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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 53 – Related to ESBWR Design Certification Application – DCD Chapter 4– RAI Numbers 4.4-3, 4.4-4, 4.4-18, 4.4-21, 4.4-22, 4.4-23, 4.4-40, 4.4-41**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter.

If you have any questions about the information provided here, please let me know.

Sincerely,

Kathy Sedney for

David H. Hinds
Manager, ESBWR

D068

Reference:

1. MFN 06-288, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application*, April 16, 2006

Enclosure:

1. MFN 06-399 – Response to Portion of NRC Request for Additional Information Letter No. 53 – Related to ESBWR Design Certification Application – DCD Chapter 4– RAI Numbers 4.4-3, 4.4-4, 4.4-18, 4.4-21, 4.4-22, 4.4-23, 4.4-40, 4.4-41

cc:	AE Cubbage	USNRC (with enclosures)
	GB Straback	GE/San Jose (with enclosures)
	eDRFs	59-6796, 59-5446, 59-5427, 59-3547,

ENCLOSURE 1

MFN 06-399

Response to Portion of NRC Request for

Additional Information Letter No. 53

Related to ESBWR Design Certification Application

DCD Chapter 4

RAI Numbers 4.4-3, 4.4-4, 4.4-18, 4.4-21, 4.4-22,

4.4-23, 4.4-40, 4.4 41

NRC RAI 4.4-3

DCD Tier 2, Section 4.4.1.5 discusses the fuel heat transfer bases. Provide additional detailed description of the heat transfer between the coolant and the fuel rod surface. Address why GE's current (conventional) heat transfer model remains valid for the ESBWR.

GE Response

Two heat transfer regimes, namely, the single-phase convective heat transfer and the nucleate boiling, are important between the coolant and fuel rod surfaces during normal operation and anticipated operational occurrences (AOOs) of a BWR and/or an ESBWR. GE uses standard, well-accepted heat transfer correlations for both of these regimes. Specifically, GE uses the Dittus-Boelter correlation (Reference 4.4-3-1) for the single-phase convective heat transfer for both fuel design (in the core simulator code) and systems analyses (in TRACG code). For nucleate boiling, GE uses the Jens-Lottes correlation (Reference 4.4-3-2) for fuel design (in the core simulator code) and the Chen correlation (Reference 4.4-3-3) for systems analyses (in TRACG). All of these three correlations are valid in rod bundles as stated in Section 4.3 (Multirod Flow Boiling) of Reference 4.4-3-4.

The ranges of applicability of the above three heat transfer correlations are discussed below:

a) Dittus-Boelter Correlation

For moderate temperature differences, the correlation is valid for flows within the following ranges (Reference 4.4-3-5):

$0.7 < Pr < 160$
$Re > 6000$
$L/D > 60$

The coolant Prandtl number (Pr) and Reynolds number (Re) for both BWR and ESBWR applications are within the above valid range of the correlation. During the development of the correlation, the entrance effect (L/D) was neglected. Since the entrance effect is small for turbulent flow, this omission of entrance effect in the correlation and its application in the GE codes do not have any significant impact.

b) Jens-Lottes Correlation

The correlation was developed from experiments on subcooled boiling of water flowing upwards under the following operating ranges (Reference 4.4-3-6):

System pressure: 0.7 to 17.2 MPa
Temperature: 115° C to 340° C
Mass-flux: 11 to 10,500 kg/m^2-s
Heat Flux: up to 12,500 kW/m^2

The operating ranges of both BWR and ESBWR are within the above valid ranges of the Jens-Lottes correlation.

c) Chen Correlation

Chen combined the effects of both nucleate boiling and forced convection vaporization heat transfer and developed his correlation based on the water and organic liquid data in the following operating ranges (Reference 4.4-3-6):

> System pressure: 0.1 to 3.48 MPa
> (The pressure range extended to 6.9 MPa as per Reference 4.4-3-7)
> Liquid inlet velocity: 0.06 to 4.5 m/s (Mass Flux: 54 to 4070 kg/m^2-s)
> Quality: 1 to 71%
> Heat Flux: 6.2 to 2400 kW/m^2

Again, the operating ranges of both BWR and ESBWR are within the above valid ranges of the Chen correlation.

On the issue of liquid velocity or mass-flux through an ESBWR fuel bundle vs. that through the fuel bundle of an operating BWR, Table 4.4-1a of the DCD Tier 2, Rev. 1 shows that the core average inlet velocity for the ESBWR is about 50 to 60% of that in a BWR/6 or ABWR. This translates to an average mass-flux of about 800 kg/m^2-s for the ESBWR. From the same Table 4.4-1a, one may calculate the average heat flux for ESBWR as 451 kW/m^2. Allowing for differences in bundle power, flow rate and local peaking, one can conclude that all three heat transfer correlations discussed above are also valid for the ESBWR conditions, thus conforming to the guidance of NRC Standard Review Plan (NUREG-0800) Section 4.4. Furthermore, the difference in the wall temperatures calculated by using the Jens-Lottes and the Chen correlations is minor since both predict very high heat transfer coefficients, as expected, for nucleate boiling.

Although there is no compelling reason to revise the DCD based on the above response, GE proposes to add the following at the end of DCD Tier 2, Subsection 4.4.1.5 to amplify the subsection:

> Standard and well-accepted heat transfer correlations between the coolant and the rod surfaces are used. Further details are given in Subsection 4.4.2.5.

References:

4.4-3-1 F. W. Dittus and L. M. K. Boelter, University of California (Berkeley) Publication, Engineering, Vol. 2, No. 13, pp 443-461, 1930. Reprinted in Int. Communication Heat Mass Transfer, Vol. 12, pp. 3-22, 1985.

4.4-3-2 W. H. Jens and P. A. Lottes, "Analysis of Heat Transfer, Burnout, Pressure Drop and Density Data for High Pressure Water," USAEC Report ANL-4627, 1951.

4.4-3-3 J. C. Chen, "A Correlation for Boiling Heat Transfer to Saturated Fluids in Convective Flow," ASME Preprint 63-HT-34, Presented at 6th National Heat Transfer Conference, Boston, 1963.

4.4-3-4 R. T. Lahey, Jr. and F. J. Moody, "The Thermal-Hydraulics of a Boiling Water Nuclear Reactor," Second Edition, Section 4.3, American Nuclear Society, La Grange Park, IL, 1993.

4.4-3-5 F. Kreith, "Principles of Heat Transfer," 3rd Edition, Harper and Row, New York, 1973.

4.4-3-6 J. G. Collier and J. R. Thome, "Convective Boiling and Condensation," Third Edition, Chapters 5 and 7, Clarendon Press, Oxford, 1994.

4.4-3-7 D. R. Liles, et al., "TRAC-PF1/MOD1 Correlations and Models," NUREG/CR-5069, LA-11208-MS, 1988.

NRC RAI 4.4-4

DCD Tier 2, Section 4.4.1.7 provides a summary of the design bases. Discuss the evaluation of steady-state MCPR and MLHGR limits for the most severe AOO, including assumptions, methods, and results.

GE Response

Section 15.2 of the DCD, Tier 2, Rev. 1 describes the results of all Anticipated Operational Occurrences (AOOs) for the ESBWR. Table 15.2-5 in that section presents the summary results including the maximum neutron flux, maximum vessel bottom pressure, maximum core average surface heat flux, and ΔCPR (maximum decrease in minimum critical power ratio or MCPR) for all AOOs analyzed quantitatively. Based on these results, Subsection 15.2.6 of the DCD, Tier 2, Rev. 1 lists the limiting events that establish the CPR operating limit as:

- Limiting pressurization event: MSIV isolation or Closure of All MSIVs
- Limiting decrease in core coolant temperature event: Loss of feedwater heating with SCRRI (Selected Control Rods Run-In) actuation.

Subsections 15.2.2.7 and 15.2.1.1 of the DCD, Tier 2 provide further details on the sequence of events, assumptions, methods and the results of the above two limiting AOOs, respectively.

Subsection 15.2.6 of the DCD also states (after correcting the 4.4 Subsection numbers):

"For the core loading in Figure 4.3-1, the resulting initial core MCPR operating limit is 1.30, using the methodologies listed in Subsections 4.4.3.1.3 and 4.4.2.1.3. The operating limit based on the initial core design will be provided by the operating license holder."

(Note: The correct Subsections 4.4.3.1.3 and 4.4.2.1.3 were mistyped as 4.4.3.1.1.3 and 4.4.1.2.3, respectively, in Subsection 15.2.6 of the DCD, Tier 2, Rev. 1. These will be corrected in the next revision of DCD, Tier 2.)

Based on the above GE response, Subsection 4.4.1.7 of the DCD, Tier 2 is proposed to be revised, in its entirety, as:

> The steady-state operating limits have been established to assure that the design bases are satisfied for the most severe AOO, discussed in Section 15.2. The effects of the limiting AOO do not result in any violation of the acceptance criteria set forth in Subsection 15.0.3.1, for which the fuel, the reactor pressure vessel or the containment are designed. Therefore, these barriers maintain their integrity and function as designed.

NRC RAI 4.4-18

DCD Tier 2, Section 4.4.2.5 references a topical report which has not yet received NRC approval. Discuss the applicability of the topical report to the ESBWR design.

GE Response:

The purpose of referring to the "Licensing Topical Report TRACG Model Description," NEDE-32176P is to guide a reader to the heat transfer and related models used in the GE systems analysis code, TRACG. As described in the GE response to NRC RAI 4.4-3, TRACG uses the standard and well-accepted Dittus-Boelter and Chen correlations for the single-phase convective regime and the nucleate boiling regime, respectively. It has also been shown in that response that these correlations are valid for the ESBWR conditions.

Although the above-referenced TRACG model description LTR (Licensing Topical Report) has not yet received NRC approval, the LTR on "TRACG Application for ESBWR," NEDE-33083P-A (Reference 4.4-9 of DCD Tier 2 Section 4.4) has been approved by NRC for the ESBWR LOCA application. Similar approval of TRACG for the ESBWR AOOs is part of the ESBWR Design Certificate Application.

To complete the information on heat transfer methods used in the fuel design, GE proposes to revise the second paragraph of DCD Tier 2, Subsection 4.4.2.5, in its entirety, as:

> The Jens-Lottes (Reference 4.4-7) heat transfer correlation is used in fuel design to determine the cladding-to-coolant heat transfer coefficients for nucleate boiling. For the single-phase convective or liquid region, the well-established Dittus-Boelter correlation is used. The methodology for fuel cladding, gap and pellet heat transfer is described in Section 4.2.

GE also proposes to correct the typing mistakes in Reference 4.4-7 in DCD Tier 2, Subsection 4.4.8. So the entire Reference 4.4-7 will be revised as:

> 4.4-7 W. H. Jens and P. A. Lottes, "Analysis of Heat Transfer, Burnout, Pressure Drop and Density Data for High Pressure Water," USAEC Report ANL-4627, 1951.

NRC RAI 4.4-21

NUREG-0800, Standard Review Plan, Section 4.4 (Draft Rev. 2 - April 1996), Item I (Areas of Review) includes a review of the functional performance and requirements for the Inadequate Core Cooling (ICC) monitoring system hardware. Provide a reference in Section 4.4 to the appropriate section(s) of the DCD which address the ICC system.

GE Response:

The issue of Inadequate Core Cooling (ICC) monitoring system has been discussed in Appendix 1A (Response to TMI Related Matters) of ESBWR DCD, Tier 2, Rev. 1. Specifically, TMI Item II.F.2 in Table 1A-1 (TMI Action Plan Items) addresses this issue related to the ESBWR and the resolution statement from that Table is reiterated below:

"The detection of conditions indicative of inadequate core cooling is provided in the ESBWR design by the direct water level instrumentation system. Coolant level in the RPV is measured by both wide range and fuel zone instruments. The four divisions of wide range instruments cover the range from above the core to the main steam lines. The four channels of fuel zone instruments cover the range from below the core to the top of the steam separator. The RPV water level is the primary variable indicating the availability of adequate core cooling. Indication of water level by the differential pressure method is acceptable (without diverse methods of sensing and indication) because adequate redundancy and unambiguity is provided from the bottom of the core support plate to the centerline of the main steam lines. The ESBWR has addressed the issue regarding erroneously high water level indication upon vessel depressurization due to the release of dissolved non-condensable gases in the reference leg. The ESBWR water level instrumentation system design includes a constant metered addition of purge water from the CRD hydraulic system to prevent the build-up of dissolved gasses in the fixed leg. This is consistent with the approved ABWR design as well as the modifications made by the majority of the BWR fleet."

In view of above, GE proposes to add the following Subsection 4.4.4.7 in the DCD, Tier 2:

4.4.4.7 Inadequate Core Cooling (ICC) Monitoring System

The issue of Inadequate Core Cooling (ICC) monitoring system has been discussed in Appendix 1A (Response to TMI Related Matters) of this ESBWR DCD, Tier 2. Specifically, the TMI Item II.F.2 in Table 1A-1 (TMI Action Plan Items) addresses this issue related to the ESBWR.

NRC RAI 4.4-22

NUREG-0800, Standard Review Plan, Section 4.4 (Draft Rev. 2 - April 1996), Item I (Areas of Review) includes a review of the technical specifications (TS) regarding safety limits and limiting safety system settings. Provide a reference in Section 4.4 to the applicable core thermal-hydraulic technical specifications and bases.

GE Response:

There are two Chapters, namely, 16 (Technical Specifications) and 16B (Bases) in the ESBWR DCD, Tier 2, Rev. 1, which describe all ESBWR technical specifications and bases including those applicable to the core thermal-hydraulics. The specific sections that contain information on the safety limits and limiting safety systems settings related to the core thermal-hydraulics are as follows:

- Sections 2.0 (in Chapter 16) and B 2.0 (in Chapter 16B) on Safety Limits (SLs) and associated Bases.
- Sections 3.2 (in Chapter 16) and B 3.2 (in Chapter 16B) on Power Distribution Limits and associated Bases.
- Sections 3.3 (in Chapter 16) and B 3.3 (in Chapter 16B) on Instrumentation and associated Bases.
- Sections 3.4 (in Chapter 16) and B 3.4 (in Chapter 16B) on Reactor Coolant System (RCS) and associated Bases.

In view of above, GE does not consider it necessary to provide any specific reference to the applicable core thermal-hydraulic technical specifications and bases in Section 4.4 of the DCD. Moreover, such additional reference would deviate from the present format of the DCD and might complicate the numbering scheme of the DCD.

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

NRC RAI 4.4-23

DCD Tier 2, Section 4.4.1.4 discusses the pressure drops and flow distributions in the fuel channels and core bypass regions.

(a) *Provide a quantitative comparison to those of conventional BWRs for both fuel channels and core bypass regions.*

b) *What is the impact on the MCPR limit as flow is reduced, or what reduction in flow would result in impacting the MCPR limit and by how much?*

GE Response:

Table 4.4-23-1 shows a comparison of the core pressure drop, bypass flow and the fuel channel flow characteristics of the ESBWR to a BWR6 and a BWR4 plant (Both are conventional jet pump forced flow BWRs). The data is taken from the core simulator at, or near, End-Of-Cycle (EOC) exposure. The cores are comprised of GE14 fuel. The BWR 4 core has four GNF 2 lead use channels. Please note that the BWR 6 plant shown here is not the same plant or at the same conditions as shown in DCD Tier 2, Table 4.4-1.

In the core simulator the pressure drop and flow distribution are calculated in the same manner as is calculated for conventional plants. The core flow rate is an input. The difference is that the core flow rate for the ESBWR is calculated by TRACG where as, in conventional plants, the core flow is set by design and the recirculation system provides the desired flow rate.

Table 4.4-23-1 ESBWR to BWR Comparison

Plant Type – All are at Rated Power with a GE14 core at EOC	Core ΔP (Pa)	Bypass Flow (% of Core Flow)	In-Channel Average Flow (kg/s)	Standard Deviation for Channel Flow distribution (% of average)
ESBWR, Nominal Flow, 1132 Channels	53.2	15.0	7.43	11.1
BWR 4, Low Flow, 764 Channels	125.2	17.0	11.62	16.1
BWR 4, Rated Flow, 764 Channels	160.9	16.9	14.05	15.8
BWR 6, Low Flow, 748 Channels	126.1	16.9	11.79	14.5

An expected flow range has been established for the ESBWR (See DCD Tier 2, Table 4.4-1). This range is based on Monte Carlo analysis described in Licensing Topical Report: NEDE-33083P Supplement 1, "TRACG Application for ESBWR Stability Analysis," December 2004. Within this range, the uncertainties are accounted for in the safety limit and ΔCPR statistical uncertainty analysis as discussed in DCD Tier 2, Subsections 4.4.2.1.2 and 4.4.2.1.3. Core flow outside of this range is not expected.

No DCD Tier 2 changes will be made in response to this RAI.

NRC RAI 4.4-40

Section 1.5.4 of NEDC-33239P discusses the bypass region calculation. The temperature of water in the vessel annulus (downcomer) region is assumed equal to the core inlet temperature. Discuss the conservatism of this assumption considering the variation in downcomer temperature with height.

GE Response:

TRACG analysis at rated conditions shows a temperature difference of approximately 0.1°K from the downcomer level just below the feedwater sparger (~16m above vessel zero) to the core inlet (~4m above vessel zero). The assumption is a valid simplification of expected conditions. There is no appreciable conservatism in the assumption.

No DCD Tier 2 or NEDC-33239P changes will be made in response to this RAI.

NRC RAI 4.4-41

Section 1.5 of NEDC-33239P states that the energy contribution from heat generation in the core shroud and upper and lower core structures is ignored. Explain why this is conservative, considering the neutron and gamma absorption heating in these structures.

GE Response:

Any heat absorbed in the metal structures surrounding the core is heat that did not go directly to the channel or bypass coolant (core coolant). Most of this heat would end up back in the core coolant via conduction to the bypass, the downcomer or the lower plenum. Some of this heat in the upper core structures would be transferred to the upper plenum without heating the core coolant. The second part of the statement in Section 1.5.4 is "This is compensated by neglecting energy losses due to neutron and gamma leakage from the core." Some of the leakage would leave the vessel entirely without heating the core coolant. These two assumptions ensure that all heat generated by the core goes directly to heating the core coolant. The net result is that slightly more heat goes into the core coolant than would be expected. Because the amount of neutron and gamma heating in the structures and the core leakage is expected to be very small compared to the total heat generated, it is probably more appropriate to consider this a simplifying assumption rather than a conservatism.

No DCD Tier 2 or NEDC-33239P changes will be made in response to this RAI.