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MFN 08-414 Supplement 1

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U.S. Nuclear Regulatory Commission
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Subject: Response to Portion of NRC Request for Additional Information Letter No. 219 Related to the ESBWR Design Certification Application – Nuclear Boiler System - RAI Numbers 5.2-71 S01 and 5.4-59 S01

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 219 dated June 30, 2008 (Reference 1).

Enclosure 1 contains the GEH responses to RAI Numbers 5.2-71 S01 and 5.4-59 S01. Enclosure 2 contains the associated DCD Tier 2 Markup.

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

Sincerely,

Richard E. Kingston
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN 08-560, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 219 Related To ESBWR Design Certification Application*, dated June 30, 2008

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 219 Related to ESBWR Design Certification Application – Nuclear Boiler System- RAI Numbers 5.2-71 S01 and 5.4-59 S01
2. Response to Portion of NRC Request for Additional Information Letter No. 219 Related to ESBWR Design Certification Application – Nuclear Boiler System- RAI Numbers 5.2-71 S01 and 5.4-59 S01 – DCD Markups

cc:		
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	eDRF	0000-0089-0145 - RAI 5.2-71 S01
		0000-0090-5495 - RAI 5.4-59 S01

Enclosure 1

MFN 08-414 Supplement 1

Response to Portion of NRC Request for Additional Information Letter No. 219 Related to ESBWR Design Certification Application

Nuclear Boiler System

RAI Numbers 5.2-71 S01 and 5.4-59 S01

NRC RAI 5.2-71 S01:

In response to RAI 5.2-71, the applicant modified DCD Sections 4.5.2 and 5.2.3 to include fabrication steps taken to minimize the effects of cold working when fabricating components made from stainless steels and nickel based alloys. In DCD Revision 5, Section 4.5.2.2, the applicant indicated that back purging is not required where the backside is fully accessible for cleaning and liquid penetrant examination. The staff requests that the applicant clarify if it intends to provide a back purge for all full penetration welds exposed to reactor coolant. If the response is yes, the staff requests that the applicant modify the DCD to state that all full penetration welds exposed to reactor coolant will receive a back purge. If the response is no, the staff requests that the applicant explain why back purging is not necessary given that this would result in unnecessary grinding and cold working of weld surfaces exposed to reactor coolant.

GEH Response:

GEH will require a back purge for all full penetration welds exposed to reactor coolant, and will incorporate the requirement into Subsection 4.5.2.2.

DCD Impact:

DCD Tier 2, Subsection 4.5.2.2 will be revised as noted on the attached markup.

NRC RAI 5.4.59 S01:

The staff reviewed GEH's response to 5.4-59. The staff needs additional information as described below to understand the capability of the RWCU/SDC to remove decay heat in Modes 4, 5, and 6, and its impact on the ESBWR shutdown PRA.

(a) Please explain in the DCD and in TS the required minimum vessel level (elevation based on RPV bottom) to ensure that RWCU/SDC is capable of removing decay heat in all modes of operation. GEH SIL 357 recommends raising water level to a point above the bottom of the pre-dryers on the steam separators. GEH SIL 357 further states that the separator predryers are located slightly higher than the lowest fluid connection between the inside and the outside of the shroud. In contrast, the GEH RAI response and DCD Rev 5 states that water level should be maintained above the first stage water spill of the steam separators. Are these the same points? The staff understands that if water level is not high enough to promote natural circulation, thermal stratification of the vessel will occur with surface coolant temperatures increasing above 212 °F while RWCU/SDC temperature indication will remain below 212 °F. Inadvertent reactor

heatup and repressurization has occurred in operating BWRs plants due to inadequate vessel circulation.
(b) Based on GEH RAI response, RWCU/SDC functionality depends on mixing. GEH is requested to provide or make available for audit the mixing calculation so that the staff can better understand the mixing in the vessel and its potential impact on the RWCU/SDC function.

(c) With the RWCU/SDC removing decay heat, if vessel level were to drop below the first stage of the water spill of the steam separators, GEH is requested to clarify if the ESBWR design has any incore thermo-couples or other indication that can monitor core temperature.

(d) If cooling water is lost to the RWCU/SDC heat exchangers, GEH is requested to include in the DCD if there are temperature limits on the RWCU/SDC pumps. This information could impact the operator's ability to recover the RWCU/SDC system after a sustained loss of RWCU/SDC support systems such as PSW.

(e) As vessel level is being raised to remove the vessel head, it appears that the vessel temperatures may stratify with a relatively hot layer of water at the top of the vessel and a cold layer near the RWCU/SDC suction piping. Please provide a calculation demonstrating that the RWCU/SDC is capable of removing decay heat at the minimum vessel level required for reactor vessel head removal. Please document this vessel level in the DCD. It appears that even if no stratification occurs at higher vessel levels, there is the potential for inadequately mixed (non-homogenous) fluid entering into the suction piping of the RWCU/SDC which could lead to piping failure from thermal cycling fatigue.

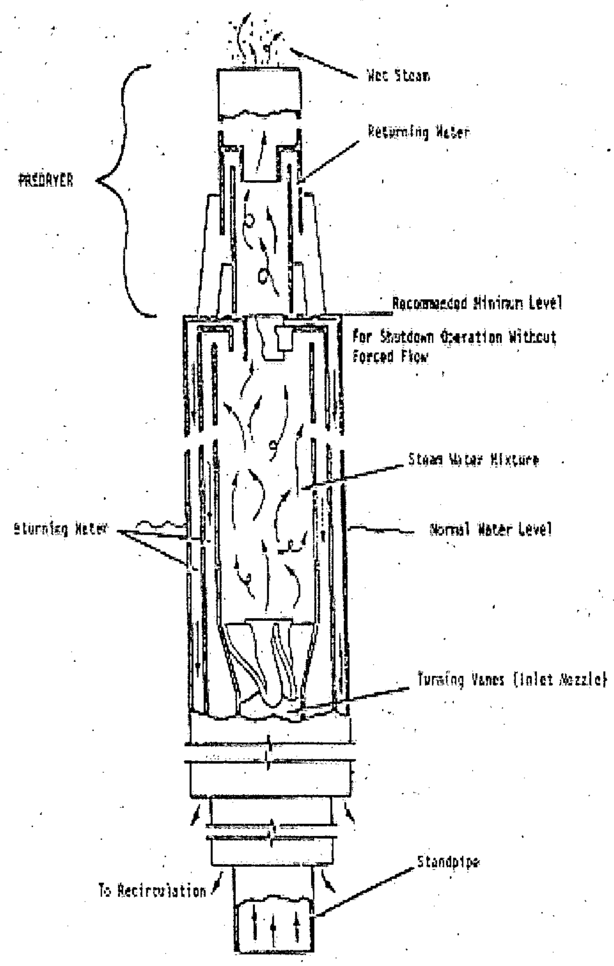
These phenomena caused a similar failure to occur at the Civaux nuclear power plant in France that had a rupture of the RHR piping in May 1998 that was 180mm in length. Please discuss in the DCD how this potential for RWCU/SDC piping failure is to be minimized acknowledging that the ESBWR shutdown CDF is dominated by LOCA events.

GEH Response:

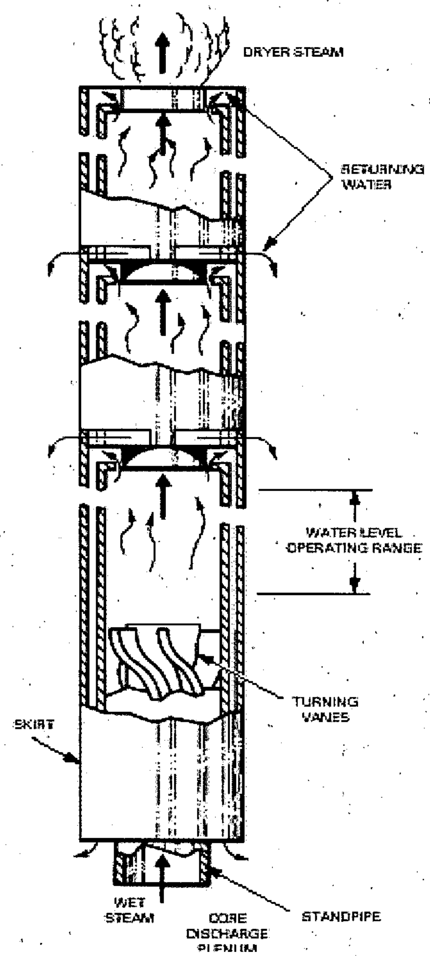
The following addresses each question

(a) Reactor vessel level is maintained at or above the separator first-stage water spillover elevation, as stated in the RAI 5.4-59 response, dated May 12, 2008. This analytical lower limit assures the reactor water cleanup shutdown cooling (RWCU/SDC) system is capable of removing decay heat in all modes of operation. The change to the DCD noted in the initial response is incorporated into DCD Tier 2, Revision 5. The first-stage spillover is the lowest level path by which water rising from the core returns to the annulus region of the vessel to maintain continuous core circulation.

The SIL 357 "bottom of the pre-dryers" water level is an administrative water level lower limit recommended to maintain shutdown cooling function in BWR series 2 through 6 designs. The SIL provides a figure depiction of a typical separator column with the upper portion identified as the pre-dryer. A similar figure schematically depicts the conceptual ESBWR steam separator column and shows that it is slightly different from the SIL-357 schematic figure.



SIL357 Typical Steam Separator



ESBWR Steam Separator

The ESBWR separator column depiction shows three elevations or stages of condensate return. The RAI 5.4-59 initial response refers to the lowest of these three stages, whereas the ESBWR equivalent of the SIL-357 administrative limit recommendation might be to maintain level at or above the second (middle) stage spillover elevation. The effect of this increased level on core circulation would be minimal since the ratio of vertical water column weight per unit area inside the core shroud compared to outside the core shroud would not be significantly changed, and would tend to favor increased circulation rate. Thus, SIL-357 represents operating experience that does apply to the ESBWR design in that an administrative lower water level limit should be established to effect a margin between the operating range for shutdown cooling mode and the analytical lower limit of the design. For the ESBWR, a determination of the recommended administrative vessel water level limits for operation of the RWCU/SDC system, during the shutdown cooling mode, are to be developed as part of human factors engineering (HFE) described in DCD Tier 2 Chapter 18.

The staff's understanding that vessel water level below the analytical lower limit of the separator column first-stage spillover elevation would result in thermal stratification is correct. Shutdown cooling core circulation would cease in this situation with the consequent onset of liquid volume heatup inside the shroud, chimney and separator column regions of the vessel. However, such a condition also represents a departure from normal operation to which the licensed unit's operating staff must be prepared to respond. This preparation is within the scope of emergency procedures that are discussed in DCD Tier 2 Section 13.5, and is not a part of the ESBWR design scope.

(b) In the initial response to RAI 5.4-59 it is assumed that the hot core flow and the cold SDC injection flow are completely mixed before returning back to the inlet of the heat exchanger. The complete mixing equation used in response to RAI 5.4-59 is:

$$\frac{((m_1 + m_2) \cdot T_{hx-out} + m \cdot T_{core})}{(m_1 + m_2) + m} = T_{mean} = T_{suc1}$$

The nomenclature used in above equation is the same as in the response to RAI 5.4-59, Figure 5.4-59-1. The assumption of complete mixing is conservative for designing the SDC/RWCU non-regenerative heat exchangers. The affect of no mixing is discussed in response to Item (e) of this supplement below.

(c) There are no core exit thermocouples in a typical BWR design and none on the chimney or separator portions of the ESBWR design. As noted in response to item (a) of this supplemental request, a postulated event in which vessel water level falls below the separator column first-stage spillover elevation during shutdown cooling operation (which implies the reactor vessel head is installed) is a condition of departure from normal unit operations. If in this postulated event the core begins to boil, steam would be detected by pressure instrumentation and by the temperature sensing elements in the isolation condenser system (ICS) steam supply lines, assuming that event response procedures make use of the ICS to remove decay heat and limit vessel pressurization.

As noted in response (a), emergency procedures are not a part of ESBWR design scope as stated in DCD Tier 2 Section 13.5.

(d) The RWCU/SDC design, described in DCD Tier 2 Subsection 5.4.8, requires that the pumps be operable at nuclear boiler rated temperature and pressure conditions, even though in all modes of intended operation the water is cooled before reaching pump suction. The pump design intended for this system will be a canned-type, based on ABWR experience, and does not use a packed shaft sealing gland or mechanical shaft seal. Therefore, the pump design requirement is to function with water at any temperature over the span of the nuclear boiler system operating range. Loss of decay heat removal by the RWCU/SDC non-regenerative heat exchanger may be due to loss of service water indirectly or loss of reactor closed-cooling (RCC) water directly, and actions for recovery of this cooling resource are not limited by the pump design.

(e) An additional calculation is not required. The cooling flow path remains the same as evaluated for the RAI 5.4-59 initial response, even with vessel water level being raised. Because the minimum analytical level for the shutdown cooling mode requires the vessel level be above the separator skirt, a pocket of relatively warmer water will remain at the top of separator assembly and in the annulus above the mixing zone (i.e., the region below the separator skirt external to the separator standpipes and above the chimney head where circulating reactor coolant mixes with incoming cooling flow from the feedwater distribution header). This pocket of stagnant water is warmer by only a few degrees, and will be subject to continued heating only by contact with other rising warm coolant from the core, and to heat losses through the vessel wall and head as the reactor vessel cools. As indicated in response to this supplemental request (a) above, the recommended administrative control range for vessel water level, during the shutdown cooling mode of operation, will be determined by the HFE process as described in DCD Tier 2 Chapter 18.

When the vessel head is lifted, the water level is raised to flood over the dryer assembly in order to maintain radiation shielding and contamination control. Some overflow from the dryer may return through the moisture drains to the annulus. However, the total natural circulation reactor coolant flow rate is not expected to be significantly altered by the level increase as explained above and for the reasons provided in response to Item (a).

Because the minimum shutdown cooling mode vessel level is maintained above the feedwater distribution header, any significant temperature change due to the stagnant pocket of warmer water remains well above the mid-vessel nozzles that serve as the supply connections for RWCU/SDC system flow in shutdown cooling mode. Mixing of the warm and cool liquid flows in the mixing zone is dependent on temperature difference of the two flows and the relative fluid momentum of the cool flow. The response to RAI 5.4-59 used a maximum calculated temperature differential between hot (core outlet) and cold (SDC return) flows of approximately 21 °C. The affect of this temperature differential has been considered against the design fatigue curve (ASME Code, Section III, Appendix 1, Figure 1.9-1) and determined to be much lower than the endurance limit. Thus, by limiting the temperature differential in the design, there will be not be a thermal fatigue problem of the vessel nozzle or piping material for the mid-vessel suction nozzles and connected piping due to SDC operation.

DCD Impact:

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

Enclosure 2

MFN 08-414 Supplement 1

Response to Portion of NRC Request for Additional Information Letter No. 219 Related to ESBWR Design Certification Application

Nuclear Boiler System

RAI Numbers 5.2-71 S01 and 5.4-59 S01

DCD Markup

- Protective gas back purging shall be used for full penetration single side welded joints of stainless steel until a minimum of 4.8 mm (3/16 inch) of weld thickness is completed. Before welding, joints shall be back purged. ~~This back purge is not required where the backside is fully accessible for cleaning and liquid penetrant examination, and when the cleaning and liquid penetrant examination are performed.~~
- The SMAW process shall not be used on the root pass.
- Each welding pass shall be visually examined for defects and bead shape prior to applying the next pass. The weld bead and the surfaces heated during welding shall be cleaned between passes to remove oxides. Porosity, slag, lack of fusion, cracks, and other defects shall be removed by grinding, machining, or chipping prior to depositing subsequent passes. Removal of cracks, other than crater cracks, shall be verified by a liquid penetrant examination.
- Grinding shall be controlled per Subsection 4.5.2.4.

Core support structures are fabricated in accordance with requirements of ASME Code Section III, Subsection NG-4000, and the examination and acceptance criteria shown in NG-5000. The internals, other than the core support structures meet the requirements of the industry standards, for example, ASME or American Welding Society (AWS), as applicable. ASME Boiler & Pressure Vessel (B&PV) Code Section IX qualification requirements are followed in fabrication of core support structures. All welds are made with controlled weld heat input.

Electroslag welding is not applied for structural welds.

ESBWR fully complies with Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," including application of the following provisions to all stainless steel weld filler metal applied to reactor internal components: The delta ferrite content for weld materials used in welding austenitic stainless steel assemblies is verified on undiluted weld deposits for each heat or lot of filler metal and electrodes. The delta ferrite content is defined for weld materials as 5.0 Ferrite Number (FN) minimum, 8.0 FN average and 20 FN maximum for 308 L and 16 FN for 316 L. Ferrite content is determined by use of magnetic instruments calibrated according to AWS A4.2. This ferrite content is considered adequate to prevent any micro-fissuring (Hot Cracking) in austenitic stainless steel welds. This procedure complies with the requirements of Regulatory Guide 1.31.

4.5.2.3 Non-Destructive Examination

Materials for core support structures are in conformance with and certified to ASME Section III, Subsection NG. Examination of materials (examination methods and acceptance criteria) is specified in NG-2500. Examination methods and acceptance criteria for core support structure weld edge preparations and welds are provided in NG-5000. Tubular products that are pressure boundary components (CRD and in-core housings) are examined as outlined in Subsection 5.2.3.3.3. For non-ASME Code reactor internal structures and associated welds, examinations are established based on relevant design and analysis information, and take guidance from NG-2500 and NG-5000 respectively.