Enclosure

SAFETY EVALUATION ON THE USE OF A SINGLE-EARTHQUAKE DESIGN FOR SYSTEMS, STRUCTURES, AND COMPONENTS IN THE ABWR STANDARD PLANT

A. INTRODUCTION

Appendix A to 10 CFR Part 100 requires, in part, that all structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subject to an operating basis earthquake (OBE). Changes to Appendix A to Part 100 are being proposed to redefine the OBE to a level such that the function of the OBE can be satisfied without the need to perform explicit responses analyses.

The purpose of this safety evaluation is to identify the necessary changes to existing seismic design criteria that are acceptable to the NRC staff for implementing the proposed rule change as it pertains to the design of safety-related systems, structures, and components in the General Electric Advanced Boiling Water Reactor (ABWR). These criteria apply only to the ABWR standard plant design and are not intended to replace the seismic design criteria approved by the Commission in the licensing bases of currently operating facilities. The guidelines provided herein are proposed for use as a pilot program for implementing the proposed rule change specifically for the ABWR.

B. BACKGROUND

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the staff requested the Commission's approval to decouple the level of the OBE ground motion from that of the safe-shutdown earthquake (SSE). The Commission approved the staff's position in its Staff Requirements Memorandum (SRM) of June 26, 1990.

In the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," the staff further requested the Commission to approve eliminating the OBE from the design of systems, structures, and components in both evolutionary and passive advanced reactors designs. The proposed amendment to 10 CFR Part 100, Appendix A would allow, as an option, that the OBE be eliminated from design certification when the OBE is established at less than or equal to one-third the SSE. In this manner, the OBE serves the function as an inspection level earthquake below which the effect on the health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect for damage. The elimination of the OBE from design was requested by the Electric Power Research Institute (EPRI) and also recommended by the Advisory Committee on Reactor Safety (ACRS) in its letter of April 26, 1990.

In the draft Commission paper, "Design Certification and Licensing Policy

Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs," the staff examined the safety impact of eliminating the OBE as it pertains to civil structures, piping systems, and equipment seismic qualification. Several recommendations were made by the staff to ensure that eliminating the OBE would not result in a significant decrease in the overall plant safety margin. The following sections of this safety evaluation contain the specific actions needed for the ABWR standard plant design to ensure that adequate safety margins are maintained when the OBE is eliminated from the design. The sections identify those actions needed for: (1) piping systems, (2) concrete and steel structures, (3) equipment seismic qualification, and (4) pre-earthquake planning and post-earthquake operator actions.

C. ASME CODE CLASS 1, 2, AND 3 COMPONENTS AND CORE SUPPORT STRUCTURES

The dynamic analysis methods to be used for seismic analyses of ASME Code Class 1, 2, and 3 components and core support structures in the ABWR shall use those methods described in the ABWR SSAR as approved by the NRC staff in its Final Safety Evaluation Report (FSER). The loads and load combinations to be used for evaluating ASME Code Class 1, 2, and 3 components and core support structures are provided in the ABWR SSAR and discussed in the staff's FSER. The OBE may be eliminated from the applicable design load combination when the following supplemental criteria are used.

1. Fatigue

In order to ensure adequate design considerations for the fatigue effects of earthquake cycles, it is necessary to establish a bounding load definition and number of earthquake cycles to account for the more frequent occurrences of lesser earthquakes and their aftershocks. For the ABWR, an acceptable cyclic load basis for fatigue analysis of earthquake loading for ASME Code Class 1, 2, and 3 components and core support structures is two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). This is equivalent to the cyclic load basis of one SSE and five OBE events as currently recommended in the Standard Review Plan (NUREG-0800) Section 3.9.2. Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987.

2. Seismic Anchor Motion (SAM)

For the ABWR, the effects of displacement-limited, seismic anchor motion (SAM) due to a safe-shutdown earthquake should be evaluated for safety-related ASME Code Class 1, 2, and 3 components and component supports to ensure their functionality during and following an SSE. The SAM effects should include (but are not limited to) relative displacements of piping between building floors and slabs, at equipment nozzles, at piping penetrations, and at connections of small-diameter piping to large-diameter piping.

For piping systems, the effects of seismic anchor motions due to a safe-shutdown earthquake should be combined with the effects of other normal operational loadings that might occur concurrently as specified in Section C.3.1 and C.3.2 of this safety evaluation.

3. <u>Piping Stress Limits</u>

For ASME Code Class 1, 2, and 3 piping, the design requirements in the 1989 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Subsections NB, NC, and ND shall be met. In addition, the following changes and additions to paragraphs NB-3650, NC-3650, and ND-3650 are necessary and shall be satisfied for piping systems when the OBE is eliminated from the design.

3.1 ASME Code Class 1 Piping Stress Limits

(a) For primary stress evaluation (NB-3654.2), earthquake loads are not required to be evaluated for consideration of Level B Service Limits for Eq.(9).

(b) For satisfaction of primary plus secondary stress intensity range (NB-3653.1), in Eq. (10), M_i shall be either (1) the resultant range of all loads considering one-half the range of the safe-shutdown earthquake or (2) the resultant range of moment due to the full range of the safe-shutdown earthquake alone, whichever is greater. The use of the safe-shutdown earthquake is intended to provide a bounding design for the cumulative effects of earthquakes of a lesser magnitude and is therefore to be included in consideration of Level B Service Limits for Eq.(10). A reduced range (with an equivalent number of fractional vibratory peak cycles) of the safe-shutdown earthquake moment may be used for consideration of Level B Service Limits (but with a range not less than one-third of the maximum SSE moment range).

(c) For satisfaction of peak stress intensity (NB-3653.2), the load sets developed in NB-3653.1 based on the above Position C.3.1(b) should be used in calculating the peak stress intensity, S_p, and the alternating stress intensity, S_{alt}, for evaluating the fatigue effects and cumulative damage.

(d) For simplified elastic-plastic discontinuity analysis (NB-3653.6), if Eq. (10) cannot be satisfied for all pairs of load sets, then the alternative analysis as described in NB-3653.6 should be followed. In addition, the following condition shall be satisfied:

$$S_{sam} = C_2 \frac{D_o}{2I} (M_i^* + M_i^{**}) \leq 6.0\ S_m \qquad \text{Eq. (12a)}$$

where: S_{sam} is the nominal value of seismic anchor motion stress

M_i^* is the same as M_i^* in Eq. (12)

M_i^{**} is the same as M_i in Eq. (10), except that it includes only moments due to seismic anchor motion displacements caused by a safe-shutdown earthquake

The combined moment range ($M_i^* + M_i^{**}$) shall be either (1) the resultant range of thermal expansion and thermal anchor movements plus one-half the range of the safe-shutdown earthquake anchor motion or (2) the resultant range of moment due to the full range of the safe-shutdown earthquake anchor motion alone, whichever is greater

3.2 ASME Code Class 2 and 3 Piping Stress Limits

(a) For consideration of occasional loads (NC/ND-3653.1), earthquake loads (i.e., inertia and seismic anchor motion) are not required for satisfying Level B Service Limits for Eq.(9).

(b) For consideration of thermal expansion or secondary stresses (NC/ND-3653.2), M_c in Eq. (10) is not required to include the moment effects of seismic anchor motions due to an earthquake.

(c) For consideration of secondary stresses in Level D Service Limit (NC/ND-3655), the following condition should be satisfied:

$$S_s = i \frac{M_c^* + M_c}{Z} \leq 3.0\ Sh \qquad \text{Eq. (10b)}$$

where: M_c^* is the range of moments due to seismic anchor motions due to a safe-shutdown earthquake

M_c is the range of moments due to thermal expansion

4. Pipe Break Postulation Without OBE

It is recognized that pipe rupture is a rare event which might only occur under unanticipated conditions, such as those which might be caused by possible design, construction, or operational errors; unanticipated loads or unanticipated corrosive environments. The staff's observation of actual piping failures have found that they generally occur at high stress and fatigue locations, such as at the terminal ends of a piping system at its connection to component nozzles. Currently, in accordance with Standard Review Plan (NUREG-0800) Section 3.6.2, Revision 2 dated June 1987, pipe breaks are postulated in high energy piping at locations of high stress and high fatigue usage factor. The load combination used in calculating the high stress and usage factor includes normal and upset load conditions (i.e., pressure, weight, thermal, OBE, and other operational transient loadings).

From a historical viewpoint, the criteria for postulating high energy breaks at specified locations were first introduced in the early 1970s. The basis for the mechanistic approach for selecting pipe break locations was derived from the premise that although pipe breaks could result from random events induced by unanticipated conditions, the failure mechanism and the expected location of failure would likely be caused by local conditions of high stress or high fatigue in the piping. In order to insure that a sufficient number of pipe breaks would be postulated, breaks were recommended to be postulated for a wide spectrum of events to envelope the uncertainties of unanticipated failure mechanisms. Breaks were postulated at terminal ends of the piping, at high stress and high fatigue locations, and as a minimum at two additional intermediate locations when the stresses were below the high stress threshold limit. The resulting criteria which were incorporated in Standard Review Plan Section 3.6.2 resulted in many postulated pipe break locations and caused the installation of numerous pipe rupture mitigation devices in nuclear plants.

In the mid-1980s, the NRC's Executive Director for Operations initiated a comprehensive review of nuclear power plant piping to identify areas where changes to the piping requirements could improve the licensing process as well as the safety and reliability of nuclear power plants. The NRC's Piping Review Committee (PRC) in an integrated effort with the nuclear industry under the Pressure Vessel Research Council conducted a comprehensive study of piping criteria including the mechanistic pipe break postulation guidelines. The PRC found that when an excessive number of pipe rupture mitigation devices (i.e., pipe whip restraints and jet impingement shields) are installed on high energy piping systems, the potential exists for piping systems to be overly constrained. This condition was found in several nuclear plants in which massive pipe restraints adversely affected the ability of the high temperature piping to freely expand during normal plant operation. The PRC also found through numerous dynamic tests and field observations of non-seismically designed piping systems that had undergone high seismic loadings that butt-welded piping possesses an inherent ability to withstand large seismic inertial loadings without failure.

As a result of the PRC's effort, the NRC staff recognized that the mechanistic pipe rupture criteria for selecting locations of pipe breaks resulted in an excessive number of pipe rupture mitigation devices that could hinder the normal operation of the plant and might not contribute significantly to the overall safety of the plant. Accordingly, the Standard Review Plan was revised to reduce the number of postulated pipe breaks by (1) eliminating the need to postulate pipe breaks at the two arbitrary intermediate locations and (2) providing a leak-before-break approach in lieu of postulating pipe breaks when the system and material specific information is adequate to justify its application.

Based on recent dynamic pipe tests conducted by the EPRI and NRC, it has been demonstrated that the piping can withstand seismic inertial loadings higher than an SSE without rupturing. Thus, the staff believes the likelihood of a pipe break in a seismically-designed piping system due to an earthquake magnitude of one-third SSE is remote. Operating experience has shown that pipe breaks are more likely to occur under conditions caused by normal operation (e.g., erosion-corrosion, thermal constraint, fatigue, and operational transients).

On the basis of the above discussion, the staff concludes that no replacement earthquake loading should be used to establish postulated pipe rupture locations. Instead the criteria for postulating pipe breaks in seismically-designed, high energy piping systems should be based on factors attributed to normal and operational transients only. The staff's revised criteria for pipe break postulation are provided below. The revised criteria are intended to ensure that breaks are postulated to occur at the most likely locations and to reduce the number of pipe rupture mitigation devices (e.g., pipe whip restraints and jet impingement shields) that might hinder plant operation without providing a compensatory level of safety.

The elimination of earthquake loads in the revised pipe break criteria below is justified, in part, on the fact that the equipment environmental qualification and compartment pressurization analyses for the ABWR are based on a worst-case break assumption in each compartment and are not postulated at mechanistic break locations. In addition, GE should commit to a monitoring program for erosion-corrosion that provides assurances that procedures or administrative controls are in place to assure that the NUMARC program (or another equally effective program) is implemented and the structural integrity of all high-energy (two-phase as well as single-phase) carbon-steel systems is maintained as discussed in Generic Letter 89-08 and NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," April 1989.

Consistent with the above staff finding, the guidelines provided in SRP Section 3.6.2, Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," may be revised as follows:

B.1.b (1).(a): Footnote 2 should read, "For those loads and conditions in which Level A and Level B stress limits have been specified in the Design Specification (excluding earthquake loads)."

B.1.b.(1).(d): "The maximum stress as calculated by the sum of Eqs. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits have been specified in the system's Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) excluding earthquake loads should not exceed $0.8(1.8\ S_h + S_A)$."

D. CONCRETE AND STEEL STRUCTURES

1. SSE Relative Displacements Between Structures

As discussed in Appendix 3G to the SSAR, the seismic response (building displacements, structural member forces, floor response spectra, etc.) of the reactor building could be significantly underestimated, when the through-soil, structure-to-structure interaction effect is not considered. GE did not consider this effect in the analyses of certain ABWR structures such as the control building, ultimate heat sink pump house, radwaste building, and turbine building. This effect might be more pronounced for these other buildings because they are lighter than the reactor building and the energy feedback from the reactor building during an earthquake could significantly affect the seismic responses of these buildings.

Therefore, the staff concludes that the effects of through-soil, structure-to-structure interaction under SSE loadings for all structures housing seismically-designed piping should be determined under SSE loadings to establish the relative displacements between buildings (seismic anchor movement for piping systems).

2. Seismic Instrumentation

GE should ensure that adequate design provisions allow for the placement of seismic instrumentation in the free field so that the control room operator can be immediately informed through the event indicators when the response spectra level and the cumulative absolute velocity (CAV) experienced at this location exceeds the shutdown level and can take the necessary actions. The details of the instrumentation requirements are discussed in Section F of this safety evaluation.

3. Use of RG 1.143 and 1.27

The staff guidelines in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," and in RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," provide for a seismic design of radwaste buildings and ultimate heat sink features based on the operating basis earthquake. With the elimination of the OBE, the staff finds that these structures and features should be designed to withstand the safe-shutdown earthquake. The structural design criteria using the SSE loading should use the appropriate loads and load combinations provided in Standard Review Plan Section 3.8.4.

E. EQUIPMENT SEISMIC QUALIFICATION

When equipment qualification for seismic loadings is performed by analysis, testing, or a combination of both, the staff recommends the use of the IEEE Standard 344-1987 as endorsed in Regulatory Guide 1.100, Revision 2. This standard has detailed requirements for performing seismic qualification using five OBE events followed by an SSE event. With the elimination of the OBE, it is necessary to qualify equipment with the equivalent of five OBE events followed by one SSE event. Therefore, the staff concludes that equipment should be qualified with five ½SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five ½SSE events may be used in accordance with Appendix D of IEEE Standard 344-1987 when followed by one full SSE.

F. PRE-EARTHQUAKE PLANNING AND POST-EARTHQUAKE OPERATOR ACTIONS

The design certification of the ABWR using a single-earthquake (SSE) design is predicated on the adequacy of pre-earthquake planning and post-earthquake inspections for damage that are to be implemented by the COL applicant. The COL applicant shall submit to the NRC staff as a part of its application the procedures it plans to use for pre-earthquake planning and post-earthquake actions. For the ABWR, the NRC staff finds acceptable the criteria developed by the Electric Power Research Institute (EPRI) in EPRI Reports EPRI NP-5930, EPRI NP-6695, and EPRI TR-100082 together with the amendments, additions, and changes outlined below for evaluating the need to shut down the plant following an earthquake.

EPRI NP-5930

The EPRI Report NP-5930 shall be used with the following exceptions:

1. A free field instrument must be used for determining the CAV and the spectral acceleration level.

2. The response spectrum check is as follows:

   The 5% damped ground response spectrum for the earthquake motion at the site exceeds (1) the corresponding OBE response spectral acceleration between 2 and 10 Hz, or it exceeds an acceleration of 0.20g between 2 and 10 Hz whichever is greater, or (2) it exceeds the corresponding OBE response spectral velocity between 1 and 2 Hz or a velocity of 6 inches per second between 1 and 2 Hz, whichever is greater.

3. The licensee shall consider as sufficient evidence to shut down the plant the simultaneous exceedance of the 5% damped ground response spectrum enumerated in item 2 and the CAV exceedance of 0.16 g-sec for any one frequency on any one component of the free field ground motion. The CAV shall be determined in accordance with EPRI Report EPRI NP-100082. Also, any evidence of significant damage observed during the plant walkdown in accordance with the EPRI Report NP-6695 recommendations shall be sufficient cause for plant shutdown.

4. The instrumentation installed at the nuclear power plant shall be capable of on-line digital recording of all three components of the ground motion and of converting the recorded (digital) signal into the standardized CAV and the 5 percent damped response spectrum. The digitizing rate of the time history of the ground motions shall be at least 200 samples per second and the band-width shall be at least from 0.20 Hz to 50 Hz. The pre-event memory of the instrument shall be sufficient to record the onset of the earthquake.

5. The system must be capable of routinely calibrating the response spectrum check of 0.20g. Also, the CAV of 0.16g-sec should be calibrated with a copy of the October, 1987 Whittier, California earthquake or an equivalent calibration record provided for this purpose by the manufacturer of the instrumentation. In the event that an actual earthquake has been recorded at the plant site, the above calibration shall be performed to demonstrate that the system was functioning properly at the time of the earthquake.

EPRI NP-6695

The EPRI Report NP-6695 shall be used with the following exceptions:

Section 3.1. Short-Term Actions

Item 3. "Evaluation of Ground Motion Records"

There is a time limitation of four hours within which the licensee shall determine if the shutdown criterion has been exceeded. After an earthquake has been recorded at the site, the licensee shall provide a response spectrum calibration record and CAV calibration record to demonstrate that the system was functioning properly.

Item 4. "Decision on Shutdown"

Exceedance of the EPRI criterion as amended by the NRC or observed evidence of significant damage as defined by EPRI NP-6695 shall constitute a condition for mandatory shutdown unless conditions prevent the licensee from accomplishing an orderly shutdown without jeopardizing the health and safety of the public.

Add item 7. "Documentation"

The licensee shall record the chronology of events and control room problems while the earthquake evaluation is in progress.

Section 4.3. "Guidelines" (p. 4-3)
Because earthquake-induced vibration of the reactor vessel could lead to changes in neutron fluxes a prompt check of the neutron flux monitoring instruments shall be made to indicate if the reactor is stable. Therefore, this check should be added to the checks listed in this section.

Section 4.3.4.1. "Safe Shutdown Equipment" (p. 4-7):

In addition to the safe shutdown systems on this list containment integrity must be maintained following an earthquake. Since the containment isolation valves may have malfunctioned during the earthquake, inspection of the containment isolation system is necessary to assure continued containment integrity.

Section 4.3.4. "Pre-Shutdown Inspection"

Exceeding the EPRI criterion or evidence of significant damage should constitute a condition for mandatory plant shutdown, as the staff stated in its recommendation for Section 3.1, item 4, "Decision on Shutdown."