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Subject: **Response to Portion of NRC Request for Additional Information Letter Nos. 100 and 101 Related to ESBWR Design Certification Application – Safety Analyses – RAI Numbers 15.0-28, 15.0-29 and 15.3-31**

Enclosure 1 contains GE-Hitachi Nuclear Energy Americas (GEH) response to the subject NRC RAIs transmitted via References 1 and 2.

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 07-327 – Letter from US Nuclear Regulatory Commission (NRC) to Robert E. Brown, *Request for Additional Information Letter No. 100 Related to ESBWR Design Certification Application,* dated May 30, 2007
2. MFN 07-357 – Letter from US Nuclear Regulatory Commission (NRC) to Robert E. Brown, *Request for Additional Information Letter No. 101 Related to ESBWR Design Certification Application,* dated June 21, 2007

Enclosures:

1. Response to NRC Request for Additional Information Letter Nos. 100 and 101 Related to ESBWR Design Certification Application – Safety Analyses, RAI Numbers 15.0-28, 15.0-29 and 15.3-31

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GEH /San Jose (with enclosures)
	RE Brown	GEH /Wilmington (with enclosures)
	eDRF	0075-1788

Enclosure 1

MFN 07-522

Response to Portion of NRC Request for Additional Information Letter Nos. 100 and 101 Related to ESBWR Design Certification Application

Safety Analyses

RAI Numbers 15.0-28, 15.0-29, and 15.3-31

NRC RAI 15.0-28:

Provide technical basis of unavailability of the Isolation Condenser.

In Section 15A.3.10, Stuck Open Relief valve, GE estimates this initiating event frequency by taking credit for the availability of the Isolation Condenser (IC) System for the ESBWR. It is assumed that the probability of the IC being unavailable is less than 0.1. There is no justification for this number in this section. Please provide the technical basis for this number.

GEH Response:

The Isolation Condenser System (ICS) is used to transfer decay and residual heat from the reactor after a reactor isolation event. The ICS limits reactor pressure and temperature within an acceptable range so that safety/relief valve operation is not warranted.

The ICS consists of four, high pressure, independent loops, each containing an Isolation Condenser (IC) that condenses steam on the tube side and transfers heat to water in a large Isolation Condenser/Passive Containment Cooling pool. The ICs are connected by piping to the reactor pressure vessel. The steam side connection between the vessel and the IC is normally open and the condensate line is normally closed. The ICS is placed into operation by opening at least one of the two valves in the condensate line. This causes the condensate accumulated in the system to drain to the reactor, thus causing steam from the reactor to fill the tubes that transfer heat to the cooler pool water.

The high reliability of ICS is due to the redundancy and diversity built into the system. In each loop, the only active components required to operate for system initiation are the two normally closed valves in the condensate line. These valves are redundant and diverse. Furthermore, it is expected that only three out of the four ICS loops need to operate in order to prevent the safety/relief valves from opening.

Due to the simplicity of the system, its redundancy, and its diversity, the unavailability of 0.1 assumed for ICS in Section 15A.3.10 is considered very conservative. To further prove the conservatism of this assumption, quantifications of the conditional probability of ICS failing, given different initiating events, were performed using the ESBWR PRA model for internal events (NEDO-33201, Rev. 2). The conditional probabilities of ICS failing, resulting from these quantifications, are less than 4.0E-04 for any one of the over-pressurization events considered in Section 15A.3.10.

DCD Impact:

No DCD changes will be made in response to this RAI.

NRC RAI 15.0-29:

Discuss inconsistency of data used for the Stuck Open Relief Valve initiating event in ESBWR PRA and Section 15A.3.10.

In Section 15A.3.10, GE provides a best estimate value for the expected frequency of a stuck open SRV in an ESBWR of 3.28E-04/yr. However, the traditional number used for existing BWR plants is about 4.6E-2/yr (NUREG/CR-5750). In addition, the number used in the ESBWR PRA is 2.23E-2/yr (NEDO-33201 Rev.2, Section 2). Please explain why the best estimate ESBWR frequency (i.e., 3.28E-04/yr) was not used in the ESBWR PRA.

GEH Response:

The scope of Section 15A.3.10 is to assess whether the frequency of the Stuck Open Safety Relief Valve (SORV) event is lower than once in 100 years, with enough certainty, so the event does not have to be classified as an Anticipated Operational Occurrence (AOO). Therefore, the statistical data from NUREG/CR-5750 was used to determine an ESBWR-specific frequency of the subject event.

NUREG/CR-5750 estimates the SORV (Category G2) frequency of 4.6E-2/yr based on existing BWR experience. The NUREG/CR-5750 calculation includes Safety Relief Valve (SRV) opening events caused not only by transients, but also spurious openings, and openings due to intentional action during surveillance testing. In the case of the ESBWR, the frequency of SRV openings while at power will be reduced substantially. The main reason for this reduction is that the ESBWR design includes an Isolation Condenser System, which is designed to limit the reactor pressure to a value below the SRV set point, in case of an isolation event. Also, the ESBWR operation will not include surveillance testing requiring opening of SRVs while the reactor is at power. Appendix 15A evaluates spurious SRV openings followed by failure to close in a separate section dedicated to Inadvertent Opening of a Safety Relief Valve (IORV).

Based on the above discussion, the frequency of the SORV event for the ESBWR, as estimated in Section 15A.3.10, is much lower than the frequency for BWR Category G2 of NUREG/CR-5750.

The ESBWR PRA chooses to be conservative in estimating the IORV initiating event frequency by using generic BWR data for SORV. The frequency of 2.23E-2/yr for SORV in BWRs, used in estimating the IORV initiating event for the ESBWR PRA, comes from NUREG/CR-6928.

DCD Impact:

No DCD changes will be made in response to this RAI.

NRC RAI 15.3-31:

The calculation of the event frequency (Section 15A.3.8) assumes 0.0 /pry frequency for the following: incorrect set point or spring adjustment; spring relaxation; and operator error. Is the operator error referring to an error in setting or adjusting the valve spring or some other operator error? Please justify the above values in light of operating experience associated with SRVs of a similar design to ESBWR.

GEH Response:

The operator-error contributor to Inadvertent Opening of a Safety Relief Valve (IORV) discussed in Section 15A.3.8 is the error of commission by an operator using manual controls to open the SRVs. This is explained in Subsection 15A.3.8.2, in the paragraph titled "Operator Error." The operator error in setting or adjusting the valve spring is covered by the "Incorrect setpoint or spring adjustment" failure mode contributing to the IORV event.

The following is the justification of zero-frequencies for the events listed by the subject RAI:

Incorrect Set Point or Spring Adjustment: The value for this contributor was revised to 1.8E-03/yr in Revision 4 of the DCD. This was changed in response to RAI 15.0-23, Item A.

Spring Relaxation: Based on the fact that no evidence was found for this event to have occurred in the operating BWR history, spring relaxation has an insignificant contribution to IORV.

Operator Error: The control system of the power-actuated SRVs of the ESBWR is designed to minimize the possibility of accidental manual actuation.

Manual actuation of the SRVs can be performed from video display units (VDU) in the main control room. Safety-related and nonsafety-related VDUs provide a display format that allows the operator to manually open each SRV independently. Each display utilizes an "arm/fire" configuration that requires at least two deliberate operator actions. Operator use of the "arm" portion of the display causes a plant alarm. Also, Automatic Depressurization System (ADS) can be manually initiated as a system to open all SRVs and DPVs, instead of each valve individually. To perform this action, each safety-related VDU provides a display with an "arm/fire" switch (one per division). If the operator uses any two of the four switches, the ADS sequence seals in, and starts the ADS valve sequencing. This requires at least four deliberate operator actions.

Based on the design described above, it is considered that the probability of inadvertent opening of a SRV due to operator error is insignificant compared to the probability of an IORV due to the other contributors to this event.

DCD Impact:

No DCD changes will be made in response to this RAI.