GE Energy

Robert E. Gamble
Manager, ESBWR

1989 Little Orchard St., M/C 365
San Jose, CA 95125-1030
USA

T 408 925 3352
F 408 925 5665
Robert.gamble@ge.com

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Project 717

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U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **GE Responses to NRC Request for Additional Information Regarding NEDO-33175, Revision 1, "Classification of ESBWR Abnormal Events and Determination of their Safety Analysis Acceptance Criteria"**

Enclosure 1 contains GE responses to the subject NRC Request for Additional Information (RAI), transmitted via the Reference letter. The RAI responses reflect the discussions held with the NRC staff on April 14 and 23, 2005, via telecon.

If you have any questions about the information provided here, or if we can provide any additional information to assist in your review of NEDO-33175, Revision 1, please contact Kathy Sedney at 408-925-5232 or Kurt Schaefer at 408-925-2426.

Sincerely,

Robert E Gamble

Robert E. Gamble
Manager, ESBWR

D068

Reference:

MFN 05-056, Letter from Amy E. Cubbage to Robert E. Gamble, *Request for Additional Information Regarding NEDO-33175, Revision 1, "Classification of ESBWR Abnormal Events and Determination of their Safety Analysis Acceptance Criteria,"* June 2, 2005

Enclosure:

1. MFN 05-059 – GE Responses to NRC Request for Additional Information Regarding NEDO-33175, Revision 1, "Classification of ESBWR Abnormal Events and Determination of their Safety Analysis Acceptance Criteria"

cc:	WD Beckner	USNRC
	LA Dudes	USNRC
	AE Cubbage	USNRC
	MB Fields	USNRC
	GB Stramback	GE
	eDRF 0000-0042-4488	

ENCLOSURE 1

MFN 05-059

GE Responses to NRC Request for Additional Information Regarding NEDO-33175, Revision 1, "Classification of ESBWR Abnormal Events and Determination of their Safety Analysis Acceptance Criteria"

Responses to Requests for Additional Information (RAI)
"Classification of ESBWR Abnormal Events and Determination of Safety Analysis Acceptance Criteria" NEDO-33175, Revision 1
ESBWR Pre-Application Review
General Electric Company

Note: The referenced pages from AP1000 DCD Tier 2, AP1000 FSER page, ABWR DCD Tier 2 and ABWR FSER NUREG-1503 follow the RAIs and their GE responses.

1. In Tables 4 and 9, you listed 18 abnormal events under a new "accident" category. When you submit an application for design certification of the ESBWR design, you will be required to provide the U.S. Nuclear Regulatory Commission (NRC) with the radiological consequence analysis for each of these events to ensure they meet the radiological dose acceptance criteria. Each analysis should include, but not be limited to, the source term, release point, release pathway to the environment, and the associated atmospheric dispersion factor (P/Q values) for the control room air intake.

GE Response:

The actual level of detail in the DCD Tier 2 safety analyses is outside the scope of NEDO-33175. However, GE plans to include a radiological evaluation in DCD Tier 2 for each of the NEDO-33175, Revision 1 "accidents." If a specific event results in a unique radioactive material release, then its evaluation will include all the information listed in the RAI. However, if a specific event does not result in a fission produce barrier failure (i.e., no radioactive material release), or it is bounded by a similar event, then its evaluation will explain why there is no release or how it is bounded by the similar event's analysis results. Therefore, all the information listed in the RAI will not always need to be included.

(No change to the NEDO-33175 is needed.)

2. The radiological consequence acceptance criteria for the waste gas system leak or failure should be 0.1 rem total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) and low-population zone (LPZ), and 5 rem TEDE in the control room. This is consistent with the guidance contained under Item B.1.(a), of Branch Technical Position (BTP), Effluent Treatment Systems Branch (ETSB) 11-5 in NUREG-0800, "Standard Review Plan (SRP)" Section 11.3 (July 1981). In the BTP, it states:

   "... the resulting total body exposure to an individual at the nearest exclusion boundary will not exceed 0.5 rem. This is consistent with the guidelines of 10 CFR Part 20 and is substantially below the guidelines of 10 CFR Part 100."

   The criterion in BTP 11-5 was written prior to the present Part 20 being expressed in terms of TEDE dose. Rather, the criterion was based upon the Part 20 doses which were in effect at the time (the total body dose). Application of the present Part 20 dose criteria (TEDE) to BTP ETSB 11-5 equates to a dose criterion of 0.1 rem TEDE.

GE Response:

Because the ESBWR Offgas System pressure boundary is designed to withstand dynamic overpressure from potential hydrogen detonation of at least 17 times the normal system operating pressure, a structural failure in the Offgas System is not a credible event. For the ESBWR, the only plausible event scenario that could result in a waste gas release requires two independent operator errors and an instrumentation isolation trip or (mechanical) isolation function failure to occur, and would result in only the release of noble gases. The postulation of a Waste Gas System Failure for the ESBWR goes beyond the 10 CFR 50 Appendix A single failure criterion, and thus, it does not qualify as a design basis event. Therefore, at most, the Waste Gas System Failure for the ESBWR should be treated similar to a less severe *design basis accident*, and treating this event as an *accident* (*Infrequent Incident*, per the response to RAI 4) would be conservative.

The above conclusion is consistent with Draft Rev. 3 - April 1996 version of ETSB 11-5, which applies "a small fraction of 10 CFR 100 limit" as the dose acceptance criterion. The ABWR FSER NUREG-1503 Subsection 11.3.2 (page 11-11) gives the acceptance criterion for an "offgas system leak or failure as assumed in BTP ETSB 11-5, Revision 0, July 1981" as a whole body dose "less than 10 percent of the 10 CFR Part 100 limits." ABWR is consistent with the limits in ETSB 11-5 Draft Rev. 3 - April 1996, NEDO-33175, and the generic *Infrequent Incident* radiological acceptance criterion of 2.5 rem TEDE.

10 CFR 20.1301(a)(1) states "The total effective does equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 millisievert) in a year." This limit clearly only applies to annual releases from normal operation, and the ESBWR event scenario cannot be considered part of normal operations. Therefore, the 10 CFR 20.1301(a)(1) 0.1 rem limit should not be applied.

The RAI position is that "The criterion in BTP 11-5 was written prior to the present Part 20 being expressed in terms of TEDE dose. Rather, the criterion was based upon the Part 20 doses which were in effect at the time (the total body dose). Application of the present Part 20 dose criteria (TEDE) to BTP ETSB 11-5 equates to a dose criterion of 0.1 rem TEDE." The Part 20 change to 0.1 rem TEDE was implemented prior to 1992, which is ten years prior to the 2002 AP1000 certification submittal, and thus, if the suggested 0.1 rem TEDE criterion is a NRC approved and published acceptance criterion for the Waste Gas System Failure event, then it would have been applied to the AP1000. However, page 11-18 of the NRC FSER for the AP1000 states

> "In its response to RAI 460.005(A), the applicant provided a waste gas system leak or failure analysis, and the justification for the assumptions used in that analysis. The analysis was performed to demonstrate that the WGS design meets the applicable guidelines of BTP ETSB 1-5. The BTP stipulates that the total body dose at the exclusion area boundary (EAB), as a result of the release of radioactivity for two hours from a postulated failure of the WGS, calculated in accordance with BTP assumptions, should not exceed 0.5 rem.... The applicant calculated a 0- to 2-hour total body dose within 0.5 rem, which satisfies BTP ETSB 11-5. Based on the above, the staff finds the analysis acceptable."

GE has not been able to locate any published NRC document that establishes the 0.1 rem TEDE acceptance criterion. Therefore, it is concluded that the suggested 0.1 rem TEDE acceptance criterion has not been officially established by the NRC.

Use of the 0.1 rem TEDE acceptance criterion creates a verification/QA problem for GE. As demonstrated above, GE has located published objective evidence for the 2.5 rem TEDE and the 0.5 rem whole body acceptance criteria, but GE has no objective evidence that a 0.1 rem TEDE acceptance criterion for the Waste Gas System Failure formally exists. Thus, using that criterion for the ESBWR analysis would not be verifiable, and during a QA audit, GE could be sited under 10 CFR 50 App. B Criteria VI.

The suggested 0.1 rem TEDE acceptance criterion also creates a consistency problem, that future COL applicants will probably object to. An airborne release of 0.5 rem whole body ~ 0.5 rem TEDE, Part 20 has not changed, and BTP ETSB 11-5 has not changed. Thus, either the AP1000 FSER is in error (and should be corrected), or the ESBWR is being treated unfairly by being forced to have an acceptance criteria 1/5 that of the AP1000.

GE believes there are two logical and defendable options. First, based on the ESBWR design and the ABWR, the acceptance criteria would be 2.5 rem TEDE. Second, based on the actual published 1981 version of BTP ETSB 11-5 and AP1000, the acceptance criteria would be 0.5 rem TEDE.

(Any change to the NEDO-33175 has not yet been determined.)

3. The radiological acceptance criteria for all abnormal events classified in the new "accident" category (except the liquid-containing tank failure) should be 0.1 rem TEDE. The radiological acceptance criterion for the liquid-containing tank failure should remain as 10 CFR Part 20, Appendix B, Table 2, Column 2 as proposed.

GE Response:

The following is also supported by the responses to RAIs 2 and 4.

The NEDO-33175 Rev. 1 concluded that the only applicable regulations (except for the liquid-containing tank failure) for basing accident acceptance criteria are 10 CFR 50.34(a) and 10 CFR 100.11, while using the 10%, 25% and 100% proportions from the applicable Chapter 15 SRPs and Reg. Guides. A some what equivalent NRC sponsored SRP 11.3 review, which generated ETSB 11-5 Draft Rev. 3 - April 1996, concluded that the radiological acceptance criteria for the Waste Gas System Leak or Failure Analysis should be 25 mSv (2.5 rem), which is a "small fraction of (the) 10 CFR Part 100 limit." ETSB 11-5 Draft Rev. 3 - April 1996 replaced the use of 10 CFR 20 with 10 CFR 100 as the basis for the radiological acceptance criterion. Thus, the GE review, the Chapter 15 SRPs and the NRC sponsored SRP 11.3 review affectively came to the same conclusion, which is that radiological acceptance criteria for abnormal events beyond normal operations should be based on 10 CFR 50.34(a)/100.11, and not 10 CFR 20.

Therefore, without adequate objective evidence that the 0.1 rem TEDE accident (or *infrequent incident*, see the response to RAI 4) acceptance criterion has a published generic NRC acceptance, the 0.1 rem TEDE acceptance criterion cannot be justified, and thus, should not be used for the ESBWR or any other plant.

(No change to the NEDO-33175 is planned at this time.)

4. We recommend that the new category of events titled "accidents" be renamed "infrequent events" or "infrequent incidents" consistent with a dose acceptance criteria of 0.1 rem which is associated with normal operation.

GE Response:

GE agrees to rename the "*accidents*" category, and will use the term "*infrequent incidents*" to be consistent with ABWR DCD Tier 2 Subsection 15A.2.2. As a result of this change, the category name "*design basis accidents*" is simplified to be just "*accidents.*"

The following builds upon the response to RAI 3, which also addresses the use of a 0.1 rem acceptance criterion.

Per 10 CFR 50 App. A, normal operations include *AOOs,* however, *infrequent incidents* do not qualify AOOs, and thus, using "a dose acceptance criteria of 0.1 rem, which is associated with normal operation," is not appropriate. As shown in ABWR DCD/Tier 2 Table 15A-2, the acceptance criterion for infrequent incidents is "a small fraction of 10 CFR 100," which corresponds to 2.5 rem TEDE. The ABWR SER, NUREG-1503, Section 15.2 addresses non-accident events that result in radiological releases, and in three locations, the SER specifically states that the acceptance criteria is 10% of 10 CFR 100, and not once refers to 10 CFR 20.

The AP1000 event category, which is equivalent to the ESBWR *infrequent incidents* event category, is termed *infrequent faults.* For those events, AP1000 DCD Subsection 15.0.1.3 states "The release of radioactivity is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, in accordance with the guidelines of 10 CFR 100," and lists 10 events that are within that category. Of these events, two (in DCD Subsections 15.1.5 and 15.6.5) events were significant enough for radiological analyses to be provided. One event has a radiological acceptance criterion of 10% of the 10 CFR 50.34 25 rem TEDE, and the other has a radiological acceptance criterion of 100% of the 10 CFR 50.34 25 rem TEDE. Thus, the AP1000 was NRC approved with acceptance criteria of 100 and 250 times the 0.1 rem value proposed for the ESBWR.

Therefore, based on 10 CFR 50 App. A, ABWR licensing and AP1000 licensing, the 6.3 TEDE limit for ESBWR *infrequent incidents* should not change.

However, the review of RAI 4 spurred a proposed change to NEDO-33175, Table 6. Currently in NEDO-33175, the dose acceptance criterion for *AOOs In Combination With An Additional Single Active Component Failure or Single Operator Error* is in terms of a dose rate, while all the other dose acceptance criteria are based on integrated TEDE doses. The following proposed change eliminates this inconsistency, while complying with Part 20 with a 40 mrem margin to the 0.1 rem 10 CFR 20.1301 annual dose limit.

> The 10 CFR 20.1301(a)(1) 0.1 rem annual dose limit combined with (i.e., subtracting) the 10 CFR 20.1302(b)(2)(ii) 0.05 rem annual limit (for normal air born releases) would be a more appropriate radiological acceptance limit for an *AOO In Combination With An Additional Single Active Component Failure or Single Operator Error* (i.e., an *AOO with a single failure*), than the limit proposed in NEDO-33175, Rev. 1. Assuming that an individual at the exclusion boundary annually receives 0.05 rem from normal operations (which is conservative, when compared to the 10 CFR 50, App. I 10 millirad ALARA annual air born gamma dose guideline), and applying the 10 CFR 20.1301(a)(1) 0.1 rem annual dose limit.

The radiological acceptance criterion for an *AOO with a single failure* should be changed to 0.05 rem TEDE (i.e., 0.1 - 0.05). GE recommends this change to NEDO-33175.

5. The spent fuel cask drop accident in Tables 4 and 9 should be classified as a design-basis accident with its radiological acceptance criteria of 6.3 rem TEDE at the EAB and LPZ, and 5 rem TEDE control room. This accident is described in SRP Section 15.7.5.

GE Response:

GE agrees RAI 5, and, consistent with the response to RAI 4, will modify Tables 4 and 9, accordingly.

6. Review of the liquid-containing tank failure involves the groundwater and surface water environment to delay, disperse, dilute, or concentrate accidental radioactive liquid effluent releases (due to spills, leakages, or rupture of radwaste tanks), with emphasis on relating the effects of such releases to existing and known future uses of groundwater and surface water resources. This issue involves both site-specific hydrologic characteristics as well as radiological dose to the public. The radiological consequence analysis should include consideration of tank volumes, processing and decay of tank content (source term), and potential pathways of radioactive liquid to the environment. This is a design- and site-specific issue that should be classified as a Combined License Action item (e.g., AP1000 design certification).

GE Response:

GE agrees with RAI 6, and will modify NEDO-33175 and DCD Tier 2, accordingly.

7. In Table 9, you stated that the resulting control room doses from all abnormal events for meeting General Design Criteria (GDC) 19 are bounded by the loss-of-coolant accident (LOCA). The staff does not agree with this note based on its past review experience. The control room P/Q values for each event (accident) differ depending upon its release point relative to the control room air intake. Thus, the control room dose resulting from the LOCA may not necessarily bound GDC 19 criteria for all other accident events.

GE Response:

As stated above, the actual level of detail in the DCD Tier 2 safety analyses is outside the scope of NEDO-33175. However, since NEDO-33175 Rev. 1 was issued, the ESBWR DCD radiological analyses have progressed, and it is now known that the Main Steamline Break Outside Containment also may be the limiting event for control room dose. Therefore, Table 9 will be revised, and may be revised later, if needed.

8. Note "++" in Table 9 pertaining to the liquid-containing tank failure should be deleted. These tanks contain insignificant amounts of gaseous activity because they are normally either open to the atmosphere or the gaseous activities are continually released through the monitored plant vent (typically). Therefore, the gaseous activity release pathway from the liquid-containing tank failure is no longer required in the SRP.

GE Response:

GE agrees RAI 8, and will modify Table 9, accordingly.

9. Even though the ESBWR is a new design, we understand that the ESBWR design is based on current BWR technology and extensive BWR operating experience around the world. The development of initiating events is a complex task needing the use of operational experience, engineering judgment, PRA studies and deterministic analysis of transients and accidents. Please describe how deterministic analysis and BWR operating experience will be used with the PRA results to classify ESBWR abnormal events. It is expected that the ESBWR design control document (DCD) will include a design-specific justification for each event that is classified in the new "accident" category of events.

GE Response:

DCD will include design-specific justifications for each event that is classified in the NEDO-33175 Rev. 1 "accident" (changing to *infrequent incidents*) category of events.

The event frequencies associated with each of the events identified are being evaluated using the most straightforward methodology appropriate. The following events will probably require the development of a fault tree in addition to initiating event frequency data:

- Turbine Trip with Total Turbine Bypass Failure
- Loss of Feedwater Heating With Failure of Selected Control Rod Run-In
- Inadvertent SDC Function Operation
- Inadvertent Opening of a Depressurization Valve
- Inadvertent Opening of a Safety/Relief Valve
- Feedwater Controller Failure – Maximum Demand
- Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves
- Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves
- Generator Load Rejection with Total Turbine Bypass Failure

The following events will require only data evaluations or reference to previous evaluations that have been performed:

- Control Rod Withdrawal Error During Refueling
- Control Rod Withdrawal Error During Startup
- Control Rod Withdrawal Error During Power Operation
- Stuck Open Safety/Relief Valve
- Liquid-Containing Tank Failure
- Fuel Assembly Loading Error, Mislocated Bundle
- Fuel Assembly Loading Error, Misoriented Bundle
- Spent Fuel Cask Drop Accident
- Waste Gas System Leak or Failure

In many cases, simple fault tree analyses are sufficient to show that certain events are very unlikely. This is true because of the high reliability and fault tolerance associated with dual and triple redundant digital control systems.

The deterministic evaluation of initiating events is separated from the Event Classification. Deterministic analyses are performed assuming a specified event has occurred. The Event Classification is performed independently of deterministic safety analysis, to evaluate how likely it is for that event to occur. The result of the probabilistic analysis only affects the event acceptance criteria, not the initial conditions and assumptions used in the deterministic safety analysis.

Evaluations will utilize easily supportable assumptions necessary to support the event classification. More detail will be added to the analysis only when bounding assumptions are overly conservative in supporting the event classification. BWR operating experience will be used when relevant information is available about equipment failure or initiating event probabilities.

10. GE proposes to classify the "Fuel Assembly Loading Error, Mislocated Bundle" and "Fuel Assembly Loading Error, Misoriented Bundle" as "Accidents" for the ESBWR. By letter dated May 17, 2004, Global Nuclear Fuels (GNF) submitted proposed Amendment 28 to GESTAR II to change the acceptance criteria for these events for operating BWRs. By letter dated May 17, 2004, the staff requested additional information regarding proposed Amendment 28. The staff is waiting for the response to these RAIs from GNF. GE should classify these events as anticipated operational occurrences (AOOs) pending completion of the staff's review of GESTAR II Amendment 28.

GE Response:

Part of the responses to the informal RAIs on Amendment 28 are statistical data from 1995 to 2005 for 29 plants (i.e., 290 reactor years of operation), and that no BWR core has operated with a mislocated or misoriented fuel bundle during that period. Thus, for the existing BWR/2-6s actual plant data confirms that the frequency of occurrence of any FALE is far less than required to have that event classified as AOO.

The ESBWR is more advanced than the BWR/2-6s and will have, as a minimum, the same technology, e.g., X and Y, to prevent a "Fuel Assembly Loading Error, Mislocated Bundle" and "Fuel Assembly Loading Error, Misoriented Bundle" as does the ABWR. These events are classified not as *AOOs* or *infrequent incidents*, but as "Limiting Fault (Design Basis Accidents)" in ABWR DCD Tier 2 Table 15A-11. This position has been accepted by the NRC in NUREG-1503, Section 15.3. Therefore, the Fuel Assembly Loading Errors (FALEs) for ESBWR should be considered similar to those for the ABWR and not to FALEs for the 25+ year-old operating plants (designed in the 1960s and 1970s).

Therefore, GE believes it is appropriate to classify the FALEs as *accidents* in the next revision of NEDO-33175.

11. Section 15.0 of the SRP states that "for new applications, loss of offsite power should not be considered as a single failure event; rather it should be assumed in the analysis of each event without changing the event category. The applicant's Safety Analysis Report should discuss each transient and analysis to justify that it conforms to GDC 17 requirements. This position is based upon interpretation of GDC 17, as documented in the FSER for the ABB-CE System 80+ design certification. Please confirm that GE will follow this guidance in the transient and accident analysis for Chapter 15 events for the ESBWR.

GE Response:

This is not a concern for the ESBWR, because no *AOO, infrequent incident* or *accident* mitigation function requires either offsite AC power or onsite emergency diesel generator AC power.

12. GE has proposed to categorize "Overpressure Protection" as a "Special Event." Overpressure protection is considered by the staff to be a protection feature rather than an event. Please provide additional clarification of the basis for GE's proposed categorization.

GE Response:

The event term will be changed to "MSIV Closure With Flux Scram."

13. Please describe how uncertainties in the determination of the event frequencies will be accounted for the in the classification of events.

GE Response:

Uncertainties will not be directly addressed in the determination of event frequencies. It is anticipated that the event frequencies of all of the NEDO-33175 Rev. 1 "accident" (changing to *infrequent incidents*) category events will be 2-3 times below the criterion for classifying an event as an *AOO.*

These margins are large enough to judge that the classifications will not be subject to change because of input sensitivities.

14. GE provided interpretations of several NRC regulations in the topical report. Please note that the NRC staff does not necessarily agree with all of the interpretations provided.

GE Response:

GE understands that the NRC does not agree with some of GE's interpretation of the regulations, and plans to work with the NRC staff to come to common understandings.

- Loss of ac power to the station auxiliaries (see subsection 15.2.6)
- Loss of normal feedwater flow (see subsection 15.2.7)
- Partial loss of forced reactor coolant flow (see subsection 15.3.1)
- Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition (see subsection 15.4.1)
- Uncontrolled RCCA bank withdrawal at power (see subsection 15.4.2)
- RCCA misalignment (dropped full-length assembly, dropped full-length assembly bank, or statically misaligned assembly) (see subsection 15.4.3)
- Startup of an inactive reactor coolant pump at an incorrect temperature (see subsection 15.4.4)
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (see subsection 15.4.6)
- Inadvertent operation of the passive core cooling system during power operation (see subsection 15.5.1)
- Chemical and volume control system malfunction that increased reactor coolant inventory (see subsection 15.5.2)
- Inadvertent opening of a pressurizer safety valve (see subsection 15.6.1)
- Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate containment (see subsection 15.6.2)

15.0.1.3 Condition III: Infrequent Faults

Condition III events are faults that may occur infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. The release of radioactivity is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, in accordance with the guidelines of 10 CFR 100. By definition, a Condition III event alone does not generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

- Steam system piping failure (minor) (see subsection 15.1.5)
- Complete loss of forced reactor coolant flow (see subsection 15.3.2)
- RCCA misalignment (single RCCA withdrawal at full power) (see subsection 15.4.3)

- Inadvertent loading and operation of a fuel assembly in an improper position (see subsection 15.4.7)

- Inadvertent operation of automatic depressurization system (see subsection 15.6.1)

- Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (see subsection 15.6.5)

- Gas waste management system leak or failure (see subsection 15.7.1)

- Liquid waste management system leak or failure (see subsection 15.7.2)

- Release of radioactivity to the environment due to a liquid tank failure (see subsection 15.7.3)

- Spent fuel cask drop accidents (see subsection 15.7.5)

15.0.1.4 Condition IV: Limiting Faults

Condition IV events are faults that are not expected to take place, but are postulated because their consequences include the potential of the release of significant amounts of radioactive material. They are the faults that must be designed against, and they represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in doses in excess of the guideline values of 10 CFR 100. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults are classified in this category:

- Steam system piping failure (major) (see subsection 15.1.5)

- Feedwater system pipe break (see subsection 15.2.8)

- Reactor coolant pump shaft seizure (locked rotor) (see subsection 15.3.3)

- Reactor coolant pump shaft break (see subsection 15.3.4)

- Spectrum of RCCA ejection accidents (see subsection 15.4.8)

- Steam generator tube rupture (see subsection 15.6.3)

- LOCAs resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (see subsection 15.6.5)

- Design basis fuel handling accidents (see subsection 15.7.4)

15.1.5.4.6 Doses

Using the assumptions from Table 15.1.5-1, the calculated total effective dose equivalent (TEDE) doses for the case with accident-initiated iodine spike are determined to be less than 0.9 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 2.0 rem at the low population zone outer boundary. These doses are small fractions of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being 10 percent or less. The TEDE doses for the case with pre-existing iodine spike are determined to be less than 0.8 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 0.8 rem at the low population zone outer boundary. These doses are within the dose guidelines of 10 CFR Part 50.34.

At the time the main steam line break occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE. When this is added to the dose calculated for the main steam line break, the resulting total dose remains less than the values reported above.

15.1.6 Inadvertent Operation of the PRHR Heat Exchanger

15.1.6.1 Identification of Causes and Accident Description

The inadvertent actuation of the PRHR heat exchanger causes an injection of relatively cold water into the reactor coolant system. This produces a reactivity insertion in the presence of a negative moderator temperature coefficient. The overpower/overtemperature protection functions (neutron overpower, overtemperature, and overpower ΔT trips) are intended to prevent a power increase which could lead to a DNBR less than the safety analysis limit. In addition, because the cold leg temperature is reduced which depressurizes the reactor coolant system during this event, the low cold leg temperature or low pressurizer pressure protection functions could generate a reactor trip. These protection functions do not terminate operation of the PRHR heat exchanger.

The inadvertent actuation of the PRHR heat exchanger could be caused by operator error or a false actuation signal. Actuation of the PRHR heat exchanger involves opening one of the isolation valves, which establishes a flow path from one reactor coolant system hot leg, through the PRHR heat exchanger, and back into its associated steam generator cold leg plenum.

The PRHR heat exchanger is located above the core to promote natural circulation flow when the reactor coolant pumps are not operating. With the reactor coolant pumps in operation, flow through the PRHR heat exchanger is enhanced. The heat sink for the PRHR heat exchanger is provided by the IRWST, in which the PRHR heat exchanger is submerged. Because the fluid in the heat exchanger is in thermal equilibrium with water in the tank, the initial flow out of the PRHR heat exchanger is significantly colder than the reactor coolant system fluid. Following this initial insurge, the reduction in cold leg temperature is limited by the cooling capability of the PRHR heat exchanger. Because the PRHR heat exchanger is connected to only one reactor

If the post-LOCA cooling solution has a pH of less than 6.0, part of the cesium iodide may be converted to the elemental iodine form. The passive core cooling system provides sufficient trisodium phosphate to the post-LOCA cooling solution to maintain the solution pH at 7.0 or greater following a LOCA (see subsection 6.3.2.1.4).

15.6.5.3.2 In-containment Activity Removal Processes

The AP1000 does not include active systems for the removal of activity from the containment atmosphere. The containment atmosphere is depleted of elemental iodine and of particulates as a result of natural processes within the containment.

Elemental iodine is removed by deposition onto surfaces. Particulates are removed by sedimentation, diffusiophoresis (deposition driven by steam condensation), and thermophoresis (deposition driven by heat transfer). No removal of organic iodine is assumed. Appendix 15B provides a discussion of the models and assumptions used in calculating the removal coefficients.

15.6.5.3.3 Release Pathways

The release pathways are the containment purge line and containment leakage. The activity releases are assumed to be ground level releases.

During the initial part of the accident, before the containment is isolated, it is assumed that containment purge is in operation and that activity is released through this pathway until the purge valves are closed. No credit is taken for the filters in the purge exhaust line.

The majority of the releases due to the LOCA are the result of containment leakage. The containment is assumed to leak at its design leak rate for the first 24 hours and at half that rate for the remainder of the analysis period.

15.6.5.3.4 Offsite Dose Calculation Models

The offsite dose calculation models are provided in Appendix 15A. The models address the determination of the TEDE doses from the combined acute doses and the committed effective dose equivalent doses.

The exclusion area boundary dose is calculated for the 2-hour period over which the highest doses would be accrued by an individual located at the exclusion area boundary. Because of the delays associated with the core damage for this accident, the first 2 hours of the accident are not the worst 2-hour interval for accumulating a dose.

The low population zone boundary dose is calculated for the nominal 30-day duration of the accident.

For both the exclusion area boundary and low population zone dose determinations, the calculated doses are compared to the dose guideline of 25 rem TEDE from 10 CFR Part 50.34.

15.6.5.3.7.2 Core Release Source Term

The assumed core melt is a major conservatism associated with the analysis. In the event of a postulated LOCA, no major core damage is expected. Release of activity from the core is limited to a fraction of the core gap activity.

15.6.5.3.7.3 Atmospheric Dispersion Factors

The atmospheric dispersion factors assumed to be present during the course of the accident are conservatively selected. Actual meteorological conditions are expected to result in significantly higher dispersion of the released activity.

15.6.5.3.8 LOCA Doses

15.6.5.3.8.1 Offsite Doses

The doses calculated for the exclusion area boundary and the low population zone boundary are listed in Table 15.6.5-3. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.

The reported exclusion area boundary doses are for the time period of 0.8 to 2.8 hours. This is the 2-hour interval that has the highest calculated doses. The dose that would be incurred over the first 2 hours of the accident is well below the reported dose.

At the time the LOCA occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after 8 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

15.6.5.3.8.2 Doses to Operators in the Main Control Room

The doses calculated for the main control room personnel due to airborne activity entering the main control room are listed in Table 15.6.5-3. Also listed on Table 15.6.5-3 are the doses due to direct shine from the activity in the adjacent buildings and sky-shine from the radiation that streams out the top of the containment shield building and is reflected back down by air-scattering. The total of the three dose paths is within the dose criteria of 5 rem TEDE as defined in GDC 19.

As discussed above for the offsite doses, there is the potential for a dose to the operators in the main control room due to iodine releases from postulated spent fuel boiling. The calculated dose from this source is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

Appendix I, regarding population doses resulting from gaseous effluents. This is COL Action Item 11.3-1, as identified in DCD Tier 2 Table 1.8-2, and this COL Action Item addresses Criterion 8, which was discussed in Section 11.1 of this report. The staff finds this COL action acceptable for the compliance with the requirements of 10 CFR Part 50, Appendix I.

DCD Tier 2 Table 11.3-4 gives the ratios of the airborne concentrations of radionuclides, given in 10 CFR Part 20, Appendix B, Table 2, Column 1, at the site boundary, to the concentration limits for these radionuclides. The table shows that the sum of the ratios is 0.33, within the criterion of 1.00 set forth in Table 2, Note 4. In response to RAI 460.001, the applicant explained that the assumption used for the maximum release concentrations are based on 1 percent failed fuel with the exception of iodine and noble gases, which are limited by TS to 0.25 percent failed fuel. The staff finds this assumption acceptable because one percent fuel failure is consistent with SRP 11.3, and the TS limit provides a justified basis for the deviation of iodine and noble gases. On the basis of the results in DCD Tier 2 Table 11.3-4, the staff finds that the WGS design complies with 10 CFR Part 20, Section 1302, and Criterion 7, which was discussed in Section 1.1 of this report.

In its response to RAI 460.005(A), the applicant provided a waste gas system leak or failure analysis, and the justification for the assumptions used in that analysis. The analysis was performed to demonstrate that the WGS design meets the applicable guidelines of BTP ETSB 11-5. The BTP stipulates that the total body dose at the exclusion area boundary (EAB), as a result of the release of radioactivity for two hours from a postulated failure of the WGS, calculated in accordance with the BTP assumptions, should not exceed 0.5 rem. The applicant analyzed the accident using a short-term (0-2 hours) χ/Q of 6×10^{-4} sec/m^3 at the EAB, a release duration of one hour, instead of two hours, as suggested by the BTP, and other assumptions which agreed with those in the BTP. The applicant justified, in the above response, that a release duration of one hour is consistent with the isolation time of the AP1000 design. The applicant calculated a 0- to 2-hour total body dose within 0.5 rem, which satisfies BTP ETSB 11-5. Based on the above, the staff finds the analysis acceptable.

The staff reviewed all applicable information submitted in DCD Tier 2 Section 11.3 and in the applicant's responses to staff RAIs related to radwaste management systems for the AP1000. Based on the above information as discussed in the evaluation, the staff concludes that the WGS meets the requirements of 10 CFR 50.34a, as it relates to sufficient design information being provided to demonstrate that design objectives for equipment necessary to control releases of radioactive effluents to the environment have been met.

11.3.2 Conclusion

Based on the above, the staff concludes that the design of the WGS is acceptable in that it meets the acceptance criteria provided in SRP Section 11.3, as follows:

- The system is capable of maintaining gaseous effluents in unrestricted areas below the limits stated in 10 CFR 20.1302 during periods of fission product leakage at design levels for the fuel.

with GDC 60, and (5) the potential for gaseous releases resulting from hydrogen explosions in the gaseous radwaste system. The staff also reviewed the capability of the offgas system to limit the whole-body dose to less than 10 percent of the 10 CFR Part 100 limits for an individual exposure of 2 hours at the nearest exclusion area boundary (EAB) as a result of radioactive releases from a postulated offgas system leak or failure as assumed in BTP ETSB 11-5, Revision 0, July 1981. The staff reviewed all the applicable information provided in the amended SSAR as well as GE's submittals dated May 23, June 29, August 22, September 14, and October 26, 1990, December 19, 1991, and June 23, 1993.

The staff concludes that the gaseous radioactive waste management system design for the ABWR meets the applicable requirements of 10 CFR 20.1302 and GDC 3, 60, and 61 with regard to radioactivity in gaseous effluents released to unrestricted areas, fire protection, control of releases of radioactive materials, and radioactivity control in the gaseous waste management system and ventilation system associated with fuel storage and handling areas.

The staff expects that the COL applicant will provide an operational demonstration that the system design complies with Appendix I to 10 CFR Part 50 numerical guidelines for offsite radiation doses as a result of gaseous or airborne radioactive effluents during normal plant operation, including anticipated operational occurrences. Therefore, this was identified as DFSER COL Action Item 11.3.2-1. By amended SSAR Chapter 11, GE included COL License Information in Section 11.3.11.1, which states that the COL applicant will demonstrate this above compliance. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

Nonetheless, the staff has evaluated the ABWR design to determine if there is reasonable assurance that the COL applicant will be able to meet the Appendix I dose guidelines for design objectives. The ingestion, inhalation, and external irradiation of ground contamination pathway doses to applicable organs resulting from release of radioactive iodines, radioactive material in particulate form, and tritium and carbon-14 via airborne effluents depend on a number of site-dependent parameters. The population exposures (person-rem) and associated cost-benefit analysis are also site dependent. Therefore, the staff considered only if the standard design for the gaseous waste management system complies with Appendix I guidelines for external doses to any individual in an unrestricted area as a result of noble gas radionuclides in gaseous effluents. The staff concludes that there is reasonable assurance that ABWRs at sites that have an atmospheric dispersion factor (χ/Q) equal to or less than 9.8×10^{-6} sec/m^3 will meet the above dose guidelines (.05 mSv (5 mrem) per year to the total body).

Using the assumptions given in BTP ETSB 11-5 for analyzing a postulated leak or failure of a waste gas system and the EAB 0-2 hour χ/Q of 1.37×10^{-3} sec/m^3 (used in Chapter 15 of this report), the staff has determined that the wholebody dose at the EAB is less than 10 percent of the 10 CFR Part 100 limit. Therefore, the staff concludes that for all sites that have equal to or less than the above χ/Q at the EAB, the offgas system design will meet the above dose criterion and will be acceptable. This will be verified for each COL applicant.

These conclusions referred to above are based on the following findings:

(1) The ABWR design meets the requirements of GDC 60 and 61 by ensuring that the gaseous waste management system includes the equipment and instruments necessary to detect and control the release of radioactive materials in gaseous effluents.

(2) On the basis of expected radwaste inputs over the life of the plant, the staff has determined the releases of radioactive materials (noble gases, iodines, particulate, tritium and carbon-14) in gaseous effluents resulting from normal operation, including anticipated operational occurrences. The staff used the calculated releases for noble gases (Table 11.2 of this report) to determine the bounding value for χ/Q and assumed a 4-minute decay of the noble gas radionuclides during transit from the release point to the unrestricted area. The staff used the dose models and values for parameters given in RG 1.109 (Rev. 1) to evaluate compliance with Appendix I to 10 CFR Part 50. To calculate the external dose of noble gas radionuclides, the staff assumed a semi-infinite cloud model for the gaseous effluents. For the bounding χ/Q value quoted above, the staff calculated a total body dose (the limiting external dose) of 0.05 mSv/yr (5 mrem/yr), which meets the applicable Appendix I dose guideline.

(3) The ABWR design meets the requirements of 10 CFR Part 20 because the staff has considered the potential consequences resulting from reactor operation with a postulated fission product release rate consistent with an offgas noble gas release rate of 3.7×10^6 Bq/MWt-sec (100 μCi/MWt-sec) at 30 minutes decay for a BWR and estimated that, under these conditions, the concentration of radionuclides in gaseous effluents in unrestricted areas with a value of χ/Q that is equal to or less

and is appropriate and conservative for use as proposed for the ABWR.

(5) Increase in Reactor Coolant Inventory

GE analyzed inadvertent startup of the high-pressure core flooder (HPCF) pump. (Feedwater flow control failure to maximum demand is covered in Category (1).)

The transient which could cause unplanned addition to coolant inventory is the inadvertent actuation of the HPCF system. The HPCF system actuation has little effect because its flow is small compared to the recirculation flow. Because the HPCF full flow is a small contributor to total core flow, the increase in total coolant inventory is also small. GE's analysis shows that the consequences of this small inventory increase has little effect on fuel thermal margins and reactor system pressure. In accordance with SRP Section 15.5.1, this is acceptable.

(6) Decrease in Reactor Coolant Inventory

The anticipated operational occurrence of the inadvertent opening of a safety relief valve is covered in Category (1) above.

GE indicated that non-safety-grade equipment is credited for the high water level 8 trip, use of turbine bypass valves, and recirculation pump trip on load/turbine trip.

The staff questioned the appropriateness of GE taking credit for equipment that is not safety grade in the transient analysis as stated in DSER (SECY-91-355) Open Item 138. GDC 1 through 4 require that components important to safety be designed to be commensurate with the quality standards, and GDC 21 requires that the protection system be designed for high functional reliability. GE listed in a table the redundancy, isolation, environmental, seismic, periodic testing, and QA requirements for the equipment.

Even though the equipment discussed above will not be categorized as safety grade, it is of high quality and has sufficient redundancy to ensure its operability. To ensure an acceptable level of performance for the ABWR, GE committed to identify the above equipment in the ABWR technical specifications (TS) with regard to availability, set points and surveillance testing. This was DFSER TS Item 15.1-1. GE included the level 8 trip, the RIP trip and the turbine bypass in the proposed ABWR TSs. This is acceptable and TS Item 15.1-1 is resolved.

By letter dated August 23, 1989, GE informed Gulf States Utilities Company of a condition that could be reportable under 10 CFR Part 21, applicable to the River Bend Station. This condition involves a slow closure of one main turbine control valve. This low probability event, which was not previously considered, results from a turbine control valve that GE assumes to close as a result of an unspecified failure in the turbine control circuit or in the servo-mechanism hardware. According to GE, if the valve closes in less than 2.3 seconds, a reactor scram will be initiated as a result of high neutron flux and no safety limits will be exceeded. However, if the valve closes in more than 2.3 seconds, the reactor scram will be initiated by high reactor pressure. During this slow-closure case, the MCPR safety limit may be exceeded if the maximum combined flow limiter is set for less than 113 percent of rated steam flow. GE based the consequences of this postulated event on its assessment of a generic BWR/6 analysis. This was identified as DSER (SECY-91-355) Open Item 139. The staff, however, requested that GE address the event for ABWR applicability. In response, GE performed an ABWR-specific analysis for the slow closure of one turbine control valve with the remaining three control valves remaining open. In this case, the neutron flux increase will not reach the high neutron flux scram set point. Since the available turbine bypass capacity will be high enough to bypass all steam flow not passing through the remaining three turbine control valves, the reactor power settles back to its steady state. (The total steam flow through three control valves will increase to about 85 percent, and the remaining 15 percent of flow will pass through the slow-opening control valve and the bypass valves.) During this transient, the peak fuel surface heat flux will not exceed 104 percent of its initial value. The MCPR remains above the safety limit and is acceptable. Therefore, DSER Open Item 139 was resolved.

15.2 Trip of All Reactor Internal Pumps and Pressure Regulator Down-Scale Failure

For the postulated trip of all of the RIPs with offsite power available, GE postulated a common-mode failure of the adjustable speed drives. GE estimated that a fraction of low burnup fuel rods will achieve boiling transition during this event although test results indicate no fuel failures would occur. The staff classified this postulated event in the special category of anticipated transients involving a common-mode software failure and established a special acceptance criterion for the radiological dose calculation. The staff will not require that fuel failure be assumed in dose calculations for fuel rods that are under approximately 600 °C (1111 °F) for less than 60 seconds. This time and temperature criterion is based on test data for fuel that has achieved up to 20 gigawatts days per metric ton (GWD/MtU) (18 GWD/t) burnup; thus, it may be applied only to fuel with burnup of less than 20 GWD/MtU (18 GWD/t). For fuel with greater burnup, the dose calculations must assume fuel failure for all fuel rods that achieve transition boiling. In the equilibrium cycle, the higher burnup fuel accounts for about 45 percent of the total fuel bundles. The power generated by these bundles is usually 20 percent less than that of the hottest bundles, and less than 0.2 percent of these rods are expected to enter transition boiling. Because none of the hottest fuel rods exceed the time and temperature failure criterion, the radiological dose requirements limit of 10 percent of 10 CFR Part 100 are satisfied.

For the pressure regulator down-scale failure to occur, all three channels would have to suffer a common-mode failure before the pressure regulator would go either up or down the scale. If the pressure regulator failed down-scale, the steam control valves would close causing the reactor pressure and reactivity to increase. When reanalyzing this postulated event, GE proposed to assume that any fuel rods that achieve transition boiling fail for the purposes of the radiological dose calculation. 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. GE originally categorized this event as an accident. The staff believes that it is more appropriate to apply a special classification for such an event. 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.

According to GE analysis, during this event, it is estimated that less than 0.2 percent of fuel rods enter transition boiling and the requirement that the limit of 10 percent of 10 CFR Part 100 not be exceeded is met.

The staff will treat the above two postulated events as special cases, applicable only for the ABWR. This is due to the unique design features of the ABWR instrumentation and control systems, which reduce the frequency of such events; therefore, allowing these events to be recategorized as special cases. This resolved DSER (SECY-91-355) Open Item 136.

15.3 Accidents

GE analyzed RIP seizure and shaft break accidents. In the unlikely event of the pump motor shaft of 1 of the 10 RIPs stops instantaneously, a very rapid decrease of pump flow will result from the large hydraulic resistance introduced by the stopped rotor or shaft and cause pump seizure or shaft break. Consequently, core inlet flow and core cooling capability decreases. However, GE's analysis shows that with only 1 out of 10 RIPs seized, the core flow decrease is small (< 10 percent), so the event is mild. The RIP seizure and shaft break do not result in any fuel failure. This satisfies the dose limit criteria of 10 CFR Part 100 and is acceptable.

GE's analyses of the mislocated fuel bundle accident, misoriented fuel bundle accident, rod ejection accident, and control rod drop accident are discussed below.

(1) Mislocated Fuel Bundle Accident

Three errors must occur for this event to take place: (a) a bundle must be misloaded into a wrong location in the core; (b) the bundle, which was supposed to be loaded where the mislocation occurred, also is put in an incorrect location or discharged; and (c) the misplaced bundles are overlooked during the core verification process after core loading. A fuel loading error not detected by in-core instruments after fueling operations may result in an undetected reduction in thermal margin during power operations. However, GE evaluated the probability and consequences of a misplaced fuel bundle accident in equilibrium, first cycle, and subsequent cycle cores for current operating reactors and concluded that no fuel failure will occur and no radioactive material will be released from the fuel. The staff approved this analysis for operating plants and it is also applicable for the ABWR. This satisfies the criteria of 10 CFR Part 100 as required by the SRP for this event and is acceptable.

(2) Misoriented Fuel Bundle Accident

GE notified the staff by a 10 CFR Part 21 report (GE letter dated June 19, 1992, from S.J. Stark, "10 CFR Part 21, Reportable Condition, Rotated C or S-Lattice Fuel Assembly") that a fuel misorientation event may lead to fuel damage in BWR/6 designs. The staff required GE to evaluate the applicability of the issue for the ABWR and discuss its evaluation in the SSAR. This was DFSER Open Item 15.3-1.

Table 15A-1 Unacceptable Consequences Criteria Plant Event Category: Normal Operation

Unacceptable Consequences
1-1 Release of radioactive material to the environs that exceed the limits of either 10CFR20 or 10CFR50.
1-2 Fuel failure to such an extent that were the freed fission products released to the environs via the normal discharge paths for radioactive material, the limits of 10CFR20 would be exceeded.
1-3 Nuclear system stress in excess of that allowed for planned operation by applicable industry codes.
1-4 Existence of a plant condition not considered by plant safety analyses.

Table 15A-2 Unacceptable Consequences Criteria Plant Event Category: Moderate Frequency Incidents (Anticipated Operational Transients)

Unacceptable Consequences
2-1 Release of radioactive material to the environs that exceed the limits of 10CFR20.
2-2 Reactor operation induced fuel cladding failure.
2-3 Nuclear system stress exceeding that allowed for transients by applicable industry codes.
2-4 Containment stresses exceeding that allowed for transients by applicable industry codes.

Table 15A-3 Unacceptable Consequences Criteria Plant Event Category: Infrequent Incidents (Abnormal Operational Transients)

Unacceptable Consequences
3-1 Radioactive material release exceeding of a small fraction of 10CFR100.
3-2 Fuel damage that would preclude resumption of normal operation after a normal restart.
3-3 Generation of a condition that results in consequential loss of function of the reactor coolant system.
3-4 Generation of a condition that results in a consequential loss of function of a necessary containment barrier.

Table 15A-11 Limiting Faults (Design Basis Accidents)

NSOA Event No.	Event Description	NSOA Event Figure No	Safety Analysis Section No.	BWR Operating State A	B	C	D
28	Control Rod Ejection Accident	15A-33	15.4.8	X	X	X	X
29	Control Rod Drop Accident	15A-34	15.4.9	X	X	X	X
30	Control Rod Withdrawal Error—Power Operation	15A-35	15.4.2				X
31	Fuel-Handling Accident	15A-36	15.7.4	X	X	X	X
32	Loss-of-Coolant Accident Resulting from Spectrum of Postulated Piping Breaks Within the RCPB Inside Containment	15A-37 and 15A-38	15.6.5			X	X
33	Small, Large, Steam and Liquid Piping Breaks Outside Containment	15A-39 and 15A-40	15.6.4			X	X
34	Gaseous Radwaste System Leak or Failure	15A-41	15.7.1	X	X	X	X
35	Augmented Offgas Treatment System Failure	15A-42	15.7.1	X	X	X	X
36	Liquid Radwaste System Leak or Failure	15A-43	15.7.2	X	X	X	X
37	Liquid Radwaste System Storage Tank Failure	15A-44	15.7.3	X	X	X	X
41	Trip of All RIPs	15A-48	15.3.1			X	X
42	Loss of RHR Shutdown Cooling	15A-49	15.2.9	X	X	X	X
43	RHR Shutdown Cooling Increased Cooling	15A-50	15.1.6	X	X	X	X
46	Pressure Regulator Failure—Closure of all Bypass and Control Valves	15A-53	15.2.1			X	X
50	Misplaced Fuel Bundle Accident	15A-57	15.4.7	X	X	X	X
51	Reactor Internal Pump Seizure	15A-58	15.3.3				X
52	Reactor Internal Pump Shaft Break	15A-59	15.3.4				X