reactor coolant system design *states* "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."

GCD 17 - Electric power systems states "An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified *acceptable fuel design limits* and *design conditions of the reactor coolant pressure boundary* are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of *postulated accidents.*"

GDC 19 - Control room states "Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."

GDC 38 - Containment heat removal states "A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels."

10 CFR 20.1301 regulates limits for individual members of the public. Paragraph (a)(2) states "The dose in any unrestricted area from external sources, ..., does not exceed 0.002 rem (0.02 mSv) in any one hour."

10 CFR 50.34(a)(1)(ii)(D)(*1*) states "An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE)."

10 CFR 50.34(a)(1)(ii)(D)(*2*) states "An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE)."

10 CFR 50.46(a)(3)(b) provides the acceptance criteria for the ECCS-LOCA Performance Analysis as follows:

"(1) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) *Maximum cladding oxidation.* The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

(3) *Maximum hydrogen generation.* The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) *Coolable geometry.* Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) *Long-term cooling.* After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat

shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

10 CFR 50.62 addresses the ATWS event, but does not specifically require a performance/safety analysis to be performed, and thus, does provide acceptance criteria for an ATWS performance analysis. However, generic BWR ATWS performance analysis acceptance criteria are in Section 4 of Reference 6, which was used by the NRC in generating 10 CFR 50.62. These acceptance criteria are provided in Section 4.4.

10 CFR 50.67(b)(2) states

"(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident."

10 CFR 100.11(a)(1) states "An exclusion area of such size that an individual located at any point on its boundary for <u>two hours</u> immediately following onset of the postulated fission product release would not receive a total radiation dose to the <u>whole body in excess of 25 rem</u> or a total radiation dose in excess of <u>300 rem to the thyroid from iodine exposure</u>."

10 CFR 100.11(a)(2) states "A low population zone of such size that an individual located at any point on its outer boundary who is <u>exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage)</u> would not receive a total radiation dose to the <u>whole body in excess of 25 rem</u> or a total radiation dose in excess of <u>300 rem to the thyroid from iodine exposure</u>."

The radiation dose acceptance criteria from 10 CFR Parts 34, 67 and 100 are consistent. Therefore, for the most part, the following subsections only refer to 10 CFR 100.

2.2 NUREG-0800 Standard Review Plan Section 15

SRP sections that evaluate events or criteria that are not applicable to the ESBWR (e.g., PWR only) are not addressed in the following subsections. (Note: In multiple statements, the SRP does not specify the correct title to the current revision of Regulatory Guide 1.105. In the following subsections, where such SRP statements are quoted, the correct title is provided.)

2.2.1 *Classification Terms*

SRP 15.0 uses the terms "*anticipated operational occurrences*" and "*postulated accidents*" in some paragraphs, and the terms "*transients*" and "*accidents*" in another paragraph. However, it does not define any of those terms.

SRP 15.0 states "BWR applicants must demonstrate that for *anticipated transients combined with the worst single failure* and assuming proper operator actions, the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery." However, the SRP does not define "*anticipated transient.*"

Draft SRP 15.0.2 states:

- 'In order to establish a licensing basis, licensees must analyze *transients* and *accidents* per the requirements of 10 CFR 50.34, 10 CFR 50.46, and where applicable, per NUREG-0737, "Clarification of TMI Action Plan Requirements."
- The guidance in this section should be applicable to most of the *transients* and *accidents* described in the SRP.
- In this SRP, *accidents* and *transients* refer to those events that are defined in NUREG-0800 to be analyzed to meet the requirements of the General Design Criteria (GDC).'

However, of the terms used in SRP 15.0 through Draft SRP 15.0.2, only the term "*anticipated operational occurrences*" is used in the GDC.

SRP 15.1.1 - 15.1.4, Section II effectively divides "*transients*" into "*moderate-frequency transients,*" "*most limiting transients,*" and "*incident of moderate frequency in combination with any single active failure, or operator error*" without specifically defining any of these terms. SRP 15.1 states

"for the *most limiting transients*, the plant responds to the *transients* in such a way that acceptance criteria regarding fuel damage and system pressure are met;"

"An *incident of moderate frequency in combination with any single active failure, or operator error* shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such *accidents*,...;" and

"The term '*moderate-frequency*' is used in this SRP section in the same sense as in the descriptions of design and plant process conditions in References 9 and 10," which are GDC 10 and 15. However, GDC 10 and 15 do not use the term "*moderate-frequency.*" These GDC use the term "*anticipated operational occurrences.*"

Draft Rev. 2 of SRP 15.2.1 - 15.2.5, Section I uses the terms "*initiating events which are expected to occur with moderate frequency,*" "*initiating events and associated transients,*" while Section II uses the terms "*anticipated operational occurrences,*" "*moderate-frequency events,*" "*incident of moderate frequency,*" "*transients,*" "*incident of moderate frequency in combination with any single active failure, or operator error.*" None of these terms is specifically defined in that SRP, and are almost used interchangeably, except that "*incidents of moderate frequency,*" and an "*incident of moderate frequency with a single active component failure*" have different acceptance criteria.

The use of abnormal event terms and their lack of definition in **Draft Rev. 2 of SRP 15.2.6, Draft Rev. 2 of SRP 15.2.7, and Draft Rev. 3 of SRP 15.4.1**, are similar to the above SRPs.

Draft Rev. 3 of SRP 15.4.2 consistently uses the term "*anticipated operational occurrence.*"

For a fuel-loading error, **Draft Rev. 2 of SRP 15.4.7** uses the terms "*events having a moderate-frequency of occurrence*" and "*accidents,*" and specifies an acceptance criterion of a "*small fraction of the 10 CFR Part 100 guidelines.*"

Rev. 2 of SRP 15.5.1 - 15.5.2, Section I states "This Standard Review Plan (SRP) section is intended to be applicable to *moderate frequency events* that increase reactor coolant inventory. These *transients* ...." A note to Section I states that the term "*frequent*" is used in that SRP as used in Section 15.2 of the ABWR FSER, however, Section 15.2 of the ABWR FSER does not contain the term "*frequent.*"

Rev. 2 of SRP 15.6.1 uses the terms "*transient,*" "*anticipated frequency classification,*" "*incidents of moderate frequency,*" "*incident of moderate frequency with a single active component failure,*" and "*anticipated operational occurrence*" almost interchangeably, except that "*incidents of moderate frequency,*" and an "*incident of moderate frequency with a single active component failure*" have different acceptance criteria.

SRPs 15.6.2, 15.6.4, 15.6.5, 15.7.4 and 15.7.5 address *accident* radioactive material releases.

SRP 15.8 refers to "*anticipated transients,*" but does not correlate it to any of the other events used in the SRPs.

2.2.1.1 Summary Conclusion

In total, the SRP inconsistently uses fifteen different terms to classify the non-DBA events, and (except for AOOs) none of those classification terms are defined or quantified in either the 10 CFR regulations or SRP 15. The SRP and the 10 CFR regulations do not have a concise classification term to identify events that are less probable than an AOO but more probable than a DBA. Therefore, for these other (non-AOO and non-DBA) events a new classification term (based on the 10 CFR regulations) is developed and defined within Section 3.

SRP Section 15 has no classification term for beyond design basis events such as ATWS and other events that assume failures beyond the single fail criterion.

2.2.2 *Acceptance Criteria*

SRP 15.0 states "BWR applicants must demonstrate that for *anticipated transients* combined with the worst single failure and assuming proper operator actions, the *core remains covered* or provide analysis to show that *no significant fuel damage results from core uncovery.*"

Draft Rev. 2 of SRP 15.1.1 - 15.1.4, Section I, Areas of Review states "The results of the transient analysis are reviewed to ensure that the values of pertinent system parameters are within the ranges expected for the type and class of reactor under review. The parameters include: core flow and flow distribution, channel heat flux (average and hot), minimum critical power ratio (MCPR), vessel water level, thermal power, vessel pressure, steam line pressure (for BWRs), steam line flow (for BWRs), feedwater flow (for BWRs), and reactivity."

Draft Rev. 2 of SRP 15.1.1 - 15.1.4, Section II, Acceptance Criteria states "The specific criteria necessary to meet the requirements of GDC 10, 15, 20, and 26 for incidents of moderate frequency are:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.

2. Fuel cladding integrity shall be maintained by ensuring that the minimum ... CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

4. An incident of moderate frequency in combination with any single active component failure, or single operator error shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, 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 (see SRP Section 4.2) that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

5. To meet the requirements of General Design Criteria 10, 15, 20 and 26 the positions of Regulatory Guide 1.105, "Setpoints For Safety-Related Instrumentation," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.

6. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53."

Draft Rev. 2 of SRP 15.2.1 - 15.2.5, Section II, Acceptance Criteria states "The criteria for incidents of moderate frequency are:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.

b. Fuel cladding integrity shall be maintained by ensuring that the minimum ... critical power ration (CPR) remains above the minimum critical power ration (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, 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 (see SRP Section 4.2) that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding."

Draft Rev. 2 of SRP 15.2.6, Section II, Acceptance Criteria states "Specific criteria necessary to meet the relevant requirements of GDC 10, 15, and 26 for events of moderate frequency* are as follows:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.
2. Fuel cladding integrity shall be maintained by ensuring that the minimum ... critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).
3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
4. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failures 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 (see SRP Section 4.2), that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.
5. To meet the requirements of General Design Criteria 10 and 15, the positions of Regulatory Guide 1.105, "Setpoints For Safety-Related Instrumentation," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.
6. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53."

Draft Rev. 2 of SRP 15.2.7, Section II, Acceptance Criteria states "Specific criteria necessary to meet the relevant requirements of GDC 10, 15, 17, and 26 for events of moderate frequency* are as follows:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.
b. Fuel cladding integrity shall be maintained by ensuring that the ... CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).
c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
d. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, 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 (see SRP Section 4.2), that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

e. To meet the requirements of General Design Criteria 10 and 15, the positions of Regulatory Guide 1.105, "Setpoints For Safety-Related Instrumentation," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.

f. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be Identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53 and GDC 17."

Draft Rev. 3 of SRP 15.4.1, Section II, Acceptance Criteria, for an Uncontrolled Control Rod Assembly Withdrawal (i.e., Rod Withdrawal Error for a BWR) From A Subcritical or Low Power Startup Condition, states "The requirements of GDC 10, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when:

a. The thermal margin limits (... MCPR for BWRs) as specified in SRP Section 4.4 are met.

b. (for PWRs)

c. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2 does not exceed 1%."

Draft Rev. 3 of SRP 15.4.2, Section II, Acceptance Criteria, for an Uncontrolled Rod Assembly Withdrawal (i.e., Rod Withdrawal Error for a BWR) At Power, states "The requirements of General Design Criteria 10, 17, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when:

a. The thermal margin limits (... MCPR for BWRs) as specified in SRP Section 4.4 are met.

b. (for PWRs)

c. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2 do not exceed 1%."

Draft Rev. 2 of SRP 15.4.7, Section II, Acceptance Criteria, for an Inadvertent Loading and Operation Of A Fuel Assembly In An Improper Position (i.e., Fuel Loading Error for a BWR), states "the following acceptance criteria are necessary to cover the event of operation with misloaded fuel caused by loading errors:

a. To meet the requirements of GDC 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel loading errors after fueling operations.

b. In the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 guidelines."

Draft Rev. 2 of SRP 15.4.7, Section II also specifies "A small fraction is interpreted to be less than 10% of the 10 CFR Part 100 reference values."

Draft Rev. 3 of SRP 15.4.9, Section II, Acceptance Criteria, for a Spectrum of Rod Drop Accidents (non-radiological analysis), states "acceptance criteria are based on meeting the requirements of General Design Criterion 28 (GDC 28) as it relates to the effects of postulated reactivity accidents, neither resulting in damage to the reactor coolant pressure boundary greater than limited local yielding nor causing sufficient damage to impair significantly the capacity to cool the core. Specific criteria necessary to meet the relevant requirements of GDC 28 are as follows:

1. Reactivity excursions should not result in radially averaged fuel rod enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

2. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Code.

3. The number of fuel rods predicted to reach assumed fuel failure thresholds and associated parameters such as the amount of fuel reaching melting conditions will be an input to a radiological evaluation. The assumed failure thresholds are a radially averaged fuel rod enthalpy greater than 170 cal/gm at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions."

Draft Rev. 3 of SRP 15.4.9, Appendix A, Section II, Acceptance Criteria, for Radiological Consequences of Control Rod Drop Accident, states

> "The acceptance criteria are based on the requirements of 10 CFR Part 100 as related to mitigating the radiological consequences of an accident. The plant site and dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated control rod drop accident if the calculated whole-body and thyroid doses at the exclusion area boundaries (EAB) and at the low population zone (LPZ) boundaries are well within the exposure guideline values in 10 CFR Part 100, paragraph 11. "Well within" is defined as 25% of the 10 CFR Part 100 exposure guideline values or 750 mSv (75 rem) for the thyroid and 60 mSv (6 rem) for whole-body doses.
>
> The fission product source term used in the dose analysis is acceptable if it meets the guidelines of Regulatory Guide 1.77."

Draft Rev. 2 of SRP 15.6.1, Section II, Acceptance Criteria, for Inadvertent Opening of a Safety Relief Valve (SRV), states "The specific criteria necessary to meet the requirements of General Design Criteria 10, 15, and 26 for incidents of moderate frequency are:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.

b. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failures must be assumed for all rods for which the DNBR or CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

e. To meet the requirements of General Design Criteria 10, 15, and 26, the positions of Regulatory Guide 1.105, "Setpoints For Safety-Related Instrumentation," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.

f. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53."

Draft Rev. 3 of SRP 15.6.2, Section II, Acceptance Criteria, for Radiological Consequences of The Failure of Small Lines Carrying Primary Coolant Outside Containment, states "The acceptance criteria for this SRP section are based on the relevant requirements of the following regulations:

1. GDC 55, as it relates to the isolation requirements of small-diameter lines connected to the primary system that are acceptable on the basis of meeting item (2) below.

2. 10 CFR 100.11 and Regulatory Guide 1.11, as they relate to the radiological consequences of a small line break carrying primary coolant outside containment.

   The plant site and the dose mitigating engineered safety feature (ESF) systems are acceptable with respect to the radiological consequences of a postulated failure outside the containment of a small line carrying reactor coolant if the calculated whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries are substantially below the exposure guideline values of 10 CFR 100.11 as stated in position C.1.b of Regulatory Guide 1.11. "Substantially below" the exposure guideline values of 10 CFR Part 100.11 means 10% of these exposure guideline values, that is, 25 mSv and 300 mSv (2.5 rem and 30 rem) for the whole-body and thyroid doses, respectively."

Draft Rev. 3 of SRP 15.6.4, Section II, Acceptance Criteria, for Radiological Consequences of Main Steam Line Failure Outside Containment (i.e., Main Steamline Break Accident), states "The acceptance criteria are based on the requirements of 10 CFR Part 100 as related to the radiological consequences of an accident. The plant site and the dose mitigating engineered safety features (ESF) are acceptable with respect to the radiological consequences of a postulated MSLB outside containment of a BWR facility if the calculated whole body and thyroid doses at the exclusion area and the low population zone boundaries do not exceed the following exposure guidelines:

1. For an MSLB with an assumed pre-accident iodine spike corresponding to the maximum iodine concentration stated in the nuclear steam supply system (NSSS) vendor's standard technical specifications, the calculated doses should not exceed the guideline values of 10 CFR 100.11.

2. For an MSLB with an assumed iodine concentration corresponding to the equilibrium value for continued full-power operation stated in the NSSS vendor standard technical specifications, the doses should not exceed a small fraction of the above guideline values, i.e., 10% or 25 mSv (2.5 rem) and 300 mSv (30 rem) respectively, for the whole body and thyroid doses.

3. The methodology and assumptions for calculating the radiological consequences should reflect the regulatory positions of Regulatory Guide 1.5 except for the atmospheric dispersion factors which are reviewed under SRP Section 2.3.4.

4. A plant specific technical specification is required for both cases of iodine activity in the primary coolant. This specification is acceptable if the calculated potential radiological consequences from the MSLB accident are within the exposure guidelines for the above two cases."

Draft Rev. 3 of SRP 15.6.5, Section II, Acceptance Criteria, for Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks With the Reactor Coolant Pressure Boundary (i.e., LOCA inside containment), states "Specific criteria necessary to meet the relevant requirements of the 10 CFR regulations identified above and necessary to meet the TMI Action Plan requirements are as follows:

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an evaluation model that satisfies the requirements of 10 CFR 50.46. Regulatory Guide 1.157 and Section I of Appendix K to 10 CFR Part 50 provide guidance on acceptable evaluation models. For the full spectrum of reactor coolant pipe breaks, and taking into consideration requirements for reactor coolant pump operation during a small break loss-of-coolant accident, the results of the evaluation must show that the specific requirements of the acceptance criteria for ECCS are satisfied as given below:

   a. The calculated maximum fuel element cladding temperature does not exceed 1200°C (2200°F)

   b. The calculated total oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

   c. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

   d. Calculated changes in core geometry are such that the core remains amenable to cooling.

e. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

2. The radiological consequences of the most severe LOCA are within the guidelines of 10 CFR Part 100. Appendices A, B and D to this SRP section provides the results of the LOCA analysis.

3. The TMI Action Plan requirements for II.E.2.3, II.K.2.8, II.K.3.5, II.K.3.30, II.K.3.31, and II.K.3.40 have been met."

Draft Rev. 2 of SRP 15.6.5, Appendix A, Section II, Acceptance Criteria, for Radiological Consequences of a Design Basis Loss-of-Coolant Accidents Including Containment Leakage Contribution, states "The acceptance criteria are based on the requirements of 10 CFR Part 100 as related to mitigating the radiological consequences of an accident. Specific acceptance criteria for the total calculated doses and for the containment leakage contribution are as follows:

1. The distances to the exclusion area boundary and to the low population zone outer boundary are acceptable if the total calculated radiological consequences (i.e., thyroid and whole body doses) for the hypothetical LOCA fall within the appropriate exposure guideline values specified in 10 CFR 100.11(a). The total dose is the combined dose from all release paths from the containment to the atmosphere. At the construction permit (CP) review stage, the staff applies exposure guideline values of 1.5 Sv (150 rem) to the thyroid and 0.2 Sv (20 rem) to the whole body in accordance with Regulatory Guides 1.3 and 1.4, SRP Section 2.3.4, SRP Section 6.5.5. This is to allow for uncertainties in meteorology and other site-related data and to allow for system design changes that might influence the final design of engineered safety features or the dose reduction factors of these features. These lower values are applied at the CP stage to provide reasonable assurance that the 10 CFR Part 100 guideline values can be met at the operating license (OL) review stage. For reviews leading to an early site permit or a combined license (COL), the radiation dose guidelines specified in 10 CFR Part 100 are used: 3.0 Sv (300 rem) to the thyroid and 0.25 Sv (25 rem) to the whole body.

2. The model for and the calculation of the post-LOCA leakage contribution to the total whole body and thyroid doses of a hypothetical LOCA are acceptable if they incorporate the appropriate conservative design basis assumptions outlined in the regulatory positions of Regulatory Guide 1.3 for a BWR facility, with the following exceptions: (a) the acceptability of guidelines for atmospheric dispersion factors (χ/Q values) is determined under SRP Section 2.3.4."

Draft Rev. 2 of SRP 15.6.5, Appendix B, Section II, Acceptance Criteria, for Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment, states "The acceptance criteria are based on the requirements of 10 CFR Part 100.11(a) as related to mitigating the radiological consequences of an accident. Specific criteria necessary to meet this requirement are as follows:

(1) ESF systems that circulate water outside the containment are assumed to leak during their intended operation (e.g., valve stem leakage) and as a result of a failure of a passive

component. Both types of leakage are included in the review. ESF atmosphere filtration systems should be provided in those areas where such leakage is postulated to occur in order to mitigate the radiological consequences from the fission product release.

(2) The postulated radiological consequences from leakage should be calculated using conservative assumptions. 50% of the core iodine inventory, based upon the maximum reactor power level, should be assumed to be mixed in the sump water being circulated through the containment external piping systems, in accordance with the values listed in Table 1 of Regulatory Guide 1.7. The atmospheric dispersion factors (χ/Q values) as determined under SRP Section 2.3.4 should be used in the analysis.

(3) The postulated radiological consequences from ESF component leakage, as calculated by the staff, should be combined, under SRP Section 15.6.5, Appendix A, with the consequences from other fission product release paths to determine the total calculated radiological consequences from the hypothetical LOCA. The acceptability of the site, with respect to the total radiological consequences, is determined by the adequacy of the exclusion area and low population zone outer boundary distances in conjunction with the operation of dose-mitigating ESF systems. For operating license (OL), combined license (COL), or early site permit applications, the total doses should be within the exposure guidelines of 10 CFR 100.11(a). For a construction permit application, the total doses should be within the guideline value of Regulatory Guides 1.3 and 1.4, as appropriate. This acceptability is determined under SRP Section 15.6.5, Appendix A."

Draft Rev. 2 of SRP 15.6.5, Appendix D, Section II, Acceptance Criteria, for Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Main Steam Isolation Valve, states "For operating license (OL), combined license (COL), or early site permit applications, the total doses should be within the exposure guidelines of 10 CFR 100.11(a). For a construction permit application the staff applies exposure guideline values of 1.5 Sv (150 rem) to the thyroid and 0.2 Sv (20 rem) to the whole body in accordance with Regulatory Guide 1.3, and SRP Section 2.3.4. The acceptability is determined under SRP Section 15.6.5, Appendix A."

SRP 15.7.3, Section II, Acceptance Criteria, for Postulated Radioactive Releases Due to Liquid-Containing Tank Failures, states "acceptance criteria are based on meeting the relevant requirements of the following regulations:

1. General Design Criterion 60 as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment.

2. 10 CFR Part 20 as it relates to radioactivity in effluents to unrestricted areas. Tanks and associated components containing radioactive liquids outside containment are acceptable if failure does not result in radionuclide concentrations in excess of the limits in 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply, in an unrestricted area, or if special design features are provided to mitigate the effects of postulated failures for systems not meeting these limits."

Draft Rev. 2 of SRP 15.7.4, Section II, Acceptance Criteria, for Radiological Consequences of Fuel Handling Accidents, states "The PERB acceptance criteria for this SRP section are based on requirements of 10 CFR Part 100 with respect to the calculated radiological consequences of a

fuel handling accident and General Design Criterion 61 with respect to appropriate containment, confinement, and filtering systems. Specific criteria necessary to meet the requirements are:

1. The plant site and dose mitigating ESF systems are acceptable with respect to the radiological consequences of a postulated fuel handling accident if the calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are well within the exposure guideline values of 10 CFR Part 100, paragraph 11. 'Well within' means 25 percent or less of the 10 CFR Part 100 exposure guideline values, i.e., 750 mSv (75 rem) for the thyroid and 60 mSv (6 rem) for the whole-body doses.

2. The radioactivity control features of the fuel storage and handling systems inside containment and in the fuel building are acceptable if they meet the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," with respect to appropriate containment, confinement, and filtering systems.

3. The model for calculating the whole-body and thyroid doses is acceptable if it incorporates the appropriate conservative assumptions in Regulatory Guide 1.25, with the exception of the guidelines for the atmospheric dispersion factors (χ/Q values). The acceptability of the χ/Q values is determined under SRP Section 2.3.4. The source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing, may be modified by other Staff approved source terms and methodologies such as those contained in NUREG-1465, 'Accident Source Terms for Light-Water Nuclear Power Plants.'

4. An ESF-grade atmosphere cleanup system is required for the spent fuel storage area to reduce the potential radiological consequences.

5. The containment design is acceptable with respect to a postulated fuel handling accident if it possesses the capability for prompt radiation detection by use of redundant radiation monitors and automatic isolation if fuel handling operations inside containment occur when the containment is open to the environment (i.e., with a containment purge exhaust system). An acceptable alternative approach is containment venting through an ESF atmosphere cleanup system or containment isolation during fuel handling operations."

Draft Rev. 3 of SRP 15.7.5, Section II, Acceptance Criteria, for Spent Fuel Cask Drop Accidents, states "The PERB (Emergency Preparedness and Radiation Protection Branch) acceptance criteria for this SRP section are based on the requirements of 10 CFR Part 100 with respect to the calculated radiological consequences of a spent fuel cask drop accident and General Design Criterion 61 with respect to appropriate containment, confinement and filtering systems.

1. The plant site and dose mitigating ESF systems are acceptable with respect to the radiological consequences of a postulated spent fuel cask drop accident if the calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are well within the exposure guideline values of 10 CFR Part 100, paragraph 11. 'Well within' means 25 percent or less of the 10 CFR Part 100 exposure guideline values, i.e., 750 mSv (75 rem) for the thyroid and 60 mSv (6 rem) for the whole-body doses.

2. The radioactivity control features of the fuel storage and spent fuel cask handling system in the fuel building are acceptable if they meet the requirements of General Design Criterion 61,

'Fuel Storage and Handling and Radioactivity Control,' with respect to appropriate containment, confinement and filtering systems.

3. The model for calculating the whole-body and thyroid doses is acceptable if it incorporates the appropriate conservative assumptions in NUREG-1465 with respect to gap release fractions and iodine chemical form. The acceptability of the atmospheric dispersion factors, χ/Q values, is determined under SRP Section 2.3.4.

4. An ESF grade atmospheric cleanup system is required for the fuel handling building to reduce the potential radiological consequences of the fuel cask drop accident.

5. The plant design with regard to spent fuel cask drop accidents is acceptable without calculation of radiological consequences if potential cask drop distances are less than 9.2 meters (30 feet) and appropriate impact limiting devices are employed during cask movements, as determined by ECGB."

2.3 Regulatory Guide 1.70, Chapter 15

RG sections that evaluate events/criteria that are not applicable to the ESBWR are not addressed in the following subsections.

2.3.1 *Classification Terms*

Regulatory Guide (RG) 1.70, Chapter 15 states "The situations analyzed should include *anticipated operational occurrences* (e.g., a loss of electric load resulting from a line fault), *off-design transients* that induce fuel failures above those expected from *normal operational occurrences*, and *postulated accidents* of low probability (e.g., the sudden loss of integrity of a major component). The analyses should include an assessment of the consequences of an assumed fission product release that would result in potential hazards not exceeded by those from any *accident considered creditable*." However, Chapter 15 does not define or quantify the terms *anticipated operational occurrence, off-design transients, normal operational occurrence, postulated accident* and *accident considered credible*. The term/classification *off-design transients* does not appear in the 10 CFR regulations or elsewhere in Chapter 15.

For "transient" and "accident" classification, Chapter 15 classifies abnormal events as "*incidents of moderate frequency*," "*infrequent incidents*" and "*limiting faults*." The RG assigns each of these classifications following event frequencies:

"a. *Incidents of moderate frequency* - these are incidents, any one of which may occur during a calendar year for a particular plant."

"b. *Infrequent incidents* - these are incidents, any one of which may occur during the lifetime of a particular plant."

"c. *Limiting faults* - these are occurrences that are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material."

These event classifications are inconsistent with the 10 CFR regulations and the three event classifications presented earlier in Chapter 15, as shown in the first paragraph of this subsection.

By regulatory definition, an AOO is an abnormal event that could occur "one or more times during the life of the nuclear power unit." Therefore, *Incidents of moderate frequency* and *Infrequent incidents* are AOOs. *Limiting faults* can be correlated to *postulated accidents*, however, there no incident classification that directly correlates with "*off-design transients*."

Chapter 15 does not specifically specify what is a *transient* and what is an *accident*. Nor does Chapter 15 specify which of the three classifications is *transients* and which is *accidents*.

Regulatory Guide 1.70 Chapter 15 has no classification terms for beyond design basis events such as ATWS and other events that assume failures beyond the SFC.

2.3.2 *Acceptance Criteria*

Chapter 15 does not specify acceptance criteria, except that it does refer to "determining adequacy of the plant design to meet 10 CFR Part 100 criteria."

2.4 NUREG-1503, ABWR FSER Chapter 15

2.4.1 *Classification Terms*

Section 15.1 states "AOOs which include *infrequent* and *moderate frequency events* are those transients expected to occur during normal or planned modes of plant operation." This event classification interpretation is consistent with that shown in RG 1.70 (see Subsection 2.3.1, above).

Section 15.2 states that a *Pressure Regulator Down-Scale Failure* event assumes that a common-mode failure occurs. "The staff includes this postulated event in the *special category of anticipated transients* involving a common-mode software failure because of the uncertainty that such an event will occur during the plant lifetime."

Section 15.3 addresses the mislocated fuel bundle, misoriented fuel bundle, rod ejection and control rod drop as *accidents*.

Per a Section 15.3 cross-reference to Section 15.4 and the text of Section 15.4, the ABWR *design basis accidents* (DBAs) are the

- control rod drop accident,
- failure of small lines carrying primary coolant outside containment,
- main steamline failure outside containment,
- LOCA,
- fuel handling accident,
- spent fuel cask drop accident, and
- reactor water cleanup system failure outside containment.

These ABWR DBAs are directly applicable to the ESBWR.

2.4.2 *Acceptance Criteria*

Section 15.1 lists the following acceptance criteria for AOOs.

- "Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values according to American Society of Mechanical Engineers (ASME) Code, Section III, Article NB-7000."
- "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. For BWRs, the minimum value of the critical power ratio reached during the transient 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 limiting value of the minimum critical power ratio (MCPR), (is) called the safety limit."
- "An incident that occurs with moderate frequency should not generate a more serious plant condition unless other faults occur independently."
- "An incident that occurs with moderate frequency in combination with any single active component failure, or operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel-rod-cladding perforations is acceptable."

For the *Pressure Regulator Down-Scale Failure* event, Section 15.2 states "The staff required that GE demonstrate that this *special event* will not exceed the limits of 10 percent of 10 CFR Part 100, which the staff considers appropriate for an event of such postulated frequency."

Section 15.3 concludes that the consequences of a *mislocated fuel bundle accident* are acceptable, because they are less than the 10 CFR Part 100 criteria.

Because of its low probability, the ABWR Design Control Document (DCD) classified the *misoriented fuel bundle* as an accident. For the *misoriented fuel bundle accident*, no radiological evaluation was performed, because an analysis shows that fuel safety limits are not exceeded by this event.

Per Section 15.3, the *rod ejection accident* consequences are bounded by the *control rod drop accident* consequences (addressed in Section 15.4), and no radiological analysis is specifically provided.

Section 15.4 states that all DBAs result in consequences less than the exposures in 10 CFR Part 100 and GDC 19.

Subsection 15.4.1 concludes that the radiological consequences of a *control rod drop accident* are acceptable, because they are "less than a small fraction of the dose reference values specified in 10 CFR 100.11."

Subsection 15.4.2 concludes that the radiological consequences of a *failure of small lines carrying primary coolant outside containment* are acceptable, because they are "well within the dose reference values specified in 10 CFR 100.11."

Subsection 15.4.3 concludes that the radiological consequences of a *main steamline failure outside containment* are acceptable, because they are "within the acceptance criteria of SRP Section 15.6.4."

Subsections 15.4.4 and 15.4.4.4 conclude that the radiological consequences of a *LOCA* are acceptable, because they are "within the dose reference values specified in 10 CFR 100.11."

Subsection 15.4.5 concludes that the radiological consequences of a *fuel handling accident* and a *spent fuel cask drop accident* are each acceptable, because they are "less than or equal to 25 percent of the 10 CFR Part 100 dose limits."

Subsection 15.4.6 concludes that the radiological consequences of *postulated radioactive releases resulting from liquid tank failure* are each acceptable, because "any potential release associated with a liquid tank failure will not result in radionuclide concentrations in water exceeding the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, in any unrestricted area."

Subsection 15.4.7 concludes that the radiological consequences of a *reactor water cleanup system failure outside containment* are acceptable, because they are less than those for the *main steamline break outside containment* in SRP Section 15.6.4."

2.5 SECY-94-084

SECY-94-084, Section A.I (Scope and Criteria) specifically addresses ATWS and Station Blackout as "beyond design basis."

3. DETERMINATION OF EVENT CLASSIFICATIONS

Because

- the 10 CFR regulations have authority over all other document types,
- the non-accident abnormal event classifications within the SRP are inconsistently used,
- the non-accident abnormal event classifications within 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 non-accident abnormal event classifications in the SRP and RG 1.70 are not consistent 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 AOO, LOCA, 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 classification of events should primarily be based on the classifications and terms used in the 10 CFR regulations.

The *design basis events* 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 these event scenarios do assume failures beyond the single failure criterion (SFC) and/or common-mode failures. Therefore, these events should be classified as *beyond design basis events*, however, their safety analyses should be included in a DCD or FSAR.

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 is 60 years, any abnormal event that has an annual probability of occurrence ≥ 1/60 shall be classified as an AOO. RG 1.70 Chapter 15 states "The analyses should include an assessment of the consequences of an assumed fission product release that would result in potential hazards not exceeded by those from any *accident considered credible*." Based on the GE generic Product Safety Standards (Reference 7), postulated accidents should have an annual probability $> 10^{-6}$. Therefore, any event with an annual probability of $\leq 10^{-6}$ is not considered credible and is not classified as a *design basis event.*

The SRP and RG 1.70 postulate events that (for the ESBWR with its advanced design features and additional redundancy) require failures beyond the SFC, require common-mode failures, and/or have annual probabilities of $\leq 10^{-6}$. Those events shall be included in the ESBWR DCD as *beyond design basis events.*

The 10 CFR regulations postulate some low probability events with specified acceptance criteria, e.g., Safe Shutdown Fire, SBO and ATWS. For these events, the 10 CFR regulations specify design/prevention/coping/mitigation features and acceptance criteria, but do not require a radiological evaluation. Plus, SBO and ATWS require failures beyond the SFC or common-mode failures. These 10 CFR mandated events shall be included in the ESBWR DCD as *beyond design basis events.*

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 *beyond design basis events* should be developed on an event-specific basis.

As probabilistic risk assessment (PRA) insights are developed and employed, some of the current *design basis events* may be re-classified as *beyond design basis events.* For these events and possible future changes, a generic set of acceptance criteria for *beyond design basis events* should be developed.

3.1 Steps For Determining Event Classifications

(1) Per GDC 10, the 10 CFR50.49 *design basis event* definition, and guidance from events addressed in the SRP;

   a. divide the types of events as *design basis events*, and by exclusion, all other events as *beyond design basis events*;

   b. determine if AOOs should be treated as accidents; and

   c. generate the criterion for determining which type of accidents shall be classified as *design basis accidents* (*DBA*s), and by exclusion, all other accidents are not *DBA*s.

(2) Per the regulatory definition of an AOO 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 guidance for the types of events that should be addressed in Chapter 15, and (d) SRP acceptance criteria for transient/AOO events that result in fuel failure, generate a classification term and criteria for determining non-AOO and non-DBA 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 *design basis event* term in the *design basis event* classifications.

(4) Based on (a) reviewing the 10 CFR regulations that have added other abnormal events (e.g., ATWS, SBO, Safe Shutdown Fire), (b) that design basis events do not include common-mode failures and/or additional failure(s) beyond the SFC, and (c) reviewing the SRP events that include common-mode failures and/or failure(s) beyond the SFC, generate criteria for classifying events as "*beyond design basis events*."

3.2 Results of Event Classification Determinations

Table 1 provides the results of the event classifications in the form of a determination criteria vs. event classification matrix. Table 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*,
- *design basis accidents*,
- *external events*,
- *natural phenomena*, and
- *non-limiting accidents*, see (3) for additional details.

(1) b. AOOs, by definition, are classified as part of *normal operations*, do not have consequences, have more restrictive acceptance criteria (e.g., GDC 10 vs. 10 CFR 100) than accidents, and thus, are not accidents and shall not be treated as accidents.

(1) c. 10 CFR 50.49 associates the *design basis events* with ensuring the safety-related functions, of which one of the safety-related functions [i.e., the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11] is directly applied *design basis accidents* (DBAs). Based on (a) the 10 CFR regulation definitions for *design bases*, *design basis events* and *safety-related*, and (b) that the 10 CFR 50 App. B plant design criteria, and 10 CFR 100 App. A site criteria are quantified to accidents with radiological consequences which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11, <u>an accident is only classified as a *DBA*, if the accident results in a radiological consequence which could result in potential offsite exposure comparable to an applicable guideline exposure set forth in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2) or 10 CFR 100.11</u>.

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

(2) An *AOO* is any abnormal event with or without assuming additional active component failure(s) or operator error(s) that has an event probability of ≥ 1/60 per year.

(3) Because

a. the other (non-AOO and non-DBA) postulated initiating events (with or without assuming a single active component failure or single operator error) in the SRP involve challenges to a fission product barrier (e.g., fuel cladding) and can include radiological evaluations;

b. SRP 15.1 and 15.2 state "An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents...," and thus, treat AOOs involving a single failure as some undefined type of accident;

c. 10 CFR 50.2 and 10 CFR 100 associate a breach of a fission product barrier with an accident and a radiological consequence;

d. the 10 CFR regulations only address AOOs and accidents;

e. AOOs must meet GDC 10, while DBAs have offsite exposures comparable to the guidelines in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2) or 10 CFR 100.11; and

f. the lowest radiological consequence acceptance criteria for an accident in the SRP is 10% of the 10 CFR 100.11 guidelines;

the other (non-AOO and non-DBA) *design basis events* should be classified as some type of accident, but not as DBAs. In this report, these events are termed as "*non-limiting accidents*." A *non-limiting accident* is defined as "an initiating event with or without assuming a single active component failure or single operator error, that is not expected to occur during the life of a nuclear power unit but is more probable than a DBA, and results in radiological consequences less than or equal to 10% of the guideline exposures in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2) or 10 CFR 100.11."

(4) *Beyond design basis events*

a. are not listed 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: Because of the ESBWR's advanced engineering and additional redundant features, abnormal events that may have been classified as AOOs for earlier plants may have to be classified as *non-limiting accidents* or *beyond design basis events* for the ESBWR.

4. 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 shall be used. Where an acceptance criterion is not specified in the 10 CFR regulations, then the criterion in the SRP shall be used. Where an acceptance criterion is not specified in regulations and the SRP, then the criterion in the FSER for the ABWR or an NRC review LTR shall be used.

A preliminary listing of the ESBWR abnormal events and their event classifications is provided in Table 2. In making the event classification determinations, when there was a real uncertainty to which classification should be applied to an event, the lower (i.e., more conservative) classification was used. Table 2 is subject to change due to future probabilistic analyses or regulatory considerations, and thus, may be revised in the future.

4.1 Anticipated Operational Occurrences

For AOOs, the GDC 10 acceptance criterion is that "specified acceptable fuel design limits are not exceeded." To meet GDC 10, SRP 15.1 and 15.2 provides more detailed (equivalent) criteria, for AOOs with and without a single active component failure or single operator error. For AOOs, which do not involve a single active component failure or single operator error, the SRP 15.1 and 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." For AOOs, which do involve a single active component failure or single operator error, the SRP 15.1 and 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 the 10 CFR 20.1301 dose rate limit apply, and thus, the maximum dose rate resulting from the event in any unrestricted area shall not exceed 0.002 rem/hr.

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 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 Technical Specifications 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 has a nearly equivalent set of AOO acceptance criteria.

Consistent with GDC 38, if an AOO involves Safety/Relief Valve (SRV) or Depressurization Valve (DV) discharge, containment and suppression pool pressures and temperatures shall be maintained below their design values.

Based on the above, Table 3 lists the DCD Chapter 15 safety analysis acceptance criteria for AOOs that do not involve an additional single active component failure or single operator error, and Table 4 lists the DCD Chapter 15 safety analysis acceptance criteria for AOOs that do involve an additional single active component failure or single operator error. These sets of acceptance criteria assume that all related safety analyses are performed with accepted models.

4.2 Non-Limiting Accidents

The 10 CFR regulations associate the consequences of postulated accidents with the guideline exposures in 10 CFR 100.11. For most DBAs, the SRP correlates the consequence acceptance criteria to 10%, 25% and 100% of the 10 CFR 100.11 guideline exposures, depending of the severity of the different types of accidents. The *non-limiting accidents*, at the most, should not result in a larger consequence than the least severe of the DBAs, and thus, their radiological acceptance criteria should be limited to 10% of the 10 CFR 100.11 guideline exposures. However, if the SRP specifies a lower (i.e., more conservative) radiological acceptance criterion (e.g., a 10 CFR 20 limit), then the SRP acceptance criterion applies.

For AOOs, reactor system stresses should not exceed ASME Code Service Level B limits. For accidents, termed as "limiting faults" in RG 1.70, reactor system stresses should not exceed ASME Code Service Level D limits. However, based on ASME code classification of events and their associated stress limits, the *non-limiting accidents* most closely correlate with ASME Code Service Level C limits.

Consistent with GDC 38, if an AOO involves Safety/Relief Valve (SRV) or Depressurization Valve (DV) discharge, containment and suppression pool pressures and temperatures shall be maintained below their design values.

If a *non-limiting accident* results in an SRV/DV discharge or fission product release to the containment, then containment stresses (i.e., pressures and temperatures) should be limited such that there is no loss of a containment barrier safety function.

GDC 19 is the basis for the acceptance criterion on control room doses from accidents.

The set of acceptance criteria for *non-limiting accident* safety analyses are provided in Table 5.

4.3 Design Basis Accidents

The 10 CFR regulations only associate the consequences of postulated accidents with the guideline exposures in 10 CFR 100.11. However, the SRP applies 10%, 25% and 100% of the 10 CFR 100 guidelines, and 10 CFR 20 limits to different accident scenarios. Because the regulation always compare *DBA* consequences to the 10 CFR 100 guidelines, any SRP accident

scenario that has an acceptance limit that is not based on 10 CFR 100 constitutes a *non-limiting accident*, which is addressed in Section 4.2.

For any accident that involves ECCS activation, the 10 CFR 50.46(a)(3)(b) acceptance criteria apply, and thus, the calculated changes in core geometry shall be such that the core remains amenable to cooling.

RG 1.70 classifies accidents as "limiting faults," which can be correlated to different service levels or design conditions in the applicable industry code, e.g., ASME Code Service Level C or D. To ensure conservatism and minimize the number of acceptance condition options, for DBAs, nuclear system, containment structural and isolation components, and suppression pool stresses (i.e., temperatures and pressures) shall be limited to ASME Code Service Level C limits.

GDC 19 is the basis for the acceptance criterion on control room doses from accidents.

The set of acceptance criteria for *DBA* safety analyses are provided in Table 6.

Because radiological acceptance criteria vary for the different accident scenarios, for each accident scenario (applicable to an ESBWR), Table 7 provides the accident classification (*non-limiting accident* or *DBA*) and the radiological acceptance criterion.

4.4 Beyond Design Basis Events

As discussed in Section 3, the acceptance criteria for each of the *beyond design basis event* safety analysis is developed on an event-specific basis, based on the coping, mitigation or acceptance criteria specified in the 10 CFR regulations, the SRP or an NRC reviewed LTR.

4.4.1 *Overpressure Protection Analysis*

For every fuel cycle an Overpressure Protection Analysis is performed. With respect to the reactor coolant pressure boundary (RCPB) pressure response, the event scenario is specifically chosen to bound all of the *design basis events*. The event requires/assumes

(1) an operator error, multiple equipment failures or a common mode failure cause(s) the MSIVs in all four main steamlines (MSLs) to simultaneously close;

(2) the two MSIV position switch circuits on three to six MSIVs fail, which causes the MSIV position scram function to fail; and

(3) the reactor is shutdown by a high neutron flux scram trip.

The Overpressure Protection Analysis demonstrates that the SRVs have adequate pressure relief capacity to prevent the RCPB ASME Code Service Level B pressure limit(s) and the Reactor Coolant System Pressure Safety Limit in the Technical Specifications from being exceeded. Therefore, this event only needs/has the following acceptance criteria.

- 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).
- The reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications.

4.4.2 *Shutdown Without Control Rods*

Assuming all control rod insertion mechanisms fail, for every fuel cycle, cold shutdown core k-effective (k_{eff}) calculations are performed at various cycle exposure points, to ensure that the Standby Liquid Control System (SLCS) can inject adequate (boron solution) negative reactivity into the core to allow for cold shutdown. This analysis plus the normal control rod shutdown margin calculations demonstrate compliance to GDC 26. The Shutdown Without Control Rods event only needs/has the following acceptance criterion.

- Under the most reactive core conditions, k_{eff} shall be 1.0.

4.4.3 *Anticipated Transient Without Scram*

Based on Reference 6, the generic BWR ATWS performance analysis acceptance criteria are summarized below.

- Peak vessel bottom pressure less than ASME Service Level C limit, which GE has conservatively interpreted to correspond to 120% of design pressure.
- Peak cladding temperature within the 10 CFR 50.46 limit of 2200°F.
- Peak cladding oxidation within the requirements of 10 CFR 50.46.
- Peak suppression pool temperature shall not exceed its design temperature.
- Peak containment pressure shall not exceed containment design pressure.
- Radiological dose consequences less than the 10 CFR 100.11 guideline exposures.

4.4.4 *Safe Shutdown Fire*

The following acceptance criteria are derived from 10 CFR Part 50.48 and Appendix R.

- Core subcriticality is achieved and maintained with adequate core shutdown margin, as specified in the plant Technical Specifications.
- Adequate reactor coolant inventory is maintained such that reactor water level is maintained above the core (i.e., top of active fuel).
- Hot shutdown conditions are achieved and maintained.
- Cold shutdown conditions are achieved within 72 hours.
- Cold shutdown conditions are maintained thereafter.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design limits, e.g., not exceed ASME Code Service Level C limits.

Safety-related or nonsafety-related equipment may be used to meet the above criteria.

4.4.5 *Station Blackout*

An SBO safety analysis shall demonstrate that the plant can cope with the effects (i.e., with minimum equipment available) of an SBO for the duration of the SBO. The ability to cope with an SBO is based on meeting the following acceptance criteria.

- There is no core uncovery.

- Achieve and maintain the plant to those shutdown conditions specified in plant Technical Specifications as Hot Shutdown.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design limits, e.g., not exceed ASME Code Service Level C limits.

4.4.6 *Shutdown from Outside Main Control Room*

A Shutdown from Outside Main Control Room safety analysis shall demonstrate that the plant can achieve and maintain safe shutdown, assuming the reactor is scrammed by the operators before they vacate the main control room. The ability to cope with a Shutdown from Outside Main Control Room event is based on meeting the following acceptance criteria.

- There is no core uncovery.
- Achieve and maintain the plant to those shutdown conditions specified in plant Technical Specifications as Hot Shutdown.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design limits, e.g., not exceed ASME Code Service Level C limits.

4.4.7 *Possible Future Beyond Design Basis Events*

The 10 CFR regulations and the SRP do not contain a generic set of safety analysis acceptance criteria for *beyond design basis events*, however, ATWS analysis acceptance criteria (in Subsection 4.4.3) cover the major safety concerns that could be related to any reactor event. Therefore, the ATWS analysis acceptance criteria are chosen as the starting point for the generic set of safety analysis acceptance criteria for *beyond design basis events*.

All potential future events are not expected to require containment isolation or involve radiological consequences that could challenge the 10 CFR 100.11 guidelines, and thus, the final generic set of *beyond design basis event* safety analysis acceptance criteria (contained in Table 8) are slightly more conservative and have more flexibility than the ATWS analysis acceptance criteria. However, in the future, if some of the current *design basis events* are re-classified as *beyond design basis events* or new *beyond design basis events* are added to the ESBWR licensing basis, then event-specific criteria may have to be developed.

5. REFERENCES

1. USNRC, "Code of Federal Regulations," Title 10, latest revisions through October 2004.
2. USNRC, "Standard Review Plan," NUREG-0800, latest revisions through October 2004.
3. USNRC, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants LWR Edition," Regulatory Guide 1.70, Revision 3, November 1978.
4. USNRC, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," NUREG-1503, Volume 1, July 1994.

5. USNRC, "SECY-94-084 - Policy and Technical Issues Associated With The Regulatory Treatment of Non-Safety Systems and COMSECY-94-024 - Implementation of Design Certification and Light-Water Reactor Design Issues," June 30, 1994.

6. General Electric Co., "Assessment of BWR Mitigation of ATWS, Volume II," NEDE-24222, Class III (Proprietary), December 1979.

7. General Electric Co., "Product Safety Standards," 22A8400, Revision 2, January 1986.

Table 1

Chapter 15 Abnormal Event Classification Determination Matrix

Determination Criteria vs. Event Classification	Annual Probability ≥ 1/60	Annual Probability < 1/60	Annual Probability > DBA	Annual Probability < DBA	Radiological Analysis Basis		Assumes A Single Active Component Failure or Single Operator Error		Event Not Listed As A Design Basis Event in 10 CFR 50.49(b)(1)(ii) and		
					10 CFR Part 20	10 CFR Part 100 & GDC 19	Yes	No	Is Postulated In A Regulation	Assumes Common-Mode Failure(s)	Assumes Failures, Beyond SFC
AOO	X						X*	X*			
Non-limiting Accident		X	X		X*	X*	X*	X*			
DBA		X				X	X				
Beyond Design Basis Event				X		X**			X***	X***	X***

* Specific event dependent.

** If applicable to a specific event.

*** Or any combination of these conditions.

Table 2

Preliminary List of ESBWR Abnormal Event Classifications

Abnormal Event	AOO	Non-limiting Accident	DBA	Beyond Design Basis Event
Loss of Feedwater Heating	X			
Inadvertent RWCU/SDC System Cooling Operation	X			
Closure of One Turbine Control Valve	X			
Generator Load Rejection with Bypass	X			
Generator Load Rejection with a Single Failure in the Turbine Bypass System	X			
Turbine Trip with Bypass	X			
Turbine Trip with a Single Failure in the Turbine Bypass System	X			
Closure of One Main Steam Isolation Valve	X			
Closure of All Main Steam Isolation Valves	X			
Loss of Condenser Vacuum	X			
Loss of Shutdown Cooling Function of RWCU/SDC System	X			
Control Rod Withdrawal Error During Refueling	X			
Control Rod Withdrawal Error During Startup	X			
Control Rod Withdrawal Error During Power Operation	X			
Inadvertent Isolation Condenser Initiation	X			
Runout of One Feedwater Pump	X			
Opening of One Turbine Bypass Valve	X			
Inadvertent Opening of a Depressurization Valve	X			
Inadvertent Opening of a Safety/Relief Valve	X			
Loss of Unit Auxiliary Transformer	X			
Loss of Grid Connection	X			
Loss of All Feedwater Flow	X			
Feedwater Controller Failure – Maximum Demand		X		
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves		X		

Table 2

Preliminary List of ESBWR Abnormal Event Classifications

Abnormal Event	AOO	Non-limiting Accident	DBA	Beyond Design Basis Event
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves		X		
Generator Load Rejection with Total Turbine Bypass Failure		X		
Turbine Trip with Total Turbine Bypass Failure		X		
Liquid-Containing Tank Failure		X		
Fuel Assembly Loading Error, Mislocated Bundle		X		
Fuel Assembly Loading Error, Misoriented Bundle		X		
Control Rod Drop Accident			X	
Failure of Small Line Carrying Primary Coolant Outside Containment			X	
Main Steamline Break Outside Containment			X	
Feedwater Line Break Outside Containment			X	
RWCU/SDC System Line Failure Outside Containment			X	
LOCA Inside Containment			X	
Fuel Handling Accident			X	
Spent Fuel Cask Drop Accident			X	
Overpressure Protection				X
Shutdown Without Control Rods (i.e., SLCS shutdown capability)				X
Safe Shutdown Fire				X
Anticipated Transients Without Scram				X
Station Blackout				X
Shutdown from Outside Main Control Room				X

Table 3

Safety Analysis Acceptance Criteria for AOOs That Do Not Involve An Additional Single Active Component Failure or Single Operator Error

*

- 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 limiting value of the minimum critical power ratio (MCPR) is the Safety Limit MCPR 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).
- If containment isolation is involved, 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 in loss of function of any fission product barrier, except for Safety/Relief Valve or Depressurization Valve discharge.

* Based on SRP Sections 15.4.1 and 15.4.2 the following acceptance criterion shall be included 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.

Table 4

Safety Analysis Acceptance Criteria for AOOs That Involve An Additional Single Active Component Failure or Single Operator Error

- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (e.g., 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.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design values.
- Except for fuel cladding, there shall be no loss of function of any fission product barrier.
- Fuel cladding failures shall be limited such that the maximum radiation dose rate in any unrestricted area shall not exceed 0.002 rem/hr.

Table 5

Safety Analysis Acceptance Criteria for Non-Limiting Accidents

- Peak vessel bottom pressure less than ASME Service Level C limit, which GE has conservatively interpreted to correspond to 120% of design pressure.
- Radiological consequences shall equal to or less than 10% of the 10 CFR 100 guideline exposures. However, if the applicable SRP section specifies a lower (i.e., more conservative) radiological acceptance criterion (e.g., a 10 CFR 20 limit), then the SRP acceptance criterion is applied.
- If containment isolation is required, containment and suppression pool pressures and temperatures shall be maintained below their design limits, e.g., not exceed ASME Code Service Level C limits.
- Control room personnel shall not receive a radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Table 6

Safety Analysis Acceptance Criteria for Design Basis Accidents

- Peak vessel bottom pressure less than ASME Service Level C limit, which GE has conservatively interpreted to correspond to 120% of design pressure.
- Radiological consequences shall equal to or less than 10%, 25% or 100% of the 10 CFR 100.11 guideline exposures, depending on the accident scenario-specific acceptance criterion in NUREG-0800.
- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.
- If containment isolation is required, containment and suppression pool pressures and temperatures shall be maintained below their design limits, e.g., not exceed ASME Code Service Level C limits.
- Control room personnel shall not receive a radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Table 7

ESBWR Accident Classifications and Radiological Acceptance Criteria

Accident*	Accident Class**		Radiological Acceptance Criteria***					
	Non-Limiting Accident	Design Basis Accident	10 CFR 20, App. B, Table 2, Column 2	10 CFR 20.1301(a)(2) ****	GDC 19	10% of 10 CFR 100.11	25% of 10 CFR 100.11	100% of 10 CFR 100.11
Feedwater Controller Failure – Maximum Demand	X				X	X		
Pressure Regulator Failure - Opening of All Turbine Control and Bypass Valves	X				X	X		
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	X				X	X		
Generator Load Rejection with Total Turbine Bypass Failure	X				X	X		
Turbine Trip with Total Turbine Bypass Failure	X				X	X		
Liquid-Containing Tank Failure	X		X		X			
Fuel Assembly Loading Errors (mislocated and misoriented)	X				X	X		
Control Rod Drop Accident (radiological analysis)		X			X		X	
Failure of Small Line Carrying Primary Coolant Outside Containment		X			X	X		
Main Steamline Break Outside Containment		X			X	X		
Feedwater Line Break Outside Containment		X			X	X		

Table 7

ESBWR Accident Classifications and Radiological Acceptance Criteria

Accident*	Accident Class**		Radiological Acceptance Criteria***					
	Non-Limiting Accident	Design Basis Accident	10 CFR 20, App. B, Table 2, Column 2	10 CFR 20.1301(a)(2) ****	GDC 19	10% of 10 CFR 100.11	25% of 10 CFR 100.11	100% of 10 CFR 100.11
LOCA Inside Containment Radiological Analysis, (including all leakage paths)		X			X			X
Fuel Handling Accident		X			X		X	
Spent Fuel Cask Drop Accident		X			X		X	
Reactor Water Cleanup / Shutdown Cooling System Failure Outside Containment		X			X	X		

* Based on SRP 15 and ABWR FSER (Reference 4) events involving a radiological consequence.

** From Table 2, "Preliminary List of ESBWR Abnormal Event Classifications."

*** Based on the 10 CFR regulations and SRP 15.

**** Usually only applicable to AOOs.

Table 8

Generic Safety Analysis Acceptance Criteria for Beyond Design Basis Events

- Peak vessel bottom pressure less than ASME Service Level C limit, which GE has conservatively interpreted to correspond to 120% of design pressure.
- Peak cladding temperature within the 10 CFR 50.46 limit of 2200°F.
- Peak cladding oxidation within the requirements of 10 CFR 50.46.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design limits, e.g., not exceed ASME Code Service Level C limits.
- Radiological dose consequences less than 10%* of the 10 CFR 100.11 guideline exposures.
- Control room personnel shall not receive a radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the event.

* 25% or 100% should be used for any *beyond design basis event* that involves a breach of the reactor coolant pressure boundary.

Note: This table is applicable to possible future *beyond design basis events* that do not have event-specific acceptance criteria, as shown within Section 4.4.