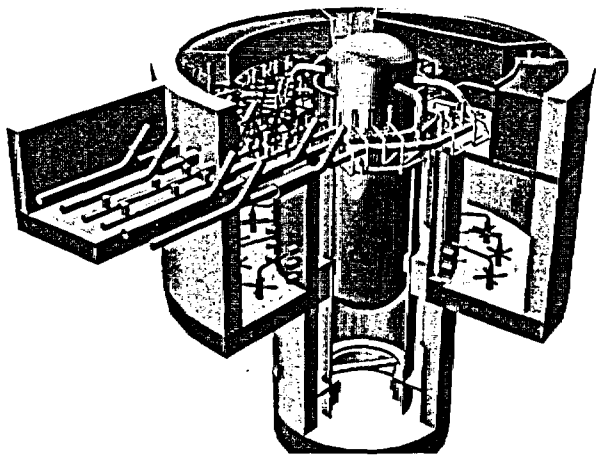

ESBWR Abnormal Event Classifications

Kurt Schaefer – GE

ESBWR Event Classification Process Meeting

January 19, 2005

imagination at work

Event Classification & Acceptance Criteria Process

Agenda

- Process
- Historical Review
- Event Classification Terms
  - 10 CFR Regulations
  - Chapter 15 of the SRP & RG 1.70
  - Final ESBWR Abnormal Event Classifications
- Safety Analyses Acceptance Criteria
  - Goal & Sources
  - 10 CFRs
  - SRP Chapter 15
  - ABWR FSER & NEDE/NEDO-24222
- Summary, Questions, etc.

Event Classification & Acceptance Criteria Process

The licensing basis for the ESBWR is in its initial development.

As part of the initial licensing basis development, the regulatory criteria for event classification were reviewed to determine the appropriate abnormal event classifications and their associated safety analysis acceptance criteria.

Regulatory hierarchy used (based on priority):

1. 10 CFR regulations
2. USNRC Standard Review Plan (SRP) Section 15
3. RG 1.70 Chapter 15
4. NUREG-1503, FSER for the ABWR Design, Chapter 15
5. SECY-94-084

Historical Review - Existing Plants

Abnormal Event Classification Terms

Reg. Guide 1.48:

Upset Plant Condition	Those deviations from the normal plant condition which have a high probability of occurrence.
Emergency Plant Condition	Those operating conditions which have a low probability of occurrence.
Faulted Plant Condition	Those operating conditions associated with extremely-low-probability postulated events.

Historical Review - Existing Plants

Abnormal Event Classification Terms (cont'd)

Late BWR/4s	BWR/6	GESSAR II	ABWR
Anticipated Operational Transients	Anticipated (Expected) Operational Transients	Anticipated (Expected) Operational Transients	Moderate Frequency Incidents (Anticipated (Expected) Operational Transients)
Abnormal Operational Transients	Abnormal (Unexpected) Operational Transients	Abnormal (Unexpected) Operational Transients	Infrequent Incidents (Abnormal (Unexpected) Operational Transients)
Design Basis Accidents	Design Basis (Postulated) Accidents	Design Basis (Postulated) Accidents	Limiting Faults (Design Basis (Postulated) Accidents)
Special Events	Special (Plant Capability) Events	Special (Hypothetical) Events	Special (Hypothetical) Events

Event Classification Terms

10 CFR Regulations

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:

(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."

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 ... (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a ***postulated accident*** ...."

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.

Event Classification Terms

10 CFR Regulations (cont'd)

App. A Definition: "***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***.

App. A, 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***.

Event Classification Terms

10 CFR Regulations (cont'd)

GDC 17, 20, 22, 26, 27, 29, 31, 41, 55, 60 and 64 each address ***anticipated operational occurrences*** and/or ***postulated accidents***.

GDC 28 addresses ***postulated reactivity accidents***.

10 CFR 50 App. B states: 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**.

What is the only accident consequence that can cause undue risk to the health and safety of the public?

A Radiological Dose

Event Classification Terms

10 CFR Regulations (cont'd)

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."

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

10 CFR 100.10(a)(4) 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***.

Event Classification Terms

10 CFR Regulations (cont'd)

For the Station Blackout (SBO) event, 10 CFR 50.2 specifies that an SBO is a "***non-design basis accident***."

The ***design basis events*** in the 10 CFRs assume an initiating event (and any resultant failures) with or without a single active component failure or operator error.

- The 10 CFR regulations have no classification term for events such as ATWS, SBO and other events that assume common mode failures or failures beyond the single failure criterion.

Event Classification Terms

Chapters 15 of the SRP & RG 1.70

SRP and RG 1.70 use **18** different classification type terms for 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

Except for AOOs, **none of those classification terms is defined or quantified in either the 10 CFR regulations or SRP 15.**

Event Classification Terms

Chapter 15 of the SRP

Draft Rev. 2 of SRP 15.1.1 - 15.1.4, Section II, Acceptance Criteria;

Draft Rev. 2 of SRP 15.2.1 - 15.2.5, Section II, Acceptance Criteria; and

Draft Rev. 2 of SRP 15.2.6, Section II, Acceptance Criteria all state:

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...

Event Classification Terms

The SRP does not have a consistent and concise classification term to identify design basis events that are less probable than an AOO but more probable than a design basis accident (DBA), and results in consequences that are not specified to be directly comparable to 10 CFR 100.

DBAs, by definition, have consequences comparable to 10 CFR 100.

- Therefore, the ESBWR term should be based on terms used in the 10 CFR's.

However, the event acceptance criteria from the SRP and ABWR FSER are based on **limited fuel failures** (i.e., MCPR Safety Limit may be exceeded) but **with acceptably low radiological consequence** (usually < 10% of 10 CFR 100).

- Therefore, this type of design basis event must be classified as some type of accident, but is not a DBA.

Event Classification Terms

Did not want to use any of the terms used in past BWR FSARs, because none of those is consistent with the 10 CFRs.

Could not ***just*** call them accidents, because that does not clearly separate them from DBAs.

Could not call them non-DBAs, because they may be confused with the beyond design basis events.

Therefore, needed a classification that used the term "accident" (to be consistent with the 10 CFR's), but is clearly separate from the DBA classification.

The result: **"non-limiting accidents"**

Event Classification Terms

The 10 CFR's require evaluations of three (non-design basis) specific event scenarios, i.e., Safe Shutdown Fire, SBO and ATWS, and often these event scenarios assume failures beyond the single failure criterion (SFC) and/or common-mode failures.

The 10 CFR's, SRP Section 15 and RG 1.70 Chapter 15 have no classification term for events that assume common mode failures or 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***."

Therefore, the phrase "***beyond design basis***" is appropriate for any abnormal event evaluated in Tier 2 that is not classified as a design basis event.

Event Classification Terms

Final ESBWR Abnormal Event Classifications:

1. **Anticipated Operational Occurrences**
   (DCD Tier 2 Section 15.2)
2. **Non-Limiting Accidents**
   (DCD Tier 2 Section 15.3)
3. **Design Basis Accidents**
   (DCD Tier 2 Section 15.4)
4. **Beyond Design Basis Events**
   (DCD Tier 2 Section 15.5)

Event Classification Terms

Proposed ESBWR Definitions

An ***anticipated operational occurrence*** (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.

Event Classification Terms

Proposed ESBWR Definitions (cont'd)

A ***non-limiting accident*** is 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.

Event Classification Terms

Proposed ESBWR Definitions (cont'd)

A ***design basis accident*** (DBA) is an accident that 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.

Event Classification Terms

Proposed ESBWR Definitions (cont'd)

Beyond design basis event is not listed as a design basis event in 10 CFR 50.49, and

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

ii. includes a common mode equipment failure or additional failure(s) beyond the single failure criterion.

Note: Does not include severe accidents and other events that are only evaluated as part of the plant PRA.

Handout 1. 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	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.

** Does not include severe accidents and other events that are only evaluated as part of the plant PRA.

*** Or any combination of these conditions.

Event Classification Terms

Chapter 15 Abnormal Event Classification Determination Matrix

(Handout 1)

Safety Analyses Acceptance Criteria

Goal:

Develop a definitive specific set of *measurable* acceptance criteria, for each abnormal event classification or when a specific event requires.

Sources:

10 CFRs, SRPs, ABWR FSER and NEDE/NEDO-24222 in that order.

Most useful: The SRPs

Safety Analyses Acceptance Criteria

10 CFRs:

GDC 10 - Reactor design

GDC 15 - Reactor coolant system design

GCD 17 - Electric power systems

GDC 19 - Control room

GDC 38 - Containment heat removal

GDC 60 – Control of releases of radioactive materials to the environment

Safety Analyses Acceptance Criteria

10 CFRs:

GDC 60 – Control of releases of radioactive materials to the environment states:

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including ***anticipated operational occurrences.***

Therefore, 10 CFR 20 applies.

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."

Safety Analyses Acceptance Criteria

10 CFRs: (cont'd)

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)."

Safety Analyses Acceptance Criteria

10 CFRs: (cont'd)

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

- Peak cladding temperature
- Maximum cladding oxidation
- Maximum hydrogen generation
- Coolable geometry
- Long-term cooling

Safety Analyses Acceptance Criteria

10 CFRs: (cont'd)

10 CFR 50.62 does not provide ATWS analysis acceptance criteria.

However, BWR analysis acceptance criteria are documented in NEDE/NEDO-24222, which is the report that the NRC used in developing 50.62.

Safety Analyses Acceptance Criteria

10 CFRs: (cont'd)

For Accident (radiological) acceptance criteria;

- 10 CFR 50.67(b)(2)
- 10 CFR 100.11(a)(1)
- 10 CFR 100.11(a)(2)

Safety Analyses Acceptance Criteria

SRP Chapter 15

Draft Rev. 2 of SRP 15.1.1 - 15.1.4, Draft Rev. 2 of SRP 15.2.1 - 15.2.5, and Draft Rev. 2 of SRP 15.2.6 provide consistent sets (regulatory perspective) acceptance criteria.

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

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).

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

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**.

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.

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.

Safety Analyses Acceptance Criteria

SRP Chapter 15

Draft Rev. 2 of SRP 15.1.1 - 15.1.4, Draft Rev. 2 of SRP 15.2.1 - 15.2.5, and Draft Rev. 2 of SRP 15.2.6 provide consistent sets (regulatory perspective) acceptance criteria.

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

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).

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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.

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.

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.

Safety Analyses Acceptance Criteria

SRP Chapter 15 (cont'd)

Draft Rev. 3 of SRP 15.4.1 & 15.4.2 for Uncontrolled Control Rod Assembly Withdrawals (i.e., Rod Withdrawal Errors for a BWR).

Draft Rev. 2 of SRP 15.4.7 for an Inadvertent Loading and Operation Of A Fuel Assembly In An Improper Position (i.e., Fuel Loading Error for a BWR).

Draft Rev. 3 of SRP 15.4.9 for a Spectrum of Rod Drop Accidents (non-radiological analysis).

Draft Rev. 3 of SRP 15.4.9, Appendix A for Radiological Consequences of Control Rod Drop Accident.

Safety Analyses Acceptance Criteria

SRP Chapter 15 (cont'd)

Draft Rev. 2 of SRP 15.6.1 for Inadvertent Opening of a Safety Relief Valve (SRV).

Draft Rev. 3 of SRP 15.6.2 for Radiological Consequences of The Failure of Small Lines Carrying Primary Coolant Outside Containment.

Draft Rev. 3 of SRP 15.6.4 for Radiological Consequences of Main Steam Line Failure Outside Containment.

Draft Rev. 3 of SRP 15.6.5 (+ Appendices) for Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary (i.e., Break inside containment).

Safety Analyses Acceptance Criteria

SRP Chapter 15 (cont'd)

Draft Rev. 2 of SRP 15.7.4, Section II, Acceptance Criteria, for Radiological Consequences of Fuel Handling Accidents.

Draft Rev. 3 of SRP 15.7.5, Section II, Acceptance Criteria, for Spent Fuel Cask Drop Accidents.

Safety Analyses Acceptance Criteria

ABWR FSER

Reviewed the equivalent locations as compared to the SRP.

Section 15.1 acceptance criteria are:

- "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**."

Safety Analyses Acceptance Criteria

ABWR FSER

Reviewed the equivalent locations as compared to the SRP.

Section 15.1 acceptance criteria are:

- "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."

Safety Analyses Acceptance Criteria

ABWR FSER (cont'd)

Section 15.1 acceptance criteria *(cont'd)*

Consistent with the SRP, the last criterion addresses an AOO type of event in combination with an additional failure, that would result in some limited radiological release, which, because AOOs are defined as part of normal operation, must meet 10 CFR 20.

Therefore, the SRP and ABWR FSER effectively provide two sets of acceptance criteria for AOOs.

- AOOs
- AOOs in combination with a single active component failure or operator error.

Safety Analyses Acceptance Criteria

ABWR FSER (cont'd)

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**.

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.

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."

Safety Analyses Acceptance Criteria

ABWR FSER (cont'd)

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."

Safety Analyses Acceptance Criteria

ABWR FSER (cont'd)

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."

Safety Analyses Acceptance Criteria

ATWS

NEDE/NEDO-24222"Assessment of BWR Mitigation of ATWS, Volume II"

Safety Analyses Acceptance Criteria - Results

AOOs

- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (i.e., not exceed ASME Code Service Level B), **and** the reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications.
- Fuel-cladding integrity should be maintained by ensuring that the reactor core is designed with appropriate margin during any conditions of normal operation, including the effects of AOOs. The minimum value of the critical power ratio (CPR) reached during the AOO should be such that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. This limiting value of the minimum critical power ratio (MCPR) is the Safety Limit MCPR in the Technical Specifications.
- Uniform cladding strain ≤ 1%.*

Safety Analyses Acceptance Criteria - Results

AOOs (cont'd)

Note:

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

Safety Analyses Acceptance Criteria - Results

AOOs (cont'd)

- 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 loss of function of any fission product barrier.

Safety Analyses Acceptance Criteria - Results

AOOs In Combination With An Additional Failure / 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.
- 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.

Safety Analyses Acceptance Criteria

Non-limiting Accidents

- Peak vessel bottom pressure less than ASME Service Level C limit, which is conservatively interpreted to correspond to 120% of design pressure.
- Radiological consequences shall be 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, i.e., 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.

Safety Analyses Acceptance Criteria

DBAs

- Peak vessel bottom pressure less than ASME Service Level C limit, which is conservatively interpreted to correspond to 120% of design pressure.
- Radiological consequences shall be 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.
- If containment isolation is required, containment and suppression pool pressures and temperatures shall be maintained below their design limits, i.e., 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.

Safety Analyses Acceptance Criteria

DBAs (cont'd)

- 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.

Safety Analyses Acceptance Criteria

ESBWR Accident Classifications and Radiological Acceptance Criteria

(Handout 2)

Safety Analyses Acceptance Criteria

Beyond Design Basis Events

(Normally provided on an event-specific basis.)

Overpressure Protection Analysis:

- 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 Tech Specs.

Shutdown Without Control Rods :

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

Safety Analyses Acceptance Criteria

Beyond Design Basis Events (cond't)

Anticipated Transient Without Scram:

- Peak vessel bottom pressure less than ASME Service Level C limit, which is 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.

Handout 2. 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
Liquid-Containing Tank Failure	X		X		X			
All other non-limiting accidents	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		
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).

** Based on the 10 CFR's and SRP 15.

*** Only applicable to AOOs.

Safety Analyses Acceptance Criteria

Beyond Design Basis Events (cond't)

Safe Shutdown Fire:

- Core subcriticality is achieved and maintained with adequate core shutdown margin, as specified in the plant Tech Specs.
- 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, i.e., not exceed ASME Code Service Level C limits.

Safety Analyses Acceptance Criteria

Beyond Design Basis Events (cond't)

Station Blackout:

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

Safety Analyses Acceptance Criteria

Beyond Design Basis Events (cond't)

Shutdown from Outside Main Control Room:

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

Safety Analyses Acceptance Criteria

Beyond Design Basis Events (cond't)

Generic for potential future events:

- Peak vessel bottom pressure less than ASME Service Level C limit, which is 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, i.e., 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.

Safety Analyses Acceptance Criteria

Beyond Design Basis Events (cond't)

Generic for potential future events: (cond't)

Note:

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

Summary

- GE has generated abnormal event classifications, based on the hierarchical order of NRC documents.
- GE has documented safety analysis acceptance criteria, based on NRC generated/accepted documents.
- GE proposes to use these event classifications and safety analyses acceptance criteria for the ESBWR.