HITACHI

GE Hitachi Nuclear Energy

Richard E. Kingston
Vice President, ESBWR Licensing

P.O. Box 780
3901 Castle Hayne Road, M/C A-55
Wilmington, NC 28402 USA

T 910.675.6192
F 910.362.6192
rick.kingston@ge.com

MFN 08-564

Docket No. 52-010

July 29, 2008

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

Subject: **Response to NRC Request for Additional Information Letter No. 172 Related to the ESBWR Design Certification – Safety Analyses – RAI Number 15.3-19S01**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) responses to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letter dated March 28, 2008. GEH response to RAI Number 15.3-19S01 is addressed in Enclosure 1.

If you have any questions or require additional information, please contact me.

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN 08-309, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 172 Related To ESBWR Design Certification Application*, dated March 28, 2008

Enclosure:

1. Response to Portion of NRC Request for Additional Information Letter No. 172 Related to ESBWR Design Certification Application – Safety Analyses – RAI Number 15.3-19S01

cc:	AE Cubbage	USNRC (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	eDRF	0000-0075-6217R2

Enclosure 1

MFN 08-564

Response to Portion of NRC Request for Additional Information Letter No. 172 Related to ESBWR Design Certification Application

Saftey Analysis

RAI Number 15.3-19S01

NRC RAI 15.3-19 S01:

SRP Sections 15.4.1, 15.4.2, and 15.4.3 address the reactivity events with a potential for inadvertent criticality.

Please provide results from a consequence analysis for an inadvertent criticality during refueling.

GEH Response:

Because of the specific reasons cited in GEH letter No. MFN 07-655 dated December 12, 2007, the probability of an inadvertent criticality during refueling with the head off of the ESBWR is extremely low. If such an event were to occur, there is a long response period available to insert control rods and achieve shutdown. During that period, the core would not be uncovered due to the depth of water during refueling. The consequences for ESBWR would be less than or comparable to those observed in other reactors which have experienced an inadvertent criticality.

It is assumed that the reactor pressure vessel is uncovered and reactor containment is accessible to the workers at the time of the accident; therefore, worker and public exposure must be considered in addition to fuel integrity issues.

In an actual refueling inadvertent criticality event at another site (see Reference 1), fluctuations in the level of radioactive noble gas were not observed in the exhaust pipe monitors or the monitoring posts. The measurements on the charcoal filters in the exhaust pipes for radioactive iodine and other materials were below the detection limits. The radiation dose was calculated to be 1.3×10^{-8} mSv from direct radiation at the water surface (since the head was removed for refueling), and 3.7×10^{-9} mSv from skyshine radiation. These values are significantly less than the total effective dose equivalent of 1 mSv specified in 10 CFR 20.1301 for individual members of the public during licensed operation. Furthermore, the workers at the site were not seriously exposed to radiation. The pocket dosimeter-based measurements of gamma radiation and the film badge-based measurements of thermal neutron radiation were below the detection limits. Thus it can be assumed that an inadvertent criticality during refueling of the ESBWR would result in a negligible dose to the workers and the public. (Reference 2)

During the same criticality event, the measurement range of the Intermediate Range Monitors was exceeded and neutron flux measurements were not available for approximately eight minutes after the accident. The utility calculated the reactivity insertion based on final control rod positions, the neutron flux measurement when they resumed, and with data from mockup tests. (Reference 2)

The reactor achieved a calculated peak power of approximately 15% with a maximum enthalpy increase of 13 cal/gUO_2 at peak power, and the maximum fuel enthalpy was

calculated to be in the range of 41-49 cal/gUO_2. Neither of these enthalpies challenges the ESBWR analysis limit of 150 cal/gUO_2.

The consequences for the ESBWR fuel would be less than or comparable to those reported above. The fuel used in the ESBWR and the Japanese plant is similar. Particularly, they both have the same Shutdown Margin (SDM) specification such that rods withdrawals in an ESBWR would produce a similar power response to that described above. In addition, both maintain a large shielding water depth during refueling so that a long reaction time is available and the Critical Power Ratio (CPR) margin is similar; i.e. the power level will not damage fuel. So it can be concluded that an inadvertent criticality during refueling of the ESBWR would not result in a threat to the integrity of the fuel.

DCD Impact:

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

References:

1. "Analysis on Criticality "Accident" Occurred at Shika 1 of Hokuriku Electric Power Company," Rev. 0, Japan Nuclear Technology Institute, April 17-2007.
2. "Investigation Report on the Criticality Accident at Shika Nuclear Power Station, Unit 1 of Hokuriku Electric Power Company in 1999 and Other Unexpected Control Rod Withdrawal Events During Plant Shutdowns," Nuclear and Industrial Safety Agency, Japan, April 20, 2007.