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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application, ASME Code Service Level C or D Acceptance Criteria, RAI 15.0-17 S02**

Enclosure 1 contains GE-Hitachi Nuclear Energy Americas (GEH) response to Supplement 2 to the NRC RAI transmitted via Reference 1.

If you have any questions or require additional information regarding the information provided here, please contact me.

Sincerely,

Kathy Sedney for

James C. Kinsey
Vice President, ESBWR Licensing

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NRO

Reference:

1. MFN 06-381, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application,* October 11, 2006.

Enclosure:

1. MFN 06-418 S02, Response to a Portion of NRC Request for Additional Information Letter No. 69, Related to ESBWR Design Certification Application, ASME Code Service Level C or D Acceptance Criteria, RAI Number 15.0-17 S02

cc:	AE Cubbage	USNRC (with enclosures)
	DH Hinds	GEH (with enclosures)
	RE Brown	GEH (w/o enclosures)
	eDRF	0058-8864 Rev. 2

ENCLOSURE 1

MFN 06-418 S02

Response to Portion of NRC Request for Additional Information Letter No. 69 Related to ESBWR Design Certification Application

ASME Code Service Level C or D Acceptance Criteria

RAI Number 15.0-17, Supplement 02

<u>NRC RAI 15.0-17</u>:

Standard Review Plan (SRP) Section 15.2.8.II.A.1, Revision 1, July 1981, states that pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures [ASME Boiler and Pressure Vessel Code, Section III, Service Level B] for low probability events and below 120% of the design pressures [ASME Boiler and Pressure Vessel Code, Section III, Service Level C] for very low probability events such as double-ended guillotine breaks.

DCD Tier 2, Rev. 1, Table 15.0-5, Acceptance Criteria for Infrequent Events, states that "pressures in the reactor coolant and main steam systems shall be maintained below the ASME Service Level C limit, which corresponds to 120% of design pressure."

Revise DCD Tier 2, Table 15.0-5 consistent with the SRP acceptance criteria for low probability events that the pressure in the reactor coolant and main steam systems should be maintained below 110 percent (Service Level B) of the design pressures.

<u>Response to RAI 15.0-17</u>:

Standard Review Plan (SRP) Section 15.2.8, Revision 1, July 1981 is titled "FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAIMENT (PWR)," and thus, is not applicable to BWRs.

Determination criteria for "low probability events" and "very low probability events" are not provided in the SRPs or the 10 CFR's, and thus, the ESBWR event types vs. Service Levels were compared to those of the operating plants and the ABWR. For the operating BWRs, the ABWR and the ESBWR, AOOs are considered to be (the low probability) events that apply ASME Boiler and Pressure Vessel Code, Section III, Service Level B, and the infrequent incidents/events, ATWS and accidents are considered to be (the very low probability) events that apply ASME Boiler and Pressure Vessel Code, Section III, Service Level C/D. Therefore, the ESBWR Service Level C acceptance criterion for infrequent events is consistent with past and existing BWR licensing bases.

No DCD change results from this RAI response.

NRC RAI 15.0-17 S01:

Section 15.0.1.2, subsection (4) defines an accident as a postulated Design Basis Event not expected to occur during the lifetime of the plant and the radiological releases not to exceed the calculated exposure in 10 CFR 50 34(a). It also states that it equates to an ASME Code Service level C or D acceptance criteria. The staff is not aware of such equivalency, except for ATWS, as GE stated in RAI response 15.0-17. ASME service levels should be justified on a case by case basis in a manner similar to ATWS and GE has not provided this justification in its response to RAI 15.0-17. Please explain how you concluded that all DBEs should be assigned service level C or D, or remove the statement from the DCD sections 15.0.1.2, 15.0.3.2 and 15.0.3.3.

Response to RAI 15.0-17 S01:

For plant design, how DBEs should be assigned ASME Code service levels is fully addressed in Tier 2 Subsection 3.9.3.1.1. (The projected Tier 2 Revision 4 version is attached, for completeness.) This explanation is consistent with the discussion in Tier 2 Subsection 15.0.1.2. Based on their probability of occurrence, some of the Chapter 15 accidents should be ASME Code Service Level D. However, for conservatism, the acceptance criterion for all Chapter 15 accident analyses is ASME Code Service Level C.

DCD Impact:

Tier 2 Subsection 3.9.3.1.1 for Rev. 4 is attached for information; however, no additional Tier 2 change is needed in response to this RAI.

3.9.3.1.1 Plant Conditions

All events that the plant might credibly experience during a reactor year are evaluated to establish design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence as discussed below and correlated to service levels for design limits defined in the ASME Boiler and Pressure Vessel Code Section III as shown in Tables 3.9-1 and 3.9-2.

Normal Condition

Normal conditions are any conditions in the course of system startup, operation in the design power range, normal hot standby (with condenser available), and system shutdown other than upset, emergency, faulted, or testing.

Upset Condition

An upset condition is any deviation from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include system operational transients (SOT), i.e., Anticipated Operational Occurrences, as defined in 10 CFR 50, Appendix A, which result from any single operator error or control malfunction, from a fault in a system component requiring its isolation from the system, or from a loss of load or power. Hot standby with the main condenser isolated is an upset condition.

Emergency Condition

An emergency condition includes deviations from normal conditions that require shutdown for correction of the condition(s) or repair of damage in the reactor coolant pressure boundary (RCPB). Such conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system. Emergency condition events include but are not limited to infrequent operational transients (IOT), e.g., infrequent events, as defined in Subsection 15.0.1.2, caused by one of the following: (a) a multiple valve blowdown of the reactor vessel; (b) LOCA from a small break or crack (SBL) which does not depressurize the reactor systems, does not automatically actuate the GDCS and Automatic Depressurization Subsystem (ADS), and does not result in leakage beyond normal make-up system capacity, but which requires the safety functions of isolation of containment and shutdown and may involve inadvertent actuation of the ADS; (c) improper assembly of the core during refueling; or (d) depressurization valve blowdown. An anticipated transient without scram (ATWS) or reactor overpressure with delayed scram (Tables 3.9-1 and 3.9-2) is a special event, as defined in Subsection 15.0.1.2, that is classified as an emergency condition.

Faulted Condition

A faulted condition is any of those combinations of conditions associated with extremely low-probability postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events, such as a LOCA, that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These events are the most drastic that must be considered in the design and thus represent limiting design bases. Faulted condition events include but are not limited to one of the following: (a) a fuel-handling accident; (b) a main steamline or feedwater line break; (c) the combination of any small/intermediate break LOCA (SBL or IBL) with the safe shutdown earthquake, and a loss of off-site power; or (d) the safe shutdown (SSE) earthquake plus large break LOCA (LBL) plus a loss of off-site power.

The IBL classification covers those breaks for which the GDCS operation occurs during the blowdown. The LBL classification covers the sudden, double ended severance of a main steamline inside or outside the containment that results in transient reactor depressurization, or any pipe rupture of equivalent flow cross sectional area with similar effects.

Correlation of Plant Condition with Event Probability

The probability of an event occurring per reactor year associated with the plant conditions is listed below. This correlation identifies the appropriate plant conditions and assigns the appropriate ASME Section III service levels for any hypothesized event or sequence of events.

Plant Condition	ASME Code Service Level	Event Encounter Probability per Reactor Year
Normal (planned)	A	1.0
Upset (moderate probability)	B	$1.0 > P \geq 10^{-2}$
Emergency (low probability)	C	$10^{-2} > P \geq 10^{-4}$
Faulted (extremely low probability)	D	$10^{-4} > P > 10^{-6}$

Safety-Related Functional Criteria

For any normal or upset design condition event, safety-related equipment and piping (Subsection 3.2.1) shall be capable of accomplishing its safety function as required by the event and shall incur no permanent changes that could deteriorate its ability to accomplish its safety function as required by any subsequent design condition event.

For any emergency or faulted design condition event, safety-related equipment and piping shall be capable of accomplishing its safety function as required by the event but repairs could be required to ensure its ability to accomplish its safety function as required by any subsequent design condition event.

NRC RAI 15.0-17 S02:

During an August 21, 2007 GE-NRC conference call, the NRC stated that the response to RAI 15.0-17 S01 was acceptable; however, the DCD should include a commitment to perform post-pressurization event inspections-testing, if any event caused an ESBWR reactor coolant system to exceed its ASME Code Service Level B pressure limit.

Response to RAI 15.0-17 S02:

The Tier 2 safety analyses demonstrate that no design bases event can cause an ESBWR reactor coolant system to exceed its ASME Code Service Level B pressure limit. However, ASME Code Section XI does provide adequate inspections/testing to confirm the operability of the safety-related components potentially affected by the hypothetical pressurization event. Therefore, the ASME Code is used as the basis for the requested post-event inspections.

DCD Impact:

The following markup to DCD Tier 2 Subsection 3.9.3.1.2 will be reflected in Revision 5 in response to this request.

3.9.3.1.2 Inspections/Testing Following The Reactor Coolant System Exceeding Service Level B Pressure Limit

If any abnormal event causes the pressure within reactor coolant system to exceed 110% of its design value (i.e., exceed the ASME Code Service Level B pressure limit), an inspection program should be satisfactorily completed, before normal plant operations may proceed. Within ASME Code, Section XI, Subarticles IWB-2400 and IWB-2500 there are inspection specifications that can determine the structural integrity of the reactor coolant system components directly affected by the pressurization event. Therefore, if the pressure of the reactor coolant system exceeds its ASME Code Service Level B pressure limit, then an inspection program will be established based on an assessment of all potentially affected safety-related reactor coolant system components, and subsequent inspections and/or testing per the appropriate portions of ASME Code, Section XI, Subarticles IWB-2400 and IWB-2500 will be performed and evaluated against the code acceptance criteria, prior to commencement of normal power operations.