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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 53 – DCD Chapter 4 – RAI Number 4.4-1**

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

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

Sincerely,

Kathy Sedney for

James C. Kinsey
Project Manager, ESBWR Licensing

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 the ESBWR Design Certification Application*, August 16, 2006

Enclosures:

1. MFN 07-021 – Response to Portion of NRC Request for Additional Information Letter No. 53 – DCD Chapter 4 – RAI Number 4.4-1

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GE/San Jose (with enclosures)
	eDRF	0061-0104

ENCLOSURE 1

MFN 07-029

Partial Response to RAI Letter No. 53 Related to ESBWR Design Certification Application

DCD Chapter 4

RAI Number 4.4-1

NRC RAI 4.4-1:

Some sections of DCD Tier 2, Section 4.4 do not contain sufficient detail for the reviewer to make a determination of acceptability. Provide additional detailed information as follows:

a) *If analyses or tests are necessary to demonstrate compliance with regulations, provide a discussion of the theoretical or experimental basis, the method used, the assumptions and boundary conditions, the limitations, and the results as applied to the ESBWR design.*
b) *Discuss the means by which the design addresses the regulatory guidance outlined in NUREG-0800, Standard Review Plan, Section 4.4, Thermal and Hydraulic Design, Revision 1, including General Design Criteria, Regulatory Guides, and other referenced documents.*
c) *If traditional computational methods are used (i.e., those which have already been accepted for conventional BWR's), provide the technical justification (qualitatively and quantitatively) as to why these computational methods remain appropriate for the ESBWR evaluation. That is, why do current computational methods apply to the ESBWR core design?*
d) *If new methods are used, identify the differences from conventional computational methods and the justification for their use. When thermal hydraulic limits are presented, such as the operating limit minimum critical power ratio (OLMCPR) and any other relevant Safety Limits, please provide a discussion of any differences from conventional methods for determining these limits.*

A top level guide to the appropriate sections would be beneficial. The DCD sections which are considered to require additional discussion are identified in subsequent RAI's. Other sections which did not have an associated question identified, but which could be enhanced by additional discussion, include: 4.4.1.3, 4.4.1.4, 4.4.1.5, 4.4.1.6, 4.4.1.7, 4.4.2.1.1, 4.4.2.1.3, 4.4.2.2, 4.4.2.3, 4.4.2.5, 4.4.2.6, 4.4.3.1.1, 4.4.3.1.2, 4.4.3.1.3, 4.4.3.3, 4.4.3.4, 4.4.3.5, and 4.4.3.6

GE Response:

GE has already responded to many relevant Section 4.4 RAIs (4.4-2 to 4.4-9, and 4.4-15 to 4.4-56), which addressed most of the above issues. The changes proposed to Section 4.4 based on the GE responses contained in MFN Letter 06-297 (Reference 4.4-1-1) and 06-349 (Reference 4.4-1-2) has been incorporated in the DCD Tier 2, Revision 2. These revisions are listed in Table 4.4-1-1 below.

Table 4.4-1-1 DCD Tier 2, Revision 2 to Section 4.4

DCD Tier 2 Subsection	NRC RAI	Reference
4.4.2.1.1 Bundle Critical Power Performance Method	4.4-5	4.4-1-1
4.4.2.1.2 Fuel Cladding Integrity Safety Limit Statistical Method	4.4-56	4.4-1-1
4.4.2.2 Void Fraction Distribution Methods	4.4-2	4.4-1-1
4.4.3.1.2 Fuel Cladding Integrity Safety Limit Evaluation	4.4-19	4.4-1-1
4.4.3.2 Void Fraction Distribution Evaluations	4.4-2	4.4-1-1
4.4.5 Loose-Parts Monitoring System **(Subsection deleted)**	4.4-7 to 4.4-9	4.4-1-2

Many other Section 4.4 revisions based on the GE responses contained in subsequent MFN letters (References 4.4-1-3 to 4.4-1-6) and the NRC Supplemental Information Request to RAI letter No. 53 (Reference 4.4-1-7) are listed later in this response. These changes will be incorporated in Revision 3 of the DCD Tier 2.

On item (a) above, in the GE responses to NRC RAIs related to the critical power, void fraction distribution, core pressure drop, core coolant flow distribution and fuel heat transfer, GE has discussed that no new tests are necessary to demonstrate compliance with regulations and the tests conducted earlier are also applicable to the ESBWR conditions. The analysis methodologies are also applicable to the ESBWR. To clarify/amplify these further, GE/GNF has revised the Licensing Topical Report (LTR) "GE14 for ESBWR – Critical Power Correlation, Uncertainty, and OLMCPR Development," NEDC-33237P (Reference 4.4-12 of the DCD Tier 2) as per Table 4.4-1-2 below. This is in addition to the proposed Revision 3 to Section 4.4 of the DCD Tier 2, discussed later.

Table 4.4-1-2 Planned Revisions to NEDC-33237P

NEDC-33237P Section	NRC RAI	Reference
1.0 Introduction	4.4-25	4.4-1-3
3.0 The GEXL Correlation	4.4-28	4.4-1-5
4.1 The GEXL14 Data Base	4.4-29	4.4-1-5
4.3 Comparison of GEXL14 Correlation with Adjusted GE14 Data	4.4-28 and 4.4-30	4.4-1-5 and 4.4-1-3
5.1 Introduction **5.13** ESBWR Operating Limit MCPR Evaluation Methodology **(a new Section to be added)**	4.4-32	4.4-1-1
Appendices A, B and C (to be added)	4.4-26	4.4-1-5

On item (b), the thermal and hydraulic design of GE ESBWR fully complies with the regulatory requirements outlined in NUREG-0800, Standard Review Plan, Section 4.4, Thermal and Hydraulic Design, Revision 1. Specifically, the ESBWR thermal and hydraulic design conforms to the General Design Criterion 10 (GDC 10), as it relates to the reactor core design with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operation or anticipated operational occurrences (AOOs). This has been discussed in Subsections 4.4.1, 4.4.1.1 and 4.4.1.1.1 of the DCD Tier 2. The ESBWR core design conforms to GDC 10 by establishing the limiting (minimum) value of CPR (Critical Power Ratio) such that at least 99.9% of the fuel rods in the core would not be expected to experience departure from nucleate boiling or boiling transition during normal operation or AOOs.

The ESBWR core design also conforms to GDC 12 to assure that power oscillations, which could result in conditions exceeding specified acceptable fuel design limits, are not possible or can be reliably and readily detected and suppressed. This has been discussed, in detail, in Appendix 4D (Stability Evaluation) of the DCD Tier 2.

The ESBWR initial test program will follow the NRC Regulatory Guide 1.68 as discussed in Subsection 14.2.3 of the DCD Tier 2. However, the Regulatory Guide 1.133 related to the Loose-Parts Monitoring System is not applicable to the ESBWR since this system has been deleted in response to the NRC RAIs 4.4-7 through 4.4-9 (Reference 4.4-1-2).

The effect of crud in the thermal-hydraulic design has been discussed in the GE response to NRC RAI 4.4-38 (Reference 4.4-1-1). Finally, the issue of Inadequate Core Cooling (ICC) monitoring system has been discussed in the GE response to RAI 4.4-21 (Reference 4.4-1-4) and a new Subsection 4.4.4.7 will be added in the DCD Tier 2.

Regarding item (c), GE uses the TRACG code for the systems analyses of ESBWR and a core simulator code for the core design. In GE responses to several of the Section 4.4 RAIs, the models used in these two codes are shown to apply to both the conventional BWRs and the ESBWR. Thus both TRACG and the core simulator code are applicable to the ESBWR evaluation and core design. NRC has already approved TRACG for the ESBWR LOCA application and ESBWR stability analysis, and approval of TRACG for the ESBWR AOO application is part of the ESBWR Design Certificate Application. The LTRs on TRACG Model Description and Qualification (References 4.4-10 and 4.4-11 in the DCD Tier 2) were also included in the NRC review of "TRACG Application for ESBWR" (Reference 4.4-9 in the DCD Tier 2) by reference. The LTR on ESBWR nuclear design, "GE14 for ESBWR Nuclear Design Report," NEDC-33239P (Reference 4.3-8 in the DCD Tier 2) is also under NRC review.

On item (d), GE reiterates that no new methods are used for the ESBWR. However, the LTR on "GE14 for ESBWR – Critical Power Correlation, Uncertainty, and OLMCPR Development," NEDC-33237P (Reference 4.4-12 in the DCD Tier 2) has been revised as per Table 4.4-1-2 to clarify/amplify various issues on the critical power correlation, uncertainties, OLMCPR development and other relevant safety limits.

GE agrees that a top level guide to direct a reader to various subsections of Section 4.4 of the DCD would be beneficial. Therefore, the following sentences will be added (in Revision 3) at the end of the first paragraph of Section 4.4, but just before Subsection 4.4.1:

"Subsections 4.4.1, 4.4.2 and 4.4.3 describe the Design Basis, Methods and Evaluations of Reactor Core Thermal Hydraulics, respectively. Within each of these subsections, several topics, e.g., critical power, void fraction, pressure drop, etc., are discussed sequentially. A reader may, therefore, go through subsections 4.4.1.n, 4.4.2.n and 4.4.3.n to obtain the complete information on a particular topic, n."

GE will also add the references and other discussion in the appropriate Subsections as per the NRC Supplemental Information Request to RAI letter No. 53 (Reference 4.4-1-7) related to RAIs 4.4-6, 4.4-15, 4.4-16, 4.4-17 and 4.4-24. Table 4.4-1-3 lists all the 4.4 Subsections, along with the RAI Nos., and revised in Revision 3 of the DCD Tier 2.

Table 4.4-1-3 Proposed DCD Tier 2, Revision 3 to Section 4.4

DCD Tier 2 Subsection	NRC RAI	Reference
4.4 (Top Level Guide)	4.4-1	This response
4.4.1.3 Core Pressure Drop and Hydraulic Loads Bases	4.4-20	4.4-1-6
4.4.1.5 Fuel Heat Transfer Bases	4.4-3	4.4-1-4
4.4.1.7 Summary of Design Bases	4.4-4	4.4-1-4
4.4.2.3 Core Pressure Drop and Hydraulic Loads Methods	4.4-6 Supplement	4.4-1-7
4.4.2.3.1 Friction Pressure Drop	4.4-6 Supplement	4.4-1-7
4.4.2.3.2 Local Pressure Drop	4.4-6, 4.4-15 and 4.4-16 Supplement	4.4-1-7
4.4.2.3.5 Total Pressure Drop Qualification	4.4-24 Supplement	4.4-1-7
4.4.2.3.6 Hydraulic Loads **(a new Subsection to be added)**	4.4-20	4.4-1-6
4.4.2.4 Core Coolant Flow Distribution Methods	4.4-17 Supplement	4.4-1-7
4.4.2.5 Fuel Heat Transfer Methods	4.4-18	4.4-1-4
4.4.3.3 Core Pressure Drop and Hydraulic Loads Evaluation	4.4-20	4.4-1-6
4.4.4.7 Inadequate Core Cooling (ICC) Monitoring System **(a new subsection to be added)**	4.4-21	4.4-1-4

In view of the above-proposed revisions to DCD Tier 2 Section 4.4 and the revision to NEDC-33237P as per Table 4.4-1-2, GE does not propose to add any further elaboration in other subsections mentioned in this RAI.

DCD Impact:

For the sake of completeness, the proposed Revision 3 of the DCD Tier 2 Section 4.4, except for the Tables that will not change, is attached to this response. Revision bars indicating changes from Revision 2 are also shown.

References:

4.4-1-1. GE Energy Letter # MFN 06-297 dated August 23, 2006, to USNRC, "Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-2 through 4.2-7, 4.3-3, 4.3-4, 4.4-2, 4.4-5, 4.4-6, 4.4-15 through 4.4-17, 4.4-19, 4.4-24, 4.4-27, 4.4-31 through 4.4-34, 4.4-36 through 4.4-38, 4.4-42 through 4.4-50, 4.4-52 through 4.4-56, 4.8-1 through 4.8-16."

4.4-1-2. GE Energy Letter # MFN 06-349 dated September 29, 2006, to USNRC, "Response to NRC Request for Additional Information Letter No. 56 – Loose Parts Monitoring System – RAI Numbers 4.4-7 through 4.4-9, Single Failure Evaluation – RAI Number 6.3-38."

4.4-1-3. GE Energy Letter # MFN 06-350 dated September 29, 2006, to USNRC, "Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.3-2, 4.3-5, 4.4-25, 4.4-30, 4.4-35, 4.4-39, 4.4-51."

4.4-1-4. GE Energy Letter # MFN 06-399 dated October 18, 2006, to USNRC, "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."

4.4-1-5. GE Energy Letter # MFN 06-405 dated October 18, 2006, to USNRC, "Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.4-26, 4.4-28 and 4.4-29."

4.4-1-6. GE Energy Letter # MFN 06-498 dated December 7, 2006, to USNRC, "Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 – RAI Number 4.4-20."

4.4-1-7. NRC (Amy Cubbage) e-mail dated November 12, 2006, to GE, "Supplemental Request for Information on Response to RAI Letter #53 (MFN-06-297)".

Appendix (Proposed Revision 3 to Section 4.4 of the DCD Tier 2)

4.4 THERMAL AND HYDRAULIC DESIGN

This section describes the design bases and functional requirements used in the thermal and hydraulic design of the fuel, core and reactivity control system and relates these design bases to the General Design Criteria (GDC). Subsections 4.4.1, 4.4.2 and 4.4.3 describe the Design Basis, Methods and Evaluations of Reactor Core Thermal Hydraulics, respectively. Within each of these subsections, several topics, e.g., critical power, void fraction, pressure drop, etc., are discussed sequentially, A reader may, therefore, go through subsections, 4.4.1.n, 4.4.2.n and 4.4.3.n to obtain the complete information on a particular topic, n.

4.4.1 Reactor Core Thermal and Hydraulic Design Basis

Thermal-hydraulic design of the core shall establish the thermal-hydraulic safety limits for use in evaluating the safety margin in accordance with GDC 10. The thermal hydraulic stability performance of the core addressing GDC 12 is covered in Section 4.3 and Appendix 4D.

Margin to specified acceptable fuel design limits is maintained during normal steady-state operation when the minimum critical power ratio (MCPR) is greater than the required MCPR operating limit (OLMCPR) and the linear heat generation rate (LHGR) is maintained below the maximum LHGR (MLHGR) limit(s). The steady-state OLMCPR and MLHGR limits are determined by analysis of the most severe anticipated operational occurrences (AOOs) to accommodate uncertainties and provide reasonable assurance that no fuel damage results during AOOs. These limits are required by the Technical Specifications.

4.4.1.1 Critical Power Bases

The objective for normal operation and AOOs is to maintain nucleate boiling and thus avoid a transition to film boiling. Limits are specified to maintain adequate margin to the onset of the boiling transition. The figure of merit utilized for plant operation is the critical power ratio (CPR). The CPR is the ratio of the bundle power at which some point within the assembly experiences onset of boiling transition to the operating bundle power. Thermal margin is stated in terms of the minimum value of the critical power ratio (MCPR) that corresponds to the most limiting fuel assembly in the core. The design requirement is based on a statistical analysis such that for AOOs at least 99.9% of the fuel rods would be expected to avoid boiling transition (Reference 4.4-8 and 4.4-9).

4.4.1.1.1 Fuel Cladding Integrity Safety Limit Bases

GDC 10 requires, and safety limits ensure, that specified acceptable fuel design limits are not exceeded during steady-state operation, normal operational transients, and anticipated operational occurrences (AOOs). Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of boiling transition have been used to mark the beginning of the region in which fuel damage could occur. The Fuel Cladding Integrity Safety Limit (FCISL) is set such that no significant fuel damage is calculated to occur during normal operation and AOOs. Although it is recognized that the onset of boiling transition would not result in damage to BWR fuel rods, a calculated fraction of rods expected to avoid boiling transition has been adopted as a convenient limit. The FCISL is defined as the fraction (%) of total fueled rods that are expected to avoid boiling transition during normal operation and AOOs. A value of 99.9% provides assurance that specified acceptable fuel design limits are met.

4.4.1.1.2 MCPR Operating Limit Calculation Bases

A plant-unique MCPR operating limit (OLMCPR) is established to provide adequate assurance that the FCISL for that plant is not exceeded during normal operation and any AOO. By operating with the MCPR at or above the OLMCPR, the FCISL for that plant is not exceeded during normal operation and AOOs. This operating requirement is obtained by statistically combining the maximum ΔCPR/ICPR (the change in CPR through the transient divided by the initial CPR) value for the most limiting AOO from conditions postulated to occur at the plant with the uncertainties associated with plant initial conditions and modeling of the transient ΔCPR.

4.4.1.2 Void Fraction Distribution Bases

The void fraction in a boiling water reactor (BWR) fuel bundle has a strong effect on the nuclear flux and power distribution. Therefore accurate prediction of the void fraction is important for evaluation of the performance of the BWR reactor and fuel. In design and licensing calculations the void fraction is evaluated using empirical correlations based on the characteristic dimensions of the fuel bundle and hydraulic properties of the two-phase flow in the fuel bundle.

4.4.1.3 Core Pressure Drop and Hydraulic Loads Bases

The accuracy on the prediction of core pressure drop is essential to the modeling of fuel and core inlet flow and hydraulic loads. Further details are given in Subsection 4.4.2.3.

4.4.1.4 Core Coolant Flow Distribution Bases

Based on the prediction of core pressure drop, the distribution of flow into the fuel channels and the core bypass regions are calculated. The core coolant flow distribution forms the basis of the prediction of steady-state and transient critical power and void fraction.

4.4.1.5 Fuel Heat Transfer Bases

The model must accurately predict heat transfer between the coolant, fuel rod surface, cladding, gap, and fuel pellet in the evaluation of core and fuel safety criteria. 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

4.4.1.6 Maximum Linear Heat Generation Rate Bases

The Maximum Linear Heat Generation Rate (MLHGR) bases are described in Section 4.2. The adequacy of MLHGR limits are evaluated for the most severe anticipated operational occurrences (AOOs) to provide reasonable assurance that no fuel damage results during AOOs.

4.4.1.7 Summary of Design Bases

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.

4.4.2 Reactor Core Thermal and Hydraulic Methods

This section contains a description of the application of NRC-approved methods to the ESBWR. Changes may be made to these techniques provided that NRC-approved (including applicability to ESBWR) methods, models, and application methodologies are used.

4.4.2.1 Critical Power Methods

The qualification of the critical power methods for ESBWR is discussed in Subsection 4.4.3.

4.4.2.1.1 Bundle Critical Power Performance Method

The bundle critical power performance methodology is described in References 4.4-8 and 4.4-16. These references describe the form of the GEXL correlation and the experimental qualification that demonstrates the GEXL correlation adequately predicts the bundle critical power over a wide range of fluid parameters, axial power shapes and heated lengths. Each fuel bundle design has a specific set of correlation coefficients developed from full-scale test data. The specific GEXL correlation applied in the analysis of GE14E for ESBWR is designated GEXL14. The applicability of GEXL14 to GE14E is addressed in Reference 4.4-12.

4.4.2.1.2 Fuel Cladding Integrity Safety Limit Statistical Method

The statistical analysis utilizes a model of the core that simulates the core monitoring system. The model produces a critical power ratio (CPR) map of the core based on steady-state uncertainties defined in Table 5-1 of Reference 4.4-12. This is coupled with the TRACG ΔCPR/ICPR results to develop the OLMCPR. Details of the procedure are documented in Section 5.13 of Reference 4.4-12 and Subsection 4.6.3 of Reference 4.4-9. Random Monte Carlo selections of all operating parameters based on the uncertainty ranges of manufacturing tolerances, uncertainties in measurement of core operating parameters, calculation uncertainties, the uncertainty in the calculation of the transient ΔCPR/ICPR and statistical uncertainty associated with the critical power correlations are imposed on the analytical representation of the core and the resulting bundle critical power ratios are calculated.

The number of rods expected to avoid boiling transition is determined for each random Monte Carlo trial. The initial MCPR during normal operation corresponds to the OLMCPR when the FCISL (99.9% of the rods are expected to avoid boiling transition) is met for a statistical combination of the trials. Reference 4.4-9 defines the statistical combination method.

4.4.2.1.3 MCPR Operating Limit Calculation Method

All ESBWR AOO events are analyzed using the TRACG model described in Reference 4.4-10. The core thermal hydraulic models have been qualified in Reference 4.4-11. Uncertainties have been developed for all High and 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.

4.4.2.2 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 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 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 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

D_H = channel hydraulic diameter

A_{ch} = channel flow area

L = incremental length

f = friction factor

ϕ_{TPF} = two–phase friction multiplier

The formulation for the two-phase multiplier is similar to that presented in References 4.4-4 and 4.4-5. The single-phase friction factor and two-phase friction multiplier in the above equation were validated by extensive comparisons to full-scale rod bundle pressure drop data provided in Reference 4.4-14.

4.4.2.3.2 Local Pressure Drop

The local pressure drop is defined as the irreversible pressure loss associated with an area change, such as the orifice, lower tie plate, and spacers of a fuel assembly.

The general local pressure drop model is similar to the friction pressure drop and is

$$\Delta P_L = \frac{w^2}{2g_c\rho}\frac{K}{A_{ch}^2}\phi_{TPL}^2$$

where

ΔP_L = local pressure drop, psi

K = local pressure drop loss coefficient

A = reference area for local loss coefficient

ϕ_{TPL} = two–phase local multiplier

and w, g_c, and ρ are as previously defined. The formulation for the two-phase multiplier is similar to that reported in Reference 4.4-5. The local loss component of the total pressure drop across a region inside the fuel assembly is deduced from the measured total pressure drop by subtracting the frictional and acceleration components. The corresponding local loss coefficient is then determined using the above local pressure drop formula. The pressure drop data taken for the specific designs of the BWR fuel assembly, discussed in Reference 4.4-14, were obtained from tests performed in single-phase water to calibrate the orifice, the lower tie plate, and the holes in the lower tie plate, and in both single and two-phase flow, to derive the best fit design values for spacer and upper tie plate pressure drop. The range of test variables was specified to include the range of interest for boiling water reactors. New test data are obtained whenever there is a significant design change to ensure the most applicable methods are used. However, the ESBWR reference fuel assembly (GE14E) utilizes the same hardware currently used in the GE14 fuel assembly, i.e. the same upper-tie plate, spacers, water rods, and the same bundle inlet design. Therefore, its pressure drop characteristics at the upper-tie plate region, spacers, water rods, and the bundle inlet region remains unchanged and no new test data is required. The applicability of the data to ESBWR is discussed in Subsection 4.4.2.3.5.

4.4.2.3.3 Elevation Pressure Drop

The elevation pressure drop is based on the relation:

$$\Delta P_E = \bar{\rho}\Delta L\frac{g}{g_c}$$

$$\bar{\rho} = \rho_f(1-\alpha) + \rho_g\alpha$$

where

ΔP_E = elevation pressure drop

ΔL = incremental length

$\bar{\rho}$ = average mixture density

g = acceleration of gravity

α = nodal average void fraction

ρ_f, ρ_g = saturated water and vapor density, respectively

Other terms are as previously defined. The TRACG void fraction model is described in Reference 4.4-10. The void fraction model utilized in the core design analysis is described in References 4.4-6 and 4.4-15.

4.4.2.3.4 Acceleration Pressure Drop

A reversible pressure change occurs when an area change is encountered, and an irreversible loss occurs when the fluid is accelerated through the boiling process. The basic formulation for the reversible pressure change resulting from a flow area change in the case of single–phase flow is given by:

$$\Delta P_{ACC} = \left(1-\sigma_A^2\right)\frac{w^2}{2g_c\rho_f A_2^2}$$

$$\sigma_A = \frac{A_2}{A_1} = \frac{\textit{final flow area}}{\textit{initial flow area}}$$

where:

ΔP_{ACC} = acceleration pressure drop

A_2 = final flow area

A_1 = initial flow area

In the case of two–phase flow, the liquid density is replaced by a density ratio so that the reversible pressure change is given by:

$$\Delta P_{ACC} = \left(1-\sigma_A^2\right)\frac{w^2\rho_H}{2g_c\rho_{KE}^2 A_2^2}$$

where:

$$\frac{1}{\rho_H} = \frac{x}{\rho_g} + \frac{1-x}{\rho_f}\text{, homogeneous density,}$$

$$\frac{1}{\rho_{KE}^2} = \frac{x^3}{\rho_g^2\alpha^2} + \frac{(1-x)^3}{\rho_f^2(1-\alpha)^2}\text{, kinetic energy density,}$$

α = void fraction at A_2

x = steam quality at A_2

Other terms are as previously defined. The basic formulation for the acceleration pressure change due to density change is:

$$\Delta P_{ACC} = \frac{w^2}{g_c A_{ch}^2}\left[\frac{1}{\rho_{OUT}} - \frac{1}{\rho_{IN}}\right]$$

where ρ is either the homogeneous density, ρ_H, or the momentum density, ρ_M

$$\frac{1}{\rho_M} = \frac{x^2}{\rho_g \alpha} + \frac{(1-x)^2}{\rho_f (1-\alpha)}$$

ρ is evaluated at the inlet and outlet of each axial node. Other terms are as previously defined. The total acceleration pressure drop in boiling water reactors is on the order of a few percent of the total pressure drop. Note that the TRACG model is different for the acceleration pressure drop modeling (see Reference 4.4-10).

4.4.2.3.5 Total Pressure Drop Qualification

The GE14 pressure drop is characterized in Reference 4.4-14. The loss coefficients are qualified against pressure drop test data. The test range includes the operating conditions for the ESBWR. The ESBWR reference fuel assembly (GE14E) utilizes the same hardware currently used in the GE14 fuel assembly, i.e. the same upper-tie plate, spacers, water rods, and the same bundle inlet design. Therefore, its pressure drop characteristics at the upper-tie plate region, spacers, water rods, and the bundle inlet region remains unchanged. Moreover, the differences in active fuel length, spacer separation, and part-length rod height between the GE14 and GE14E fuel design are accounted for in determining the local loss coefficients from the experimental data as explained in Subsection 4.4.2.3.2. Because the operating conditions and geometry are compatible, the loss coefficients can be applied to the ESBWR. The uncertainty in the core pressure drop is defined by Reference 4.4-9 Subsection 4.4.1 item C23.

4.4.2.3.6 Hydraulic Loads

Hydraulic loads are determined based on the reactor internal pressure differences (RIPDs) discussed in Subsection 3.9.5.3. The TRACG computer code is used to analyze the transient conditions within the reactor vessel following AOOs, infrequent events and accidents (e.g., LOCA).

4.4.2.4 Core Coolant Flow Distribution Methods

The core coolant flow distribution methods used in TRACG are described in Reference 4.4-10 Chapters 6 and 7. TRACG treats all fuel channels as one-dimensional (axial) components, but the vessel is modeled as a three-dimensional component. Hence, the pressure drop across two planes in the vessel is the same at all radial and azimuth locations if the geometry of the components in the vicinity of these planes has radial and azimuthal symmetry. Otherwise, this pressure differential displays some (locally) radial and azimuth non-uniformity.

The flow distribution to the fuel assemblies and bypass flow paths in the core simulator model is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. The bundle pressure drop evaluation includes frictional, local, elevation, and acceleration losses (Subsections 4.4.2.3.1 - 4.4.2.3.4). The pressure drop methodology has been qualified to the whole range of test data discussed in Reference 4.4-14. The core inlet flow is an input to the core simulator model. The value used in the core design analysis is determined based on the TRACG prediction of the natural circulation core inlet flow. In operation, the core monitoring system will determine core inlet flow based on plant instrumentation (see Chapter 7).

The bypass flow methodology is described in Reference 4.4-10 Subsection 7.5.1. The same methodology supports the core simulator model.

4.4.2.5 Fuel Heat Transfer Methods

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 convection 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

4.4.2.6 Maximum Linear Heat Generation Rate Methods

The Maximum Linear Heat Generation Rate (MLHGR) methods are described in Section 4.2. Margin to design limits for circumferential cladding strain and centerline fuel temperature is evaluated for AOOs in accordance with Reference 4.4-9 Subsection 4.6.2.1.

4.4.3 Reactor Core Thermal and Hydraulic Evaluations

Typical thermal-hydraulic parameters for the ESBWR are compared to those for a typical BWR/6 plant and the ABWR in Table 4.4-1.

4.4.3.1 Critical Power Evaluations

4.4.3.1.1 Bundle Critical Power Performance Evaluation

The bundle critical power performance results are described in Reference 4.4-12. This reference utilizes full-scale test data to support the development of the critical power correlation for ESBWR. Compliance to steady-state MCPR operating limits is demonstrated for a typical simulation of an equilibrium cycle in Appendix 4A.

4.4.3.1.2 Fuel Cladding Integrity Safety Limit Evaluation

The Fuel Cladding Integrity Safety Limit (FCISL) is defined as 99.9% of the total fueled rods are expected to avoid boiling transition during normal operation and AOOs. Section 6 of Reference 4.4-12 provides a summary of the basis for the representative operating limit MCPR used for the ESBWR to protect the FCISL. Section 5 of Reference 4.4-12 provides the basis for the uncertainties specific to the ESBWR used in this evaluation.

4.4.3.1.3 MCPR Operating Limit Evaluation

The MCPR Operating Limit ΔCPR/ICPR results are described in Section 15.2. The MCPR Operating Limit development including incorporation of the Fuel Cladding Integrity Safety Limit uncertainties is described in Reference 4.4-12.

4.4.3.2 Void Fraction Distribution Evaluations

The axial distribution of void fractions for the radially average channel and a conservative hot channel as predicted by TRACG are given in Table 4.4-2. The core average and maximum exit values are also provided. Similar distributions for steam quality are given in Table 4.4-3. The axial power distribution used to produce these tables is given in Table 4.4-4. The axial void and power distributions for the channel with the highest exit void fraction for the core reference loading pattern (Figure 4.3-1 and Appendix 4A) are given in Table 4.4-5.

The expected operating void fraction for the ESBWR is within the qualification basis of the void fraction methods. The void fractions in Table 4.4-2a and 4.4-2b are based on TRACG. The hot channel in Table 4.4-2b is a hypothetical channel with a bundle power (radial power) set so as to result in a CPR of 1.20. This hot channel has a maximum void fraction of 0.92. This is conservative compared to the assumed OLMCPR for ESBWR. The void fraction qualification database (References 4.4-11 and 4.4-15) contains void fractions in excess of 0.92 and covers the void fraction range expected for normal steady-state operation as well as AOOs. The core simulator maximum exit void fraction, for the steady-state simulation in Appendix 4A, is 0.90 as shown in Table 4.4-5.

The TRACG AOO calculations in Chapter 15 include the consideration of uncertainty in the void fraction.

4.4.3.3 Core Pressure Drop and Hydraulic Loads Evaluations

The expected operating pressure for the ESBWR is within the qualification basis of the pressure drop methods. The TRACG AOO calculations in Chapter 15 include the consideration of uncertainty in the channel pressure drop. The statistical OLMCPR method also assumes pressure drop uncertainty. The impact of uncertainty in core pressure drop is included in the results provided in Reference 4.4-12. Evaluations of hydraulic loads for the reactor internals and the fuel assembly are discussed in Subsection 3.9.5.4 and 4.2.3, respectively.

4.4.3.4 Core Coolant Flow Distribution Evaluations

The impact of uncertainty in core flow distribution is included in the results provided in Reference 4.4-12.

4.4.3.5 Fuel Heat Transfer Evaluations

The fuel heat transfer models are included in evaluations described in Section 4.2 and Chapter 15.

4.4.3.6 Maximum Linear Heat Generation Rate Evaluations

The AOO results are described in Chapter 15 Section 15.2. Compliance to steady-state MLHGR limits is demonstrated for a typical simulation of an equilibrium cycle in Appendix 4A.

4.4.4 Description of the Thermal–Hydraulic Design of the Reactor Coolant System

4.4.4.1 Plant Configuration Data

4.4.4.1.1 Reactor Coolant System Configuration

The Reactor Coolant System is described in Chapter 5. The ESBWR reactor coolant system is shown in Figure 5.1-1. The ESBWR design is similar to that of the operating BWRs, except that the recirculation pumps and associated piping are eliminated. Circulation of the reactor coolant through the ESBWR core is accomplished via natural circulation. The natural circulation flow rate depends on the difference in water density between the downcomer region and the core region. The core flow varies according to the power level, as the density difference changes with changes in power levels. Therefore, a core power-flow map is only a single line and there is no active control of the core flow at any given power level.

4.4.4.1.2 Reactor Coolant System Thermal–Hydraulic Data

The steady-state distribution of temperature, pressure and flow rate for each flow path in the Reactor Coolant System is shown in Figure 1.1-3.

4.4.4.1.3 Reactor Coolant System Geometric Data

Volumes of regions and components within the reactor vessel are shown in Figure 5.1-1. Table 4.4-6 provides the flow path length, height, liquid level, minimum elevations, and flow areas for each major flow path volume within the reactor vessel.

4.4.4.2 Operating Restrictions on Pumps

Not Applicable to the ESBWR. The ESBWR is a natural circulation design.

4.4.4.3 Power/Flow Operating Map

The core power-flow map is only a single line and there is no active control of the core flow at a given power level.

4.4.4.4 Temperature-Power Operating Map

Not Applicable to the ESBWR.

4.4.4.5 Load Following Characteristics

Load following is implemented through the Plant Automation System (PAS). This is described in Chapter 7 Subsection 7.7.4.

4.4.4.6 Thermal-Hydraulic Characteristics Summary Tables

The thermal-hydraulic characteristics are provided in Table 4.4-1 and Table 4.4-6. The axial power distributions for the average and hot channels are shown in Table 4.4-4. The axial distribution of void fractions for the average power channel and the hot channel are given in Table 4.4-2. The core average and core maximum exit void fractions are also provided. Similar distributions for coolant flow quality are provided in Table 4.4-3.

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.

4.4.5 Loose-Parts Monitoring System

This system has been deleted

4.4.6 Testing and Verification

The testing and verification techniques to be used to assure that the planned thermal and hydraulic design characteristics of the core have been provided, and remain within required limits throughout core lifetime, are discussed in Chapter 14.

4.4.7 COL Unit-Specific Information

4.4.7.1 Reactor Core Thermal and Hydraulic Design

This section contains no requirement for additional information to be provided in support of the combined license. Combined Operating License applicants referencing the ESBWR certified design would address changes to the thermal-hydraulic design of the reactor coolant system, if different from that presented in the DCD.

4.4.8 References

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4.4-3 General Electric Company, "Brunswick Steam Electric Plant Unit 1 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations," NEDO-21215, March 1976.

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4.4-5 C. J. Baroczy, "A Systematic Correlation for Two-Phase Pressure Drop," Heat Transfer Conference (Los Angeles), AIChE, Preprint No. 37, 1966.

4.4-6 N. Zuber and J. A. Findlay, "Average Volumetric Concentration in Two-Phase Flow Systems," Transactions of the ASME Journal of Heat Transfer, November 1965.

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4.4-8 General Electric Company, "General Electric BWR Thermal Analysis Basis (GETAB): Data Correlation and Design Application," NEDO-10958-A, January 1977.

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4.4-12 GE Nuclear Energy, "GE14 for ESBWR Critical Power Correlation, Uncertainty, and OLMCPR Development", NEDC-33237P, Revision 1, Class III (proprietary), scheduled December 2006.

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4.4-15 "TASC-03A, A Computer Program for Transient Analysis of a Single Channel", NEDC-32084P-A, Revision 2, Class III (proprietary), July 2002.

4.4-16 Letter, J.S. Charnley (GE) to C. O. Thomas (NRC), Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A, January 25, 1986.