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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. 126 Related to ESBWR Design Certification Application, RAI Number 14.3-189**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letter dated December 20, 2007 (Reference 1). The GEH response to RAI Number 14.3-189 is addressed in Enclosure 1.

Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markup(s) may not be fully developed and approved for inclusion in DCD Revision 5.

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

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN 07-718, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, *Request for Additional Information Letter No. 126 Related to ESBWR Design Certification Application,* December 20, 2007

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 126 Related to ESBWR Design Certification Application, RAI Number 14.3-189

cc:	AE Cubbage	USNRC (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	DH Hinds	GEH/Wilmington (with enclosure)
	eDRF	0000-0084-4260 – RAI 14.3-189

Enclosure 1

MFN 08-086 Supplement 29

*Response to Portion of NRC Request for Additional Information Letter No. 126 Related to ESBWR Design Certification Application

RAI Number 14.3-189

*Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markup(s) may not be fully developed and approved for inclusion in DCD Revision 5.

NRC RAI 14.3-189

NRC Summary:

ITAAC verification of key TRACG analysis input assumptions

NRC Full Text:

Please explain how certain key TRACG analysis assumptions listed below will be verified to be consistent with the as-built facility through the ITAAC.

(A) The pressure loss coefficients for the following components:

1. Steam Separator

2. Fuel bundle

3. Fuel support orifice

4. Control rod guide tubes

5. Shroud support guide tube

(B) Free Volumes for the following components:

1. RPV

2. Downcomer

3. Core

4. Chimney

5. Separator/dryer

(C) Hydraulic Diameter and Heated Diameter of the core

GEH Response

Note on Item (A)5: GEH clarified in a teleconference 2/21/2008 that the NRC meant "Shroud support" and not "Shroud support guide tube" which is not an ESBWR component.

Note on Item (C): GEH clarified in a teleconference on 4/11/2008 that "Hydraulic and Heated Diameter of the core" would be replaced with "Hydraulic diameter, geometry of heated surfaces, and flow area in fuel assemblies."

GEH agrees that the key TRACG analysis assumptions listed in RAI 14.3-189 will be verified in the ITAAC. The DCD Tier 1 will be revised to add the items to Section 2.1.2 and Table 2.1.2-3. Tier 2 Section 4.4.2.3 will be revised to add a clear definition of Hydraulic Diameter to DCD Tier 2.

DCD Impact

DCD Tier 1 Section 2.1.2 and Table 2.1.2-3 will be revised to added items (30) through (32) as noted on the attached markup. Tier 2 Section 4.4.2.3 will be revised to add a clear definition of Hydraulic Diameter to DCD Tier 2 as noted on the attached markup.

been developed for all high and some medium ranked model parameters. The model uncertainties are documented in Reference 4.4-9. The ΔCPR/ICPR is calculated in accordance with Reference 4.4-9.

4.4.2.1.4 MCPR Safety Limit Calculation Method

The MCPR Safety Limit is calculated in accordance with Section 5.14 of Reference 4.4-12.

4.4.2.2 Void Fraction Distribution Methods

The empirical correlations used for the calculation of the void fraction are the GE void fraction correlation that is used in the 3D core simulator and steady state thermal hydraulic calculations and the correlations for the interfacial shear that is used in TRACG. The GE void fraction model is described in Reference 4.4-15, and details on the qualification are contained in Attachment A to Reference 4.4-13. The TRACG void fraction model is described in Reference 4.4-10 and details on the qualification are contained in Reference 4.4-11.

4.4.2.3 Core Pressure Drop and Hydraulic Loads Methods

The total bundle pressure drop is defined as the sum of four components: friction, elevation, acceleration, and local losses. In these models, the bundle is also divided into control volumes over which the four components of total pressure drop are evaluated separately, thus allowing to capture the effects on pressure drop of axially variable geometry parameters such as flow area, hydraulic diameter, wetted/heated perimeters, heat flux, and spacer elevations. The hydraulic diameter is defined as four times the axial flow area divided by the wetted perimeter, which includes the fuel rod, channel inner wall, and water rod perimeters. The geometry of heated surfaces consists of the number of fuel rods and the fuel rod diameter in a fuel assembly. For fuel assembly types with partial length rods, the number of partial length rods and the partial length rod length(s) are also accounted for in the definition of fuel assembly hydraulic diameter.

The TRACG methods for core pressure drop modeling are described in Reference 4.4-10. The TRACG hydraulic formulation for core pressure drop is identical to the model utilized in the core design analysis with the exception of the acceleration pressure drop component. The models utilized in the core design analysis are as follows:

4.4.2.3.1 Friction Pressure Drop

Friction pressure drop is calculated with a basic model as follows:

$$\Delta P_f = \frac{w^2}{2g_c\rho}\frac{fL}{D_H A_{ch}^2}\phi_{TPF}^2$$

where

ΔP_f = friction pressure drop, psi

w = mass flow rate

g_c = conversion factor

ρ = average nodal liquid density

(17) The opening pressure for the SRVs mechanical lift mode satisfies the overpressure protection analysis.

(18) The opening time for the SRVs in the overpressure operation of self-actuated or mechanical lift mode, which is measured from when the pressure exceeds the valve set pressure to when the valve is fully open, shall be less than or equal to the design opening time.~~The opening time for the SRVs (in the overpressure operation of self-actuated or mechanical lift mode) is measured from when the pressure exceeds the valve set pressure to when the valve is fully open shall be less than or equal to the design opening time.~~

(19) The steam discharge capacity of each SRV satisfies (i.e., is greater than or equal to that used in) the overpressure protection analysis.

(20) The opening pressure for the SVs satisfies (i.e., is less than or equal to that used in) the overpressure protection analysis.

(21) The opening time for the SVs is measured from when the pressure exceeds the valve set pressure to when the valve is fully open shall be less than or equal to the design opening time.

(22) The steam discharge capacity of each SV satisfies (i.e., is greater than or equal to that used in) the overpressure protection analysis.

(23) The relief-mode actuator (and safety-related appurtenances) can open each SRV with the drywell pressure at design pressure.

(24) When actuated by an ~~initiator~~(igniter charge), the booster assembly opens each DPV in less than or equal to the design opening time (opening time to full rated capacity)~~and design conditions~~.

(25) Each DPV minimum flow capacity is sufficient to support rapid depressurization of the RPV (i.e., has a flow capacity that is greater than or equal to the design flow capacity under design basis conditions).

(26) The equipment qualification of the NBS components is addressed in Tier 1, Section 3.8.

(27) The containment isolation portions of the NBS are addressed in Tier 1, Subsection 2.15.1.

(28) Vacuum breakers are provided on SRV discharge lines to reduce the post-discharge reflood height of water in the discharge lines.

(29) The SRV discharge line (SRVDL) vacuum breakers close to prevent steam bypass to the drywell during SRV discharge, and open following a discharge completion to permit pressure equalization with the drywell and prevent ingestion of a water slug into the SRVDL.

(30) The pressure loss coefficient of each of the following components is within the uncertainty band of the pressure loss coefficient used in the natural circulation flow analysis:

- Steam separator
- Fuel bundle
- Fuel support piece orifice
- Control rod guide tubes
- Shroud support

(31) The free volume for each of the following components is within the uncertainty band of the free volume used in the natural circulation flow analysis:

- RPV
- Downcomer
- Core
- Chimney
- Separator/dryer

(32) The hydraulic diameter, geometry of the heated surfaces, and flow area in fuel assemblies are within the uncertainty band of the geometry used in the natural circulation flow analysis.

(33) NBS software is developed in accordance with the software development program described in Section 3.2.

Refer to Subsection 2.2.15 for "Instrumentation and Controls Compliance with IEEE Standard 603."

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.2-3 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria for the NBS.

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
30. The pressure loss coefficient of each of the following components is within the uncertainty band of the pressure loss coefficient used in the natural circulation flow analysis. • Steam separator • Fuel bundle • Fuel support piece orifice • Control rod guide tubes • Shroud support	As-built component records will be inspected and compared against inputs to the natural circulation analysis, considering the uncertainty analysis, performed to calculate pressure loss coefficients.	A report exists that documents that the pressure loss coefficient of each of the following components is within the uncertainty band of the pressure loss coefficient used in the natural circulation flow analysis: • Steam separator • Fule bundle • Fuel support piece orifice • Control rod guide • Shroud support
31. The free volume for each of the following components is within the uncertainty band of the free volume used in natural circulation flow analysis: • RPV • Downcomer • Core • Chimney • Separator/dryer	Inspection of as-built component records will be performed to determine the component free volume for each of the listed components	A report exists that concludes that the free volume of each of the following components is within the uncertainty band of the free volume used in the natural circulation flow analysis: • RPV • Downcomer • Core • Chimney • Separator/dryer

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
32. The hydraulic diameter, the geometry of heated surfaces, and flow area in fuel assemblies are within the uncertainty band of the geometry used in the natural circulation flow analysis.	As-built dimension inspection and analyses will be performed to determine the geometry of the fuel assemblies to be loaded.	A report exists and concludes that the hydraulic diameter, the geometry of heated surfaces, and flow area in the fuel assemblies are within the uncertainty band of the geometry used in the natural circulation flow analysis.
33. NBS software is developed in accordance with the software development program described in Section 3.2.	Section 3.2.	Section 3.2.