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Licensing Topical Report

Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring

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LIST OF CONDITIONS OF USE FOR ESBWR

1. On the Applicability of the LHGR Infinite Lattice Pin Power Uncertainty on the Lattice Exposure Range

   a) The LHGR Infinite Lattice Pin Power Uncertainty must be representative of the full range of fuel lattice exposure range. The calculated peak pellet exposure shall be confirmed to comply with the corresponding licensing limit approved by the NRC. The licensing limit for GE14E has been established in the design analyses described in NEDC-33242P.

2. On the Calculation of the LHGR Infinite Lattice Peaking Factor Uncertainty

   a) The LHGR Infinite Lattice Peaking Factor Uncertainty value is determined as the 95th percentile upper tolerance limit (UTL, bias plus two standard deviations) using the statistical analysis of the population of peak power as a function of exposure. The GE14E specific LHGT Infinite Lattice Peaking Factor Uncertainty determined using this approach is [[ ]]. This uncertainty will be determined whenever a new fuel product is applied to a particular ESBWR core loading.

3. On the Model Uncertainty Value for R-factor uncertainty calculation

   a) The method for calculating the R-factor uncertainty is described in NEDC-32601P-A. When determining the R-factor uncertainty for ESBWR analyses, the Infinite Lattice Peaking Model Uncertainty value will be assumed as equal to, or more conservative than, the LHGR Infinite Lattice Peaking Factor Uncertainty value for a particular ESBWR core loading.

4. On the Power Allocation Uncertainty

   a) The Power Allocation Uncertainty is a component of the LHGR and OLMCPR calculation uncertainties. Its value is determined using a 95th percentile UTL statistical analysis on gamma scan data. The value determined using this approach is reported in NEDC-33173P-A as [[ ]].

The applicability of conditions 2 and 4 is dictated by the 95th percentile UTL statistical approach approved in NEDC-33173P. Should the NRC approve an alternative approach for establishing the aforementioned uncertainties in subsequent supplement(s) to or revision(s) of the NEDC-33173P LTR, the approved, alternative approach may be adopted in NEDE-33197P-A in lieu of conditions 2 and 4 without separate NRC review and approval.

Any change on the uncertainty values of conditions 2, 3, and 4 shall be transmitted by GEH to the NRC prior to a licensee incorporating the change into their safety analysis basis.

5. <u>On the Applicability of the GT Adaption Technique</u>

   a) Modifications to the adaption technique in the PANAC11-based, Gamma-Thermometer-based Core Monitoring System (GT-CMS) described in NEDE-33197P-A document are considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and may not be used for licensing calculations without prior NRC review and approval.

   b) The parameters used to compensate for biases introduced in the GT sensor signal by the proximity to spacers and/or fuel type changes will be determined only when a new bundle design (i.e. new axial lattice composition) or a new spacer design (i.e. material) is applied to a particular ESBWR core loading, as described in the section 8.6 of the NEDE-33197P-A. The parameters will be incorporated into the GT-based monitoring system in a cycle-specific basis as required.

   c) The failure of a GT heater is considered a lost of calibration capability of the full GT string (all sensors). Therefore in case of failure of a GT heater, the GT-CMS will declare the GT string as inoperable.

6. On the Applicability of the Nuclear Analysis Codes

   a) Modifications to the TGBLA06/PANAC11 codes or the GT-CMS software that result in inconsistency with the NEDE-33197P-A LTR are considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and may not be used for licensing calculations without prior NRC review and approval of the necessary revision(s) to the LTR.

For conditions 5.a and 6.a: 10 CFR 52.98 outlines the regulatory change processes that may apply to address the potential for future code updates. Requirements for prior NRC review and approval of future code updates are consistent with the definition of a methodology change in 10 CFR 50.59(a)(1) and the criteria of 10 CFR 50.59(c)(2)(viii) to ensure that the methodology is not adversely impacted for reload licensing or core monitoring purposes."

REVISION CONTROL TABLE

Number	Description	Date
0	First issuance of the licensing topical report.	September 2005
1	General changes (company brand name, proprietary marking). Incorporation of responses to several RAIs as listed in eDRF object 0000-0074-1188.	September 2007
2	Several changes to incorporate RAI responses. Chapter 4 added a section for Void Fraction Response and Bypass Subcooling. Chapter 5 added a subsection for Core Monitoring with GT Adaption. Reorganization of Chapter 8 to incorporate GT Adaption details. Chapter 9 added a subsection for GT Adaptive Core Monitoring Accuracy and tables with MCPR, LHGR and GT interpolation uncertainties.	August 2008
3	Issuance of the NRC approved document. The suffix "-A" is appended to the document number. Responses to RAI 7.2-71 and 7.2-72 were incorporated into the document. Addition of this control table. Appendix A of revision 2 is removed but its content is presented in MFN 08-293 (current reference 6) and in MFN 07-544, Supplement 1 that is added as the new reference 11. Minor changes in the body text to reflect the revised referencing. Attachment 1 added and contains NRC Safety Evaluation report.	October 2010

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ABSTRACT

A Gamma Thermometer (GT) System for LPRM calibration and power shape monitoring is described. The major hardware and software components are identified and a practical GT response model is developed. Three in-plant tests at Limerick 2 (Exelon), Tokai 2 (JAPC) and Kashiwazaki-Kariwa 5 are provided to qualify the sensors and assess accuracy. An adaptive core monitoring simulation is used to assess the impact of the GT System on core monitoring. Lastly, an uncertainty analysis is performed to quantify the impact on bundle power uncertainty and on thermal limits uncertainties.

ACRONYMS AND ABBREVIATIONS

Term	Definition
BOC	Beginning of cycle
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CMS	Core Monitoring System
CPR	Critical Power Ratio
DACS	Data Acquisition and Calibration System
DAS	Data Acquisition System
DLC	Data Logging Computer
EOC	End Of Cycle
EWS	Engineering Work-Station
ESBWR	Economic Simplified Boiling Water Reactor
FS	Full Scale
GETAB	General Electric Thermal Analysis Basis
GT	Gamma Thermometer
GTC	Gamma Thermometer Control Unit
GTM	Gamma Thermometer Monitoring Unit
HPS	Heater Power Supply
HSU	Heater Switching Unit
LPRM	Local Power Range Monitor
LTR	Licensing Topical Report
MCPR	Minimum Critical Power Ratio
MLHGR	Maximum Linear Heat Generation Rate
NA	Not Applicable
NRC	Nuclear Regulatory Commission
O&M	Operation and Maintenance
PANAC11	PANACEA, GEH BWR Core Simulator
TIP	Traversing In-core Probe

1. INTRODUCTION

A GT is a simple device for measuring the thermal effects of intense gamma ray fields. GTs show great promise in simplifying BWR in-core instrumentation for nuclear plants by providing an economical alternative to the Traversing In-core Probe (TIP) system.

The TIP system, which may be based on either gamma or neutron sensitive detectors, provides measurements for (1) calibrating the Local Power Range Monitors (LPRMs) and (2) to adapt (improve) the power distribution for CMS used for calculating thermal parameters. Although TIP systems fulfill the intended purpose quite satisfactorily, the Operating and Maintenance (O&M) costs are high. This is primarily due to the many moving parts in the system, the complex under-vessel tubing which must often be disconnected during maintenance, and the radiation dose to maintenance personnel in the area of the stored TIP probes. A GT system, on the other hand, fulfills the functions of the TIP system, yet it has no moving parts, no under vessel tubing, virtually no radiation dose to maintenance since it is a fixed in-core probe, and is expected to be very reliable. However, as shown in Chapter 9, slightly larger uncertainties in monitoring LGHR margins are incurred as the number of measured axial locations is reduced.

A Gamma Thermometer based instrument system has been selected in place of a TIP system for the Economic Simplified Boiling Water Reactor (ESBWR) and is likely to be incorporated in future BWR designs and possibly retrofitted in existing plants.

1.1 Purpose

The purpose of this document is to supply regulators (and others) with sufficient information to understand the technical features of the GT system and to confirm the suitability of the GT system for LPRM calibration and power distribution monitoring.

1.2 Scope of Review

GEH requests that the NRC approve the GT system for calibration of LPRMs and for supplying power shape information to the core monitoring system.

1.3 Description of a Gamma Thermometer

A GT is a device used for measuring the gamma flux in a nuclear reactor. A section of a typical GT string and a GT sensor is shown in Figure 1-1. It consists of a stainless steel rod that has short sections of its length thermally insulated from the reactor coolant that flows within the LPRM/GT assembly. The insulation, normally a chamber of Argon gas, allows the temperature to rise in the insulated section in response to gamma energy deposition. A two-junction thermocouple measures the temperature difference between the insulated and non-insulated sections of the rod as shown in Figure 1-2. The thermocouple reading is thus related in a straightforward way to the gamma flux. When properly adjusted for the number and spectrum of the gamma rays produced from fission and neutron capture, the fission density in the surrounding

fuel can be inferred from the gamma flux and therefore indirectly from the thermocouple reading.

An ohmic heater wire is placed in the center of the GT rod to provide a means of calibrating the thermometers, which are expected to decline slowly in sensitivity during the first few months of operation. The calibration procedure consists of passing a known current through the heater wire and noting the increase in the sensor response. When properly calibrated in this manner, the GT can perform both of the normal functions of a TIP: calibration of the LPRMs and providing power shape information to the plant monitoring system.

A diagram of a LPRM/GT assembly that is designed to replace a standard LPRM/TIP assembly is shown in Figure 1-3. The LPRM/GT assembly consists of a GT rod with seven GT sensors and the normal compliment of four LPRMs.

1.3.1 GT Material Details

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Figure 1-1, Cross Section of a Typical Gamma Thermometer (dimensions are typical)

Note: Units are in inches.

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Figure 1-2, Heat Flow Path of GT

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Figure 1-3, Typical LPRM/GT Assembly

1.4 Advantages of the GT System

The GT System has many advantages over the TIP system:

- No moving parts to wear out
- Reduced occupation exposure
- Reduced radioactive waste
- No open tube penetrations to the containment
- Reduced space requirements in the reactor building
- More frequent LPRM calibrations
- More frequent adaptive monitoring calculations

2. GT SYSTEM REQUIREMENTS

2.1 Design Requirements

There are two general functional requirements for the GT system:

- Provide accurate information for the calibration of the LPRMs. Such calibration must occur as determined by individual plant technical specifications, although it may occur much more frequently, even multiple times in a single day, utilizing the GT system. This function is only performed during steady state core conditions.

- Provide axial shape information for use by the core monitoring system. This information is only utilized during steady state core conditions.

2.2 Codes and Standards

LPRM/GT assemblies must be designed, manufactured and tested according to all applicable U.S. Codes and Standards.

When manufactured for use in a nation other than the U.S., LPRM/GT assemblies must also comply with that nation's codes and standards.

3. GT SYSTEM DEFINITION

3.1 Typical Hardware Components

The hardware configuration for a typical GT core monitoring system is shown in Figure 3-1. The hardware includes new components such as the LPRM/GT assemblies, a control cabinet, the Data Acquisition System (DAS) and the Heater Power Supplies (HPS). A listing of the new components is provided in Table 3-1. Also listed is the number of each type of component that is required for a typical system.

Table 3-1, GT Core Monitoring Component List

Component Name	Number (*Typical*)
LPRM/GT Assemblies	64
GT Sensors per Assembly	7
Data Acquisition System Cabinets	2
Heater Power Supplies	16
GT Control Cabinet (including Work-Station)	1

3.1.1 LPRM/GT Assembly

A typical LPRM/GT assembly is shown in Figure 1-3. The LPRM/GT assemblies are similar to standard LPRM assemblies: each has four LPRMs and is designed to meet all of the normal requirements. In the ESBWR configuration, each assembly has one GT string with seven GT sensors. One GT sensor is positioned adjacent to each LPRM and one midway between each pair of LPRMs. In alternate designs nine GT sensors are used and positioned as follows: one adjacent to each LPRM, one midway between each pair of LPRMs, one midway between the bottom of the core and the lowest LPRM and finally, one midway between the highest LPRM and the top of the core. With the four GT sensor configuration, a GT sensor is positioned adjacent to each LPRM.

The environmental design ratings, including the operating temperature, neutron flux, gamma dose rate and seismic loadings are the same as the standard LPRM assemblies. The GT sensors are designed to last at least as long as the LPRMs.

For demonstration systems only, a calibration tube is included in the assembly so that the TIP system remains operational. This permits a direct comparison between the TIP and GT systems.

3.1.2 Data Acquisition System

The main function of the DAS is to transform the GT readings from an analog signal to a digital value. The environmental design ratings are similar to other electronics in the reactor building. In addition, the DAS system may perform digital filtration to remove noise.

3.1.3 Heater Power Supply

The purpose of the HPS is to provide a DC electrical current to the internal GT heaters during calibration. The current must be of sufficient magnitude to allow an accurate calibration of each GT sensor. The HPS, for economic reasons, may be multiplexed such that several GT assemblies are serviced sequentially by a single power supply. In Figure 3-1, the HPS is included in the DAS cabinets.

3.1.4 GT Control Cabinet

The GT Control Cabinet contains an Engineering Work-Station (EWS) that is the principal interface to the GT system. The EWS communicates with the other units including the DAS, ATLM, & 3D MONICORE™ cabinets through the plants communications system.

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Figure 3-1, Typical GT Core Monitoring System Configuration

3.2 Typical Software Components

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3.2.1 GT Monitor Module

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3.2.2 GT Calibration Module

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3.2.3 3D Simulator

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3.2.4 User Interface

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Figure 3-2, Typical GT Core Monitoring Software Diagram

4. GT RESPONSE MODEL

4.1 GT Response to Gamma Energy Deposition

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4.2 GT Factory Calibration

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4.3 GT In-Plant Calibration

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4.4 Void Fraction Response and Bypass Subcooling

4.4.1 Void Fraction Response

Voiding is not expected to occur within the LPRM/GT assembly during normal operating conditions. However, an analysis was performed to address the sensitivity of the GT response to voiding. The GT detects gamma rays from different energy spectra, and the total gamma cross section generally increases exponentially by atomic numbers. The gamma attenuation effect is less sensitive to the bypass void. In order to confirm this effect, the energy deposition into the GT sensors is compared to two different bypass void conditions in the gamma transport calculation of Monte Carlo nuclear code. A typical GE14E lattice is selected where the level-D LPRM located assuming the maximum [[ ]] bypass void with [[ ]] in-channel instantaneous void under a [[ ]] void history. This condition is consistent with the feedwater temperature/power operating domain for ESBWR. The other compared case has no bypass void with the same in-channel void condition above. The comparison includes the effect of different gamma source distributions due to different bypass void conditions. The relative difference of the energy deposition in GT sensors is [[ ]].

Another investigation of GT sensitivity to voids in the instrument tube is documented in the literature [5]. As described in this reference, Toshiba has performed thermo-hydraulic testing on an experimental GT assembly using the Multiple-Use Safety Experimental (MUSE) Facility. The MUSE facility can simulate BWR core conditions of saturation temperature, saturation pressure, and void fraction. The test objective was to confirm the GT sensitivity to changes in

fluid conditions. In the test, an inlet heater was used to produce voiding in the test piece. The test piece was manufactured to the same specifications as the in-plant test specimen except for the length. Figure 4-1 (extracted from Reference [5]) shows the relationship between heating power and GT output signal. Even for the highest void fraction of 55% achieved in the tests, the change in the GT output signal is negligible for all the heating powers that were tested.

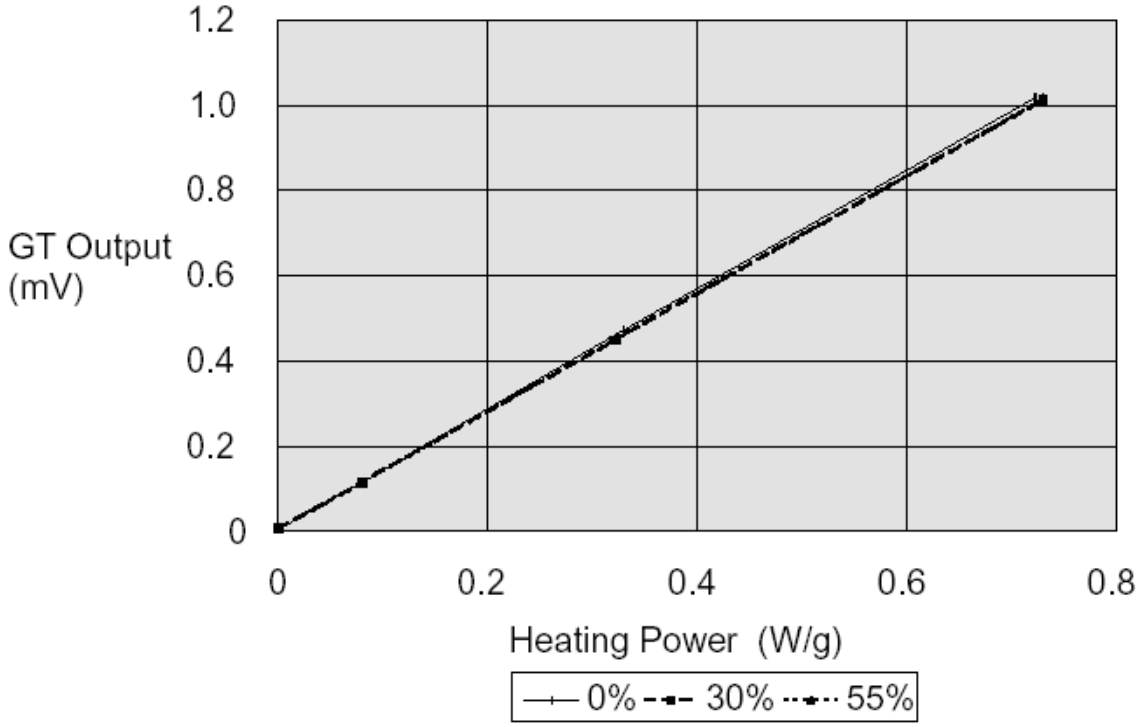

Figure 4-1, GT Output Signal vs. Heating Power Under Varying Void Fraction

* %Void Fraction

4.4.2 Bypass Subcooling

In addition to the utilization of control rods to control reactor power, the ESBWR will implement Feedwater Temperature control. However, prior to LPRM calibrations specific [[

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4.5 Delayed Gamma Compensation

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* Average thermal time constant (Joule method) of the detectors in the Joint Study

5. GT SYSTEM FUNCTIONS

The two functions of the GT System, LPRM calibration and power shape monitoring, are briefly described in the following sections.

5.1 LPRM Calibration

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5.2 Core Monitoring with GT Adaption

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6. FACTORY TESTS

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6.1 GT Factory Tests

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6.2 LPRM/GT Assembly Factory Tests

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7. IN-PLANT QUALIFICATION TESTS

There have been three in-plant tests of GT sensors in BWRs thus far. The first test was at Limerick 2 and lasted for two cycles, a total of four years. The second test, which was at Tokai 2, lasted for a single cycle of one year duration. These two tests will be described in detail in sections 7.1 and 7.2 , respectively. Published data from a third test at Kashiwazaki-Kariwa 5 are available in the open media and are summarized in section 7.3.

7.1 Limerick 2 In-Plant Test

The Limerick 2 plant, operated by Exelon (formerly PECO Energy) is an 1100 MWe BWR4. It has 764 bundles arranged in a "C" lattice configuration (equal water gaps). It has a gamma sensitive TIP system with a total of 43 calibration tubes and associated LPRM strings.

7.1.1 Test Plan

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Figure 7-1, Limerick 2 Core Map

7.1.2 Hardware Description

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Figure 7-2, Heater Current During GT Calibration

7.1.3 GT Calibration Results

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Figure 7-3, GT 1 Sensitivity

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Figure 7-4, GT 2 Sensitivity

7.1.4 Comparison with Gamma TIPs

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Table 7-1, Statistical Differences between GT and TIP Readings

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Figure 7-5, GT 1 vs. Gamma TIP

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Figure 7-6, GT 2 vs. Gamma TIP

7.1.5 Conclusions

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7.2 Tokai 2 In-Plant Test

The Tokai 2 plant, operated by Japan Atomic Power Company (JAPC), is an 1100 MWe BWR5 with 764 fuel bundles arranged in a C lattice. It has a neutron sensitive TIP system with a total of 43 calibration tubes.

7.2.1 Test Plan

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Figure 7-7, GT Rod

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Figure 7-8, Tokai 2 Core Map

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7.2.2 Hardware Description

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7.2.3 GT Calibration Results

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Figure 7-9, Fixed Alpha GT Sensitivity for RSTK-01

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Figure 7-10, Fixed Alpha GT Sensitivity for RSTK-02

7.2.4 Deleted

7.2.5 Steady State Response

7.2.5.1 Comparison with Neutron TIPs

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Table 7-2, Difference (%)* between GT and TIP Readings for RSTK-01**

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Table 7-3, Difference (%)* between GT and TIP Readings for RSTK-02**

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Table 7-4, Difference (rd)* between GT and TIP Readings for RSTK-01**

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Table 7-5, Difference (rd)* between GT and TIP Readings for RSTK-02**

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Table 7-6, Difference (rd)* between GT and TIP Four-Bundle Powers for RSTK-01

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Table 7-7, Difference (rd)* between GT and TIP Four-Bundle Powers for RSTK-02

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[[ ]]

Table 7-8, Summary of RMS (rd)* Differences between GT and TIP Four-Bundle Powers and Readings**

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Figure 7-11, Comparison of GT with TIP Readings for RSTK-01

Figure 7-12, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-13, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-14, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-15, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-16, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-17, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-18, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-19, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-20, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-21, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-22, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-23, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-24, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-25, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-26, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-27, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-28, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-29, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-30, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-31, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-32, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-33, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-34, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-35, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-36, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-37, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-38, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-39, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-40, Comparison of GT with TIP Readings for RSTK-0

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Figure 7-41, Comparison of GT with TIP Readings for RSTK-0

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Figure 7-42, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-43, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-44, Comparison of GT with TIP Readings for RSTK-02

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Figure 7-45, Comparison of GT with TIP Readings for RSTK-01

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Figure 7-46, Comparison of GT with TIP Readings for RSTK-02

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7.2.5.2 Comparison with LPRMs

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Figure 7-47, Comparison of GT with LPRM Readings for RSTK-01 (steady-state)

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Figure 7-48, Comparison of GT with LPRM Readings for RSTK-02 (steady-state)

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7.2.6 Response During Non-Steady State Conditions

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7.2.6.1 Response During Startup

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Figure 7-49, Comparison of GT with LPRM Readings for RSTK-01 (startup)

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Figure 7-50, Comparison of GT with LPRM Readings for RSTK-02 (startup)

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7.2.6.2 Response to Flow Change

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Table 7-9, Comparison of Un-Compensated GT with LPRM Readings for RSTK-01 (flow change)

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Table 7-10, Comparison of Un-Compensated GT with LPRM Readings for RSTK-02 (flow change)

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Table 7-11, Comparison of Delayed Gamma Compensated GT with LPRM Readings for RSTK-01 (flow change)

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Table 7-12, Comparison of Delayed Gamma Compensated GT with LPRM Readings for RSTK-02 (flow change)

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Figure 7-51, Un-Compensated GT vs. LPRM Readings for RSTK-01 (flow change)

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Figure 7-52, Un-Compensated GT vs. LPRM Readings for RSTK-02 (flow change)

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Figure 7-53, Delayed Gamma Comp. GT vs. LPRM Readings for RSTK-01 (flow change)

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Figure 7-54, Delayed Gamma Comp. GT vs. LPRM Readings for RSTK-02 (flow change)

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7.2.6.3 Response to Control Rod Movement

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Table 7-13, Comparison of Un-Compensated GT with LPRM Readings for RSTK-01 (control blade pattern change)

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Table 7-14, Comparison of Un-Compensated GT with LPRM Readings for RSTK-02 (control blade pattern change)

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Table 7-15, Comparison of Delayed Gamma Compensated GT with LPRM Readings for RSTK-01 (control blade pattern change)

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Table 7-16, Comparison of Delayed Gamma Compensated GT with LPRM Readings for RSTK-02 (control blade pattern change)

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Figure 7-55, Un-Compensated GT vs. LPRM Readings for RSTK-01 (CB move)

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Figure 7-56, Un-Compensated GT vs. LPRM Readings for RSTK-02 (CB move)

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Figure 7-57, Delayed Gamma Comp. GT vs. LPRM Readings for RSTK-01 (CB move)

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Figure 7-58, Delayed Gamma Comp. GT vs. LPRM Readings for RSTK-02 (CB move)

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7.2.6.4 Response During Power Down

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Figure 7-59, Un-Compensated GT vs. LPRM Readings for RSTK-01 (power down)

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Figure 7-60, Un-Compensated GT vs. LPRM Readings for RSTK-02 (power down)

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Figure 7-61, Delayed Gamma Comp. GT vs. LPRM Readings for RSTK-01 (power down)

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Figure 7-62, Delayed Gamma Comp. GT vs. LPRM Readings for RSTK-02 (power down)

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Figure 7-63, Un-Compensated GT Linearity for Sensor 2 at RSTK-01 (power down)

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Figure 7-64, Delayed Gamma Comp. GT Linearity for Sensor 2 at RSTK-02 (power down)

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7.3 Kashiwazaki-Kariwa 5 In-Plant Test

The Kashiwazaki-Kariwa 5 plant is an 1100 MWe BWR5 operated by Tokyo Electric Power Company. The core has 764 bundles arranged in a "C" lattice configuration. The TIP system is neutron sensitive and has 43 calibration tubes and associated LPRM strings.

The research reported here was sponsored jointly by Tokyo Electric Power Company, Tohoku Electric Power Company, Chubu Electric Power Company, Hokuriku Electric Power Company, The Chugoku Electric Power Company, The Japan Atomic Power Company, Toshiba Corporation, Hitachi, Ltd. and Global Nuclear Fuel – Japan (see reference 4).

7.3.1 Test Plan

The test plan was to install 8 LPRM/GT assemblies with 9 GT sensors each (4 each from two separate suppliers) into an octant of the core. A comprehensive core monitoring study would therefore be possible, subject only to the condition of octant symmetry. Standard TIP calibration tubes were installed in the LPRM/GT assemblies so that normal TIP set measurements could be taken.

In order to assess the accuracy of core monitoring with GT readings, a bundle gamma scan was to be performed at the end of one cycle of operation. The scan was to include all of the bundles in the octant of the core as well as three additional bundles chosen so that all four bundles around each LPRM/GT assembly were included.

In addition, comparisons of core monitoring results between the GT and the neutron TIP systems were to be made throughout the cycle.

The LPRM/GT assemblies were to have nine sensors each, arranged in a manner similar to the Tokai 2 In-Plant test. The core locations for the assemblies are shown in Figure 7-65.

[[

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Figure 7-65, Kashiwazaki-Kariwa 5 Core Map

7.3.2 Test Results

The gamma scan measured the strong photo peak of ^{140}La the short-lived daughter of the common fission product ^{140}Ba. Measurements were taken at 17 axial locations, generally every six inches, but excluding nodes obscured by spacers and excluding the top and bottom nodes. The ^{140}Ba distribution was calculated by an off-line core monitoring system throughout the cycle on a time interval of every 3 days. TIP adaptive core monitoring calculations were made 36 times during the cycle for the standard simulation. An equal number of GT adaptive calculations were made for the GT core monitoring simulation. Data were available only for seven of the eight GT assemblies[2], but this did not significantly detract from the results of the study.

The results showed very good agreement between the calculated and measured ^{140}Ba distributions for both the GT adaptive monitoring and the TIP adaptive monitoring. Table 7-17 shows the RMS differences between the calculated and measured distributions as measured for the 1D (axial) distribution, for the 2D (bundle) radial distribution and for the 3D (nodal) distribution.

Table 7-17, RMS Differences between Calculated and Measured ^{140}Ba Distributions

Distribution	n TIP vs. γ Scan	GT vs. γ Scan
1D (axial)	1.7%	2.1%
2D (bundle)	2.5%	2.3%
3D (nodal)	3.9%	4.1%

In addition to the gamma scan studies, comparisons were also made between the thermal limits calculated by the two core monitors. The RMS difference for the whole cycle between the MCPR calculated by the GT core monitor and the MCPR calculated by the neutron TIP monitor was 0.008. The maximum difference was 0.02.

Similarly, the RMS difference between the MLHGR calculated by the GT core monitor and the MLHGR calculated by the neutron TIP was 0.4 KW/m and the maximum difference was 1 KW/m.

[2] According to a personal communication, one of the assemblies planned for the test experienced a failure that affected all nine sensors in the assembly. Modern core monitors such as 3DMonicore are tolerant of this type of fault and are able to produce accurate results with several assemblies out of service.

7.3.3 Conclusions

The comparison with the gamma scan established that core monitoring based on GTs is nearly equivalent in accuracy to core monitoring with neutron TIPs. In addition, it was shown that the thermal limits, MCPR and MLHGR, evaluated by the two core monitoring systems were very similar throughout the cycle.

The overall conclusion was that the GT system is “practical as a substitute” for the TIP system.

8. ADAPTIVE CORE MONITORING SIMULATION

One of the most important functions of the GT system is to provide information to the Core Monitoring System for power shape monitoring and thermal margin determination. Most modern CMSs, such as 3D MONICORE™, [[

]] The uncertainty analysis is addressed in Chapter 9.

8.1 The TIP-based Monitoring System Functions

[[

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8.2 The Difference of the GT-based Monitoring System

[[

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8.3 Simulated GT Readings Study

In order to analyze the GT system performance as a core monitoring system, a study was carried out using core-wide simulated GT readings to adapt PANAC11 calculated power shapes of two BWRs where measured TIP data was collected (and selected as reference). The results are used as the basis for discussion of GT-related uncertainties in the framework of Reference 3 (SLMCPR licensing topical report). This study addresses expected ESBWR conditions, namely, the [[

]].

A modified version of [[ ]] for the GT system was created and tested to determine the GT-associated uncertainties that are described in Chapter 9. During the process, superseding information was generated that updates the responses to NRC RAI 4.2-12 S02 parts 10 and 11 as explained in the response to NRC RAI 7.2-18 S02 (references 6 and 11, respectively).

In the following subsection, the description of the reactor cores, the GT adaption technique and the description of the boundary conditions used in this study are presented.

8.3.1 Description of the Reactor Cores

The reactors selected have a power density greater than [[ ]]. One of the selected reactors is a [[ ]] and has [[ ]] and is referred to as Plant E. BOC, MOC and EOC calculated power distributions were obtained with [[ ]] and compared to measured power distributions [[ ]]. The second reactor selected is a [[ ]] referred to as [[ ]]. Its calculated power distribution was compared to the measured power distribution [[ ]] was used as the [[ ]] for both reactor power distributions. The GT readings were assumed as equal to the corresponding TIP reading (either thermal-TIP or gamma-TIP) at the GT sensor locations for the core-wide simulations.

8.3.2 GT Adaption Methodology

The selected GT-adaption method for the ESBWR monitoring system [[

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[[ ]]

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[[

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8.3.3 Definition of Boundary Conditions

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8.3.4 Additional Noise Component

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8.3.5 GT Adaption Remarks

The implementation of the GT signals into PANAC11 adaptive process and the numerical examples studied in two reactors resulted in the following remarks:

- By virtue of the shape adaption, the radial uncertainty is [[
  ]].
  Core average exposure [[

-]]

8.4 Detector Response

[[

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8.5 Detection of Anomalies and Signal Perturbations

The ramification of having an anomaly in one axial node that perturbs the power distribution locally can be described in terms of the information handled by the power monitoring system and for the actions that can be taken in-situ or previously to the core loading.

In case of unforeseen anomalies, [[

]]

Among potential perturbations that could occur in normal reactor operation and produce a signal bias, [[

]]

Consider that the ESBWR GT is located in the region between four fuel assemblies in the corner opposite from the control blade location. The spacer material [[

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[[

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8.6 Form functions for Spacer effect and Fuel Lattice Change

The generic expression for [[

]] are calculated by PANAC11 using TGBLA information.

8.6.1 Linear Model of CALGT calculation

[[

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8.6.2 Spacer Effect on CALGT calculation

In the TIP-based monitoring system, [[

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[[

]]

The calculated power at the GT sensor location will be [[

]]

[[

]]

8.6.3 Fuel Lattice Change Effect on CALGT calculation

For cases when [[

]]

Finally, Figure 8-6 represents [[

]]

9. UNCERTAINTY ANALYSIS

[[

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9.1 GT Sensor Accuracy

The ESBWR GT sensor is [[

]]

Table 9-1, SRS Requirements (Specific for the Tokai 2 Plant Test)

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				]]

[[

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Table 9-2, GT Sensor Accuracy Criteria

[[				
				]]

9.2 GT Calibration Accuracy

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[[

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Table 9-3. Evaluation of GT Calibration Accuracy based on Tokai-2 Plant Test

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9.3 GT Adaptive Core Monitoring Accuracy

The power uncertainties associated to the GT-based monitoring system are analyzed in the following sections. The uncertainty components correspond to the list of uncertainties described in Reference 3. The analyses are based on simulated GT readings and the purpose is to establish the uncertainties applicable to the MCPR and LHGR calculations for the ESBWR. At the end of the Chapter, the list of LHGR uncertainties is presented. Note that there are some uncertainties that are not related to the GT-based monitoring system but they are presented for completeness.

[[

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[[ ]]

A specific analysis was performed to establish the uncertainties caused by the LPRM calibration update and potential failure of GT instruments for the calculation of the Critical Power Ratio (CPR) and Linear Hear Generation Rate (LHGR). The analysis uses [[

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The study relies [[ ]] to simulate GT sensor signals. Plant M, [[

]]

The update uncertainty for the LPRM calibration is studied in a core-average exposure interval of [[

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This plant is monitored with [[

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[[

]] for the current BWR fleet (NEDC-32694P-A, Reference 3). It is expected that [[

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[[

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9.3.1 MCPR Uncertainty

The following subsections present the applicable ESBWR uncertainties of the MCPR calculation.

9.3.1.1 Four Bundle Power Uncertainty

[[

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9.3.1.2 Power Allocation Uncertainty

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9.3.1.3 LPRM Calibration Update Uncertainty

Figure 9-1 presents [[

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[[ ]]

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9.3.1.4 MCPR GT Failure Uncertainty

To determine the new instrument failure uncertainty for ESBWR with GT sensors, [[

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9.3.1.5 Total MCPR Update/Failure Uncertainty

For the GT application, the total update/failure uncertainty for the MCPR calculation is given as [[

]]

To obtain the total update/failure uncertainty for the ESBWR-MCPR calculation, the following formula is applied (this assumes independence among the components and separation of the effects):

[[

]]

This is [[ ]] the value presented in NEDC-32694P-A (Reference 3) for the update/failure uncertainty (0.62%) for the TIP-based system [[ ]]. Table 9-5 presents a comparison of uncertainties for the current BWR fleet and for the ESBWR.

[[

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[[

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9.3.2 LHGR Uncertainty

The following subsections present the applicable ESBWR uncertainties of the LHGR calculation.

9.3.2.1 Interpolation Uncertainty

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[[ ]]

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9.3.2.2 Random Uncertainty

[[

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9.3.2.3 LHGR GT Update Uncertainty

Table 9-7 provides more detail on the [[

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It is worth to note that the performance of the update process for the GT-based LPRM calibration is very similar to the performance of the update process for the TIP-based LPRM adaption.

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9.3.2.4 LHGR GT Failure Uncertainty

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9.3.2.5 Summary of Uncertainties for LHGR Calculation

A [[ ]] is applied in the development of the LHGR limit. As shown in the Table 9-8, [[ ]] (from [[ ]] to [[ ]]) and the [[ ]] (from [[ ]]) results in total power distribution uncertainty [[ ]] that is currently applied to the development of the LHGR limit for equilibrium core loadings. Note that the gradient effect's uncertainty decreased from [[ ]] based on [[ ]].

Table 9-8 collects [[

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The following [[

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9.3.3 Minimum Monitoring Configuration

[[

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If those conditions are met, the total update/failure uncertainty for the MCPR and LHGR calculations will [[
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[[ ]]

[[

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10. CONCLUSIONS

The Gamma Thermometer System has been successfully evaluated as a replacement for the TIP system by a comprehensive In-Plant Test Program. The primary objectives of the Test Program have been met:

1. GT sensor accuracy relative to gamma TIP, neutron TIP and LPRM measurements has been evaluated; and

2. GT sensor reliability under BWR operating conditions has been established.

The GT sensitivity trends have been followed throughout a total of three cycles of operation at two BWRs. The sensitivity trend in the most recent test consisted of a relatively rapid initial rise during the first 500 hours of operation, followed by a slow decline for the rest of the cycle. However, in-plant calibrations provide the ability to adjust for any sensitivity changes.

The GT response in the steady state has been compared with gamma TIP and neutron TIP response as well as the LPRM response. The GT response during changing conditions (startup, flow change, control blade change and power down) has been compared with the LPRM response. The GT sensors were evaluated for accuracy with a combination of factory, in-plant and core monitoring tests:

1. For the Tokia 2 test the factory tests proved that the GT sensors met all of the requirements of the SRS.

2. The in-plant tests proved the accuracy, linearity and range of the GT sensors with respect to the TIPs and LPRMs.

3. Chapters 8 and 9 present the conclusions of the power uncertainty analysis for the GT-based monitoring system in its implementation to the ESBWR for LPRM calibration and power adaption in the framework of the SLMCPR and LHGR methodology approved in NEDC-32694P-A (Reference 3). The development of this work is presented in references 6 and 11.

The overall conclusion of the GT in-plant test program is:

The GT system can be used in place of a TIP system for both LPRM calibration and power shape monitoring.

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Attachment 1

NRC SAFETY EVALUATION
Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring

SAFETY EVALUATION FOR NEDC-33239P, "GE14 FOR ESBWR NUCLEAR DESIGN REPORT," AND NEDE-33197P, "GAMMA THERMOMETER SYSTEM FOR LPRM CALIBRATION AND POWER SHAPE MONITORING," LICENSING TOPICAL REPORTS FOR REFERENCE IN THE ECONOMIC SIMPLIFIED BOILING-WATER REACTOR DESIGN CERTIFICATION APPLICATION

EXECUTIVE SUMMARY

The staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed the information contained in licensing topical reports (LTRs) NEDC-33239P, "GE14 for ESBWR Nuclear Design Report," and NEDE-33197P, "Gamma Thermometer System for LPRM Calibration and Axial Power Shape Monitoring," for application to the economic simplified boiling-water reactor (ESBWR). These LTRs describe core cycle analysis and online monitoring methodologies.

In general, core analysis methods are based on a combination of thermal hydraulic, neutronic, and gamma transport models. The combination of these models and codes provide the basis for the prediction of steady-state operational conditions, as well as the analysis input to transient calculations. In particular, the methods for core monitoring determine safety and operational limit margins and provide a means for correcting theoretical predictions of core cycle exposure behavior to instrument measurements.

Although the NRC staff has previously reviewed most of these methods for currently operating reactors, the staff has not previously reviewed these methods as applied to the ESBWR in view of its unique design features. These features include the unique fuel design, the natural circulation design and consequent range of in-core void fractions, the unique core monitoring calibration technology, and the use of a new core adaption methodology.

The staff evaluated the efficacy of these methods to demonstrate compliance with general design criteria using Section 4.3, "Nuclear Design," of the Standard Review Plan (SRP) and applicable guidance from SRP Section 15.0.2, "Review of Transient and Accident Analysis Methods." In the safety evaluation, the staff identified several conditions, limitations, and restrictions associated with the suite of methods that comprise the nuclear design and core monitoring methods. In general, these conditions, limitations, and restrictions are needed to ensure that the execution of the methodology does not invalidate the uncertainty assessment and that adequate margins are applied to ensure safety.

When the nuclear design codes and the core monitoring software were exercised within the bounds of the aforementioned conditions, the staff concluded that the methodologies were acceptable (as limited) to calculate results and to compare those results to the acceptance criteria. Therefore, the staff finds the nuclear design and core monitoring methodologies to be acceptable for reference in the ESBWR design certification application, potential future reload licensing and potential future core monitoring.

Enclosure 1

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1. INTRODUCTION

GE- Hitachi Nuclear Energy Americas LLC (GEH or the applicant) submitted to the U.S. Nuclear Regulatory Commission (NRC) NEDC-33239P, "GE14 for ESBWR Nuclear Design Report" (Ref. 1), and NEDE-33197P, "Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring" (also referred to as the Gamma Thermometer Licensing Topical Report (GT LTR)) (Ref. 2), for staff review as part of the design certification application review for the economic simplified boiling-water reactor (ESBWR). The methods and design information in these proprietary licensing topical reports (LTRs) provide the basis for information included in the ESBWR Design Control Document (DCD) Tier 2, Section 4.3, "Nuclear Design." GEH is seeking U.S. Nuclear Regulatory Commission (NRC) approval of these LTRs for reference in the ESBWR design certification application. These reports describe the generic core nuclear design and core monitoring methods. Therefore, the staff reviewed the generic applicability of these LTRs to the ESBWR.

2. REGULATORY BASIS

As required by Title 10, Section 52.47(a)(4), of the *Code of Federal Regulations* (10 CFR 52.47(a)(4)), "Contents of Applications; Technical Information," an applicant for certification of a standard design must provide a final safety analysis report (FSAR) to the NRC that describes, among other things, the performance of structures, systems, and components (SSCs) provided for the prevention or mitigation of potential accidents. The applicant is seeking generic approval of the TGBLA06/PANAC11 code suite to perform licensing analyses for the ESBWR.

ESBWR DCD Tier 2, Section 4.3, presents the ESBWR nuclear design bases. In general, as required by the General Design Criteria (GDC) in 10 CFR Part 50, Appendix A, the nuclear design must ensure that the specified acceptable fuel design limits (SAFDLs) will not be exceeded during normal operation, including anticipated operational occurrences (AOOs), and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core or sustain unstable core conditions. Specifically, the nuclear design must conform to the following GDC:

- GDC 10, "Reactor Design," requiring the reactor design (reactor core, reactor coolant system, control and protection systems) to ensure that the SAFDLS are not exceeded during any condition of normal operation, including AOOs

- GDC 11, "Reactor Inherent Protection," requiring a net negative prompt feedback coefficient in the power operating range

- GDC 12, "Suppression of Reactor Power Oscillations," requiring that power oscillations that can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed

- GDC 13, "Instrumentation and Control," requiring a control and monitoring system to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions

- GDC 20, “Protection System Functions,” requiring, in part, a protection system that automatically initiates a reactivity control system to ensure that SAFDLs are not exceeded as a result of AOOs

- GDC 25, “Protection System Requirements for Reactivity Control Malfunctions,” requiring protection systems designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems

- GDC 26, “Reactivity Control System Redundancy and Capability,” requiring, in part, two independent reactivity control systems of different design principles, one of which is capable of holding the reactor subcritical under cold conditions

- GDC 27, “Combined Reactivity Control Systems Capability,” requiring, in part, that the reactivity control systems be designed to control reactivity changes during accident conditions in conjunction with poison addition by the emergency core cooling system

- GDC 28, “Reactivity Limits,” requiring, in part, that the reactivity control systems be designed to limit reactivity accidents so that the reactor coolant system boundary is not damaged beyond limited local yielding

DCD Tier 2 provides analytical results to support compliance of the ESBWR with the above GDCs. The staff reviewed the methodologies in supporting topical reports, NEDC-33239P (Ref. 1) and NEDE-33197P (Ref. 2), to evaluate the efficacy of the proposed methodology to produce acceptable results for reference in the ESBWR DCD. The staff reviewed information in these LTRs and GEH responses to the staff’s requests for additional information (RAIs). The staff determined that these LTRs are acceptable for reference, as documented in the following sections.

The staff conducted its review of the associated topical reports in accordance with Standard Review Plan (SRP) Section 4.3, “Nuclear Design,” (Ref. 4) and SRP Section 15.0.02, “Review of Transient and Accident Analysis Methods” (Ref. 5)

3. TECHNICAL EVALUATION

3.1 Introduction

The subject LTR’s described the nuclear design methodology (TGBLA06/PANAC11), as well as the gamma thermometer (GT)-based core monitoring system (CMS). The TGBLA06 and PANAC11 codes are used to perform cycle safety analysis, and they form the calculational engine of the GT CMS. The ESBWR nuclear instrumentation differs slightly from the current operating fleet in that the GT CMS replaces the traversing in-core probe (TIP) system. This unique design feature of the ESBWR calls for augmentation of the core monitoring methods to support the GT design and warrants staff review of the instrument design and updated core monitoring methods.

This safety evaluation (SE) divides the staff review into three sections. The first section documents the staff review of the analytical capabilities of the nuclear design code suite. The second section documents the staff review of the GT-based CMS. The third section documents the efficacy of the codes to predict and monitor thermal margin.

3.2 Nuclear Design Methods (TGBLA06 and PANAC11)

TGBLA06 and PANAC11 form the nuclear design code suite applied to the ESBWR nuclear design safety analysis. TGBLA06 is a two-dimensional lattice physics code, while PANAC11 is a three-dimensional nodal diffusion code. TGBLA06 generates nuclear data that are utilized by PANAC11 to calculate the reactor core power distribution, eigenvalue,[1] control blade worth, and other core nuclear characteristics.

PANAC11 forms the basis for the CMS software 3D MONICORE. 3D MONICORE utilizes the PANAC11 computational engine and live plant data to monitor the core local power distribution and thermal margin during cycle operation.

3.2.1 Background and Previous NRC Review

The TGBLA06/PANAC11 steady-state nuclear methods are based on the TGBLA04/PANAC10 methods that General Electric (GE) (now GEH) submitted for NRC staff review in July of 1983, and which were subsequently approved by the staff in December of 1983 (Ref. 6). Several models were upgraded to form the TGBLA06/PANAC11 code suite. GEH implemented the new suite in 1996 (Ref. 7). Subsequent to the implementation, the staff conducted a review of the improved steady-state methods to a proposed extended power uprate (EPU) on a plant-specific basis for Vermont Yankee (Ref. 8). Based on this review, GEH submitted an LTR detailing the applicability of these methods (as well as others) to safety analyses for EPUs and maximum extended load line limit analysis plus (MELLLA+) plants (Ref. 9). This LTR (NEDC-33173P—also referred to as the Interim Methods Licensing Topical Report (IMLTR)) details the interim methods process.

During the conduct of its review, the NRC staff identified concerns regarding the application of the GEH nuclear design code suite to EPU and MELLLA+ plants. Many of these concerns stem from the harder spectral conditions anticipated for EPU operation (i.e., the spectrum of energies of the neutrons in a reactor operating under an EPU is shifted higher than that of the same reactor before the EPU). Generally speaking, EPU cores operate at higher void fractions than pre-EPU cores. Additionally, to maintain the same cycle length at higher thermal powers, the fuel reload batches tend to include higher fissile uranium and gadolinia loadings. These three factors all result in a “hardening” of the core average neutron spectrum. At these harder spectral conditions, the isotopic modeling capabilities of the code become increasingly important in the prediction of the core power distribution as a result of increased plutonium production during cycle depletion.

The staff noted in its review of the IMLTR that the code system has not been qualified against gamma scan data under EPU or MELLLA+ conditions (Ref. 10). The interim methods process described in the IMLTR proposes an approach to account for potentially increased uncertainties in the power distribution until additional qualification data are supplied to the NRC (GEH uses qualification in this context to connote the development and benchmarking of analytical methods). Similarly, during the IMLTR review, the staff identified concerns regarding the void quality correlation. The staff noted that, at EPU and MELLLA+ operation, the bundle average and maximum void fractions increase relative to pre-EPU conditions. The staff approves the IMLTR in this regard subject to the condition that a penalty to the operating limit minimum critical power ratio (OLMCPR) be incorporated to address concerns regarding potentially increased void fraction uncertainties for higher void fractions (Ref. 10). These two interim

[1] In the context of GEH methods, the eigenvalue refers to the effective multiplication factor.

penalties do not constitute a full list of all of the staff's concerns regarding the GEH nuclear design codes; however, they represent significant review findings that are applicable to the subject review as these topics address high void fraction conditions for EPU operation that are similar to the ESBWR operating conditions.

Concurrent with its review of the IMLTR, the staff reviewed the migration from the TGBLA04/PANAC10/TRACG02 transient analysis methods to the updated TGBLA06/PANAC11/TRACG04 transient analysis methods for the operating fleet. GEH submitted LTR NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRAGC AOO and ATWS Overpressure Transients" (also referred to as the Migration LTR) (Ref. 12) in 2006. During its review of the IMLTR, the staff deferred the review of the applicability of the Transient Reactor Analysis Code (TRACG) methods to EPU and MELLLA+ plants to the Migration LTR. The staff reviewed the applicability of the updated nuclear codes as they are utilized in the TRACG04 transient analysis methodology. Reference 13 documents the staff's review findings.

The staff conducted a detailed review of the theoretical basis of the TGBLA06/PANAC11 codes during its review of the Migration LTR. The staff review of the models is documented in Section 3.3 of Reference 13. In its review, the staff found that the modeling approach and assumptions were reasonable and acceptable for safety analysis so long as the code uncertainties were adequately captured in the analysis. During its review, the staff found that many of the penalties imposed in the IMLTR SE were also applicable to the Migration LTR.

The staff noted that the proposed ESBWR design is a high-power density plant operating under conditions of natural circulation. This operating regime, when considered in tandem with the proposed feedwater temperature power operating domain (Ref. 14), yields high core average and high maximum void fractions in the fuel bundles. These conditions are similar to those conditions expected for EPU operation in the current operating fleet of plants. Therefore, in the subject review, the staff leveraged previous experience gained through its review of the GEH methods for EPU and MELLLA+.

3.2.2 Steady-State Calculations

GEH performed steady-state calculations using the nuclear design methods to demonstrate compliance with several of the GDCs specified in SRP Section 4.3. The staff reviewed the qualification of TGBLA06/PANAC11 to perform these steady-state calculations, any features unique to the ESBWR affecting these analyses, and the overall method for evaluating compliance with the relevant GDC.

3.2.2.1 Qualification

The nuclear design methodology qualification provided in NEDC-33239P (Ref. 1) is essentially identical to the qualification provided to the staff in the form of RAI responses during its review of the IMLTR and NEDC-33006P, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus" (also referred to as the MELLLA+ Licensing Topical Report (M+LTR)) (Refs. 9 and 15). For instance, the TIP data provided in NEDC-33239P are the same TIP data provided in response to RAIs 25 and 27 issued during the IMLTR review. The staff also utilized these qualification data in its review of the Migration LTR (Ref. 13). The staff agrees that these data are applicable because, from a neutronic and thermal hydraulic perspective, the ESBWR core is substantially similar to a large EPU core.

While the ESBWR power-to-flow ratio is substantially higher than those plants in the updated experience database, the staff noted that the ESBWR neutron spectrum is expected to be largely similar to an EPU core (as opposed to a MELLLA+ core at the low flow point) because several design features result in lower nodal average void fractions for the same power-to-flow ratio. These features include (1) the N- lattice design, (2) high inlet subcooling, and (3) shorter fuel bundles.

Therefore, the staff considered whether those conditions and limitations specified in the SE for the IMLTR should also apply to the ESBWR. The staff has identified relevant conditions and addressed their applicability to the ESBWR in the following sections.

3.2.2.1.1 Condition 4—Safety Limit Minimum Critical Power Ratio 1 from the Safety Evaluation for the Interim Methods Licensing Topical Report

As described below, the staff finds that the interim process penalty regarding the [[ ]] applies to the ESBWR. The staff noted that the OLMCPR determination process, as described in Reference 16, differs slightly from the process utilized in the operating fleet. Therefore, Condition 4 from the staff's SE for the IMLTR is not directly applicable because (1) the safety limit minimum critical power ratio (SLMCPR) is derived from the OLMCPR and (2) adding 0.02 to the SLMCPR does not result in any additional thermal margin, according to the ESBWR methodology. Accordingly, the staff finds the subject LTR acceptable in this respect only if it is subject to a modified condition that the OLMCPR be derived using a [[ ]][2] that is consistent with [[ ]] reported in the IMLTR (Ref. 9). Specifically, the LTR states this condition as follows (additional staff review of the language of this condition is documented in Appendix B of this SE).

> **[[ ]]**[3]
>
> The [[ ]] is a component of the linear heat generation rate (LHGR) and OLMCPR calculation uncertainties. Its value is determined using a [[ ]] on gamma scan data. NEDC-33173P-A reports the value determined using this approach as [[ ]].
>
> This condition applies to the ESBWR for the same reasons the NRC staff approved the [[ ]] for currently operation reactors in NEDC-33173P-A. Should the NRC approve an alternative approach for establishing the aforementioned uncertainties in subsequent supplements to or revisions of the NEDC-33173P LTR, the approved, alternative approach may be adopted in NEDE-33197P-A in lieu of this condition without separate NRC review and approval.
>
> [[ ]] of the uncertainty value must be submitted to the NRC before the change is incorporated into any safety analysis basis.

The staff is aware that GEH intends to submit qualification gamma scan data as a supplement to the IMLTR (Ref. 9). This condition is intended to ensure that, if the staff should revise the

---

2 [[ ]]

3 This condition reflects the staff SE for the IMLTR and Section 9.3.1.2 of the GT LTR and is incorporated in the LTR through the applicant's response to RAI 7.2-71.

[[ ]] in subsequent reviews of supplements to the IMLTR, the uncertainty used for the ESBWR will be consistent with the approved, revised value or maintained at the aforementioned value (which would be conservatively higher than the approved, revised value).

Condition 4 from the SE for the IMLTR is derived from a modification of the [[ ]] as well as the lattice peaking factor uncertainty. Similar to the [[ ]] condition, the interim methods process details an increased lattice peaking factor uncertainty based on the [[ ]]. Both the maximum linear heat generation rate (MLHGR) limit and the OLMCPR utilize this lattice peaking factor uncertainty. In the case of the OLMCPR, the lattice peaking factor uncertainty affects the uncertainty in the bundle R-factor. The R-factor is a parameter that characterizes the coolant-averaged radial power peaking in the bundle. Therefore, the staff approves NEDE-33197P and NEDC-33239P in this regard subject to the following two conditions:

Peaking Factor Uncertainty for MLHGR Condition[4]

The LHGR infinite lattice peaking factor uncertainty value is determined [[ ]] using the statistical analysis of the population of peak power as a function of exposure. The GE14E-specific LHGR infinite lattice peaking factor uncertainty determined using this approach is [[ ]]. This uncertainty will be determined whenever a new fuel product is applied to a particular ESBWR core loading.

This condition applies to the ESBWR for the same reasons the NRC staff approved the [[ ]] for currently operation reactors in NEDC-33173P-A. Should the NRC approve an alternative approach for establishing the aforementioned uncertainties in subsequent supplements to or revisions of the NEDC-33173P LTR, the approved, alternative approach may be adopted in NEDE-33197P-A in lieu of this condition without separate NRC review and approval.

Any reduction of the uncertainty value must be submitted to the NRC before the change is incorporated into any safety analysis basis.

Peaking Factor Uncertainty for OLMCPR Condition[5]

NEDC-32601P-A describes the method for calculating the R-factor uncertainty. When determining the R-factor uncertainty for ESBWR analyses, the infinite lattice peaking model uncertainty value will be assumed as equal to, or more conservative than, the LHGR infinite lattice peaking factor uncertainty value for a particular ESBWR core loading.

Any change of the uncertainty value must be submitted to the NRC before the change is incorporated into any safety analysis basis.

---

4 This condition is consistent with the staff SE for the IMLTR and is incorporated in the LTR through the applicant's response to RAIs 4.3-2 and 7.2-71.

5 This condition is consistent with the staff SE for the IMLTR and is incorporated in the LTR through the applicant's response to RAI 7.2-71.

The staff is aware that GEH intends to submit qualification gamma scan data as a supplement to the IMLTR (Ref. 9). These conditions are intended to ensure that, should the staff revise the lattice peaking factor uncertainty in subsequent reviews of supplements to the IMLTR, the uncertainty used for the ESBWR will be consistent with the approved, revised value or conservatively higher than the approved, revised value.

The staff noted that, for the case of the lattice peaking factor uncertainty, the response to RAI 4.3-2 S02-A provides an alternative means for calculating the value of the lattice peaking factor uncertainty on a fuel-design-specific basis using the [[ ]]. Therefore, the staff approves the subject LTRs in this regard subject to the condition on this approach to ensure that the uncertainty is evaluated over the appropriate range of fuel exposure.

Peaking Factor Uncertainty and Fuel Exposure Condition[6]

The LHGR infinite lattice pin power uncertainty must represent the full range of fuel lattice exposure values. The calculated peak pellet exposure must be confirmed by GEH or the licensee referencing the LTR to comply with the corresponding licensing limit approved by the NRC. The design analysis described in NEDC-33242P, "GE14 for ESBWR Fuel Rod Thermal Mechanical Design Report," establishes the licensing limit for GE14E (Ref. 11).

3.2.2.1.2 Condition 6—R-Factor from the Safety Evaluation for the Interim Methods Licensing Topical Report

The staff finds that Condition 6 from the IMLTR is also applicable to the ESBWR, with a slight modification. Condition 6 from the staff's SE for the IMLTR states:

> The plant specific R-factor calculation at a bundle level will be consistent with lattice axial void conditions expected for the hot channel operating state. The plant-specific EPU/MELLLA+ [extended power uprate/maximum extended load line limit analysis plus] application will confirm that the R-factor calculation is consistent with the hot channel axial void conditions

For the ESBWR, GEH proposed modified language to capture the substance of this IMLTR condition without the inclusion of a plant-specific submittal for EPU or MELLLA+ operation. The staff finds that this modified wording is appropriate and ensures technical consistency with the substance of Condition 6.

R-Factor Condition[7]

The bundle R-factor must be calculated using representative lattice pin power distributions and axial void and power profiles.

---

6 This condition reflects the staff SE for the IMLTR and is incorporated in the LTR through the applicant's response to RAI 7.2-71.

7 This condition reflects the staff SE for the IMLTR and with the response to RAI 4.4-68 and incorporated in the LTR through response to RAI 7.2-71.

3.2.2.1.3 Condition 13—Application of 10 Weight Percent Gadolinia from the Safety Evaluation for the Interim Methods Licensing Topical Report

The staff considered Condition 13 in terms of the T-M methodology qualification, however, the gadolinia bias in the neutronic methods has not been quantified above 8 weight percent (w/o) gadolinia. Therefore, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to a corollary condition regarding the application to high gadolinia loadings. The downstream transient analysis incorporates TGBLA06 gadolinia biases using TRACG04. Section 3.2.3.1.3 of this SE provides additional discussion of these biases.

TGBLA06 8-Weight-Percent Gadolinia Restriction[8]

TGBLA06 is not approved to analyze fuel lattices with gadolinia burnable poison loadings in excess of 8 w/o gadolinia until the NRC staff quantifies and reviews the gadolinia bias.

Should GEH seek loadings in excess of 8 w/o gadolinia that GEH will quantify the biases and submit these biases for NRC staff review.

3.2.2.1.4 Condition 17—Steady-State 5-Percent Bypass Voiding from the Safety Evaluation for the Interim Methods Licensing Topical Report

Section 5 of the SE for the IMLTR provides the basis for the five percent bypass void limitation (Ref. 10). This basis is generic and therefore applicable to the ESBWR. The staff modified the language of the condition to reflect the fact that the ESBWR utilizes the feedwater temperature power operating domain as opposed to a flow control window.

Steady-State 5-Percent Bypass Voiding Limitation[9]

Bypass voiding will be evaluated on a cycle-specific basis to confirm that the void fraction remains below 5 percent at all LPRM levels when operating at steady-state conditions at the upper boundary of the allowable operating domain.

3.2.2.1.5 Mixed Oxide Fuel

NEDC-33239P (Ref. 1) only addresses the application of TGBLA06/PANAC11 to uranium oxide fuel. The LTR does not request approval for mixed oxide fuel. For the purpose of clarification, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following restriction:

Mixed Oxide Fuel Restriction[10]

TGBLA06 is not approved to analyze mixed oxide fuel.

The staff's approval does not extend to the application of TGBLA06 to mixed oxide fuel.

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8 This condition is incorporated in the LTR through response to RAI 7.2-71.

9 This condition reflects the staff SE for the IMLTR and with NEDO-33338 (Ref. 14) and incorporated in the LTR through response to RAI 7.2-71.

10 The staff added this condition for clarification; approval is not sought for mixed oxide fuel under the subject LTRs.

3.2.2.1.6 Bundle Isotopic Tracking

The original version of NEDC-33239P (Ref. 1) described a model for bundle isotopic tracking. The staff was uncertain as to how the ESBWR licensing framework used this model and requested additional information in RAIs 21.6-86 and 21.6-94. In response to these RAIs, GEH revised NEDC-33239P to eliminate discussion of the bundle isotopic tracking model. For clarification purposes, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following restriction on the PANAC11 nuclear design methodology (for additional information regarding the staff's technical review, see Item B.57 in Appendix B to the SE):

> **Bundle Isotopic Tracking Model Restriction**[11]
>
> The staff did not perform a review of the bundle isotopic tracking model. Therefore, staff approval of NEDC-33239P does not constitute approval of the bundle isotopic tracking model.

3.2.2.2 ESBWR-Specific Analysis Features

While leveraging previous experience in the subject review, the staff separately considered the unique aspects of the ESBWR core design. Most importantly, these design features include the instrumentation, core size, and chimneys and natural circulation. The staff separately reviewed the GT instrumentation and CMS in Section 3.3 of this report. In addition, Section 3.2.2.1.6 of this SE describes the staff's consideration of a new model included in NEDC-33239P (Ref. 1) regarding bundle isotopics.

3.2.2.2.1 Core Size

The staff noted that the PANAC11 method is based on a nodal diffusion approach, which relies on the prediction of local diffusion parameter values to solve for the core power distribution and eigenvalue. Because the core size does not impact the ability of the code to predict the values of the nodal parameters, there are no inherent limits on the PANAC11 method in terms of core size. The staff considered the core height only insofar as it indirectly affects the spectral conditions during depletion and axial leakage predictions.

The shortened core height, wider interassembly spacing, and higher inlet subcooling all tend to result in lower core average void fractions relative to the cores in the operating fleet at the same core power-to-flow ratio. Since the ESBWR is designed to operate under conditions of natural circulation, it operates at higher core power-to-flow ratios than currently operating reactors. Therefore, the staff expects the spectral conditions during core depletion to be similar to those experienced at operating fleet EPU cores. The staff has noted this aspect of the ESBWR in Section 3.2.2.1 of this SE.

The staff reviewed the reflector boundary conditions. Mixed-type boundary conditions are employed for the radial and axial reflectors. The reflector diffusion coefficient is determined based on the cross-sections. At the upper axial extreme, the reflector diffusion coefficient may be specified as a function of upper nodal relative water density to capture the effect of increased neutron leakage with decreasing water density at the top of the core (Ref. 1). Because the

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[11] The staff added this limitation for clarification. The staff communicated this limitation to GEH through RAI 21.6-86 and it was incorporated by LTR revision.

model has the capability to treat the phenomenon explicitly, the staff finds that the gross axial leakage will be accurately predicted.

Another analysis potentially sensitive to the reactor core size is the prediction of higher neutron flux harmonics for the purpose of calculating regional mode stability margin. Section 3.2.3.2 of this SE documents the staff review of the use of PANAC11 for this purpose.

3.2.2.2.2 Chimneys and Natural Circulation

Integral to the nuclear design methodology is the prediction of the channel flow rates. The channel flow rates are used to evaluate the nodal void fractions. An iterative scheme (the power-void iteration) is used to calculate the thermal hydraulic conditions and the power distribution. Therefore, the staff reviewed the PANAC11 methodology for determining the bundle flow rates because the ESBWR is unique in that the design includes chimneys. Specifically, the design (1) includes chimney partitions above the core and (2) is intended to operate under the conditions of natural circulation.

The basic premise in the core flow distribution calculation is that each channel flow rate is balanced such that each channel has the same pressure drop. To rapidly converge on the core flow distribution, the model selects characteristic flow channels based on the channel power, axial power shape, crud deposit thickness, orifice size, and channel geometry. The flow distribution for these characteristic channels is determined so that the total core flow is maintained and the pressure drop is balanced across the characteristic channels.

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In RAI 4.4-38, the staff requested additional information regarding the input assumptions in terms of the crud thickness. Since the flow rate is expected to be sensitive to local pressure losses under natural circulation conditions, the staff requested information regarding the model assumptions for crud. The staff reviewed the information provided in the response and agrees that the channel mass flow rate is insensitive to the crud thickness; therefore, the nuclear design analysis results are not significantly impacted by these assumptions (for additional information regarding the staff's technical review, see Item B.10 in Appendix B to the SE).

In RAI 4.4-39, the staff requested additional information regarding the assumption of uniform core pressure drop. The staff requested additional information since the chimney partitions may block thermal hydraulic communication between adjacent super bundles. The staff reviewed the information provided in the response and determined that: [[

]] (for additional information regarding the staff's technical review, see Item B.11 in Appendix B to the SE).

In RAI 21.6-88, the staff requested additional information regarding the characteristic channel method. The staff reviewed the information provided in the response. The response details a subtle difference between the nuclear design method as applied to the ESBWR and the previously approved method described in Reference 6. For the ESBWR, TRACG is used iteratively to calculate the core flow as opposed to using the automated heat balance module in

PANAC11. On the basis of the response, the staff determined that TRACG has sufficient capability to model natural circulation and therefore the staff concludes that the use of TRACG in the nuclear design process and the revisions to the LTR to clarify the method were acceptable. However, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following condition on the limited use of TRACG in the steady-state standard production nuclear design calculations (for additional information regarding the staff's technical review, see Item B.59 in Appendix B to the SE):

Bypass Flow Lookup Table Condition[12]

Licensing evaluations performed with PANAC11 must use bypass flow fractions consistent with all core operating states, as determined by TRACG04, and input in the core simulator to accurately determine the bypass flow. Bypass flow tables or explicit modeling of data from TRACG04 can be used for PANAC11 input values.

3.2.2.3 General Design Criterion 11

To demonstrate compliance with the requirements of GDC 11, the applicant performed calculations to determine the magnitude and nature of the reactivity feedback coefficients. These include the Doppler, moderator temperature, and void reactivity coefficients. These calculations are performed to demonstrate that the reactivity coefficients ensure inherent negative feedback.

The applicant calculated a representative Doppler reactivity coefficient based on lattice calculations for the dominant zone in each of the two fuel bundle types loaded in the reference loading pattern. The applicant performed its analyses by perturbing the fuel temperature based on reference depletion cases within the TGBLA06 lattice physics code. The application of a temperature perturbation, which was applied at several points during the depletion, is the same procedure as performing a temperature branch calculation for each depletion history. The staff has previously found that TGBLA06 is acceptable for this purpose (Ref. 6 and 7).

The Doppler coefficient was estimated based on the eigenvalue difference and the magnitude of the temperature perturbation. The staff finds this approach acceptable as the coefficient calculation is based on the accepted functional dependence of resonance absorption on fuel temperature and is meant to be an indicator of the magnitude and nature of the coefficient. The Doppler coefficient is a strong function of the fertile uranium (e.g. uranium-238) inventory, which is not strongly dependent on the depletion. The applicant's calculations based on lattice depletion indicate that the variation in Doppler coefficient through irradiation is not significant, as illustrated in tabulated eigenvalues for the dominant lattices. These results are expected and confirm that the lattice calculations for several depletion points can be applied to a core with various bundles at different exposures.

Furthermore, the staff finds the lattice perturbation calculation acceptable because the Doppler coefficient was shown to be similar for all the depletion points. The Doppler coefficient is a resonant absorption phenomenon and is not sensitive to core neutron leakage. Therefore, the staff finds that infinite lattice calculations for this purpose are acceptable.

The applicant calculated the moderator temperature coefficient using the PANAC11 core simulator with fixed thermal hydraulic conditions and perturbed lattice parameters. GEH used

[12] This condition reflects the response to RAI 21.6-88 incorporated in the LTR through response to RAI 7.2-71.

the TGBLA06 code to perform moderator temperature branch cases. These nuclear parameters were then exchanged into the core simulator model to simulate, neutronically, the change in moderator temperature for each node in the core model. The applicant calculated the moderator temperature coefficient based on differences in core eigenvalue. The staff finds this approach acceptable because the core model solves for the eigenvalue based on perturbed nodal nuclear parameters, but no modifications are made in the core simulator. The perturbation is carried out in TGBLA06 in much the same manner as a branch case calculation, for which TGBLA06 is suited. Therefore, the combination of these methods would translate the effect on nodal parameters from a perturbation in moderator temperature to a change in the core eigenvalue.

GEH calculated the void reactivity coefficient in the power range of operation by perturbing the inlet enthalpy to the core simulator model and tabulating the difference in core eigenvalue and core average void content. The applicant performed several perturbations to average the void coefficient for the core. The staff finds this approach acceptable because perturbing inlet enthalpy will shift the boiling boundary within the core simulator, thereby perturbing the void distribution throughout the core. By performing several perturbations, the applicant ensures that the point estimates for the void coefficient are consistently negative. While this approach gives an estimate of a whole core void reactivity coefficient, the applicant further identified a limiting condition (cold shutdown) for the void coefficient and performed several calculations at cold shutdown conditions.

For these nominal corewide calculations, the void reactivity is based on lattice parameters developed by TGBLA06 that account for the instantaneous as well as historical void history. Under high void exposure, the buildup of plutonium results in a nonlinear behavior in the reactivity dependence on the void fraction. References 7 and 9 analyze this phenomenon for expanded operating domain boiling-water reactors (BWRs). In response to RAI 4.3-3, the applicant provided a validation of the efficacy of the production technique to predict fuel isotopics relative to explicit depletion calculations carried out at high void fractions (for additional information regarding the staff's technical review, see Item B.4 in Appendix B to the SE). Therefore, while other NRC staff reviews have addressed the application of these methods at high void conditions, the applicant chose to provide a secondary analysis emulating the reactivity effects of introducing voids at cold conditions.

The cold calculations cannot be performed using the inlet perturbation because the coolant is subcooled throughout the entire core at cold conditions. The cold condition will be the limiting scenario (meaning least negative void coefficient) in that the spectrum is over-moderated and the magnitude of the void coefficient decreases with decreasing void.

To perform these calculations for the cold void reactivity coefficient, the applicant applied a method similar to that for the moderator temperature coefficient, whereby the core simulator was used with lattice parameters that were perturbed by doing void branch calculations using TGBLA06. [[

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]] The difference in core

eigenvalue is then used to verify that, in the most limiting condition, the void reactivity coefficient remains negative. The staff finds that this approach, regardless of the high void exposure void reactivity bias, provides an estimate of void reactivity feedback that is sufficiently accurate and conservative to demonstrate a net negative prompt feedback coefficient and thus ensure compliance with GDC 11.

3.2.2.4 General Design Criteria 25, 26, and 27

GDC 25 requires that the protection system be designed to assure that SAFDLs are not exceeded assuming a single malfunction of the reactivity control systems. GDC 26 requires two reactivity control systems based on different design principles. GDC 27 requires that the combined capability of the reactivity control systems, in conjunction with poison addition by the ECCS, be sufficient to assure that the capability to cool the core is maintained under accident conditions. As described below, several PANAC11 eigenvalue calculations demonstrate compliance with GDC 25, 26, and 27.

PANAC11 calculates core eigenvalues based on the nodal parameters and the thermal hydraulic model. The control rods are explicitly treated because the TGBLA06-generated lattice parameters include bladed as well as unbladed lattice parameters. PANAC11 includes a sophisticated model to account for the impact of the bladed or unbladed exposure history on the nodal reactivity. The combined depletion cases, bladed branch cases, and control blade history models allow for accurate prediction of the impact of the control blades on nodal, and hence corewide, reactivity. The efficacy of the PANAC11 control blade history model is sufficient to capture the impact of bladed and unbladed periods of irradiation on the reactivity. However, the shutdown margin (SDM) calculations include both the reactivity effects of xenon and the reactivity effects of boron in the case of standby liquid control system injection.

To evaluate the SDM, PANAC11 must consider the design-basis eigenvalues. The design-basis eigenvalues account for known biases in the core simulator in terms of predicting the core reactivity. The staff requested additional information regarding the design-basis eigenvalues in RAIs 4.4-45 and 4.4-46. The staff reviewed the responses and found that the design-basis eigenvalue accounting methodology is acceptable (for additional information regarding the staff's technical review, see Item B.13 in Appendix B to the SE).

The accuracy in the prediction of the SDM depends on the accuracy of the neutronic methods to predict the distributed criticality. Distributed criticality refers to the overall core criticality under conditions in which the control blades throughout the core are removed and the core is coupled neutronically. Local criticality, by contrast, refers to a configuration in which the core becomes critical with all rods in and one, or possibly two, adjacent control blades removed. A local cold critical test demonstrates the method's ability to predict directly the worth of the strongest rod out configuration.

In its response to the staff's RAIs during review of the IMLTR, the applicant provided recent cold critical demonstration data for two BWR/4 reactors based on in-sequence measurements. Local critical measurements are not available. The results provided for [[

]]. However, because of the use of a 1 percent Δk/k design SDM, this case still meets the 0.38 percent Δk/k TS requirement (Ref. 29). However, these data reflect the need for additional margin in the SDM calculations to compensate for potentially higher model uncertainties.

Some licensees of existing plants perform local cold critical measurements on an infrequent basis. These local cold critical measurements can be used to assess the calculation accuracy of the neutronic methods for the worth of the strongest rod. Reference 30 provides data to demonstrate the accuracy of the prediction of both distributed and local critical measurements for high-power-density plants. The overall standard deviation of predictions of the cold critical eigenvalue was [[ ]].

The GEH methodology applies a [[ ]] to account for observed differences between local and distributed cold critical data. Figure 4-3 of Reference 1 graphically depicts this approach.

The impact of EPU operation was assessed by the review of data provided in Reference 30, which provides data for three successive cycles (23, 24, and 25) for a plant that began EPU operation in cycle 25. However, both cycles 24 and 25 were designed for EPU operation. The data provided indicate that there is essentially no change in the cold critical prediction based on EPU core designs.

Because the ESBWR core design is substantially similar neutronically to EPU core designs, the staff finds that (1) there is reasonable assurance that ESBWR operation will not result in an unexpected trend in eigenvalue bias and (2) relevant operating reactor cycle data form a valid basis for determining the predictable cold eigenvalue margin in the SDM determination before cycle operation.

However, with additional uncertainties accounted for in the design-basis SDM, the staff finds that the design-basis SDM of 1 percent Δk/k affords little uncertainty margin to account for errors in the prediction of the cold eigenvalue bias. During startup, the demonstration eigenvalue is determined by entering the critical rod pattern into the core simulator and predicting the eigenvalue. When the demonstration eigenvalue exceeds unity, this indicates that the code is overpredicting the cold eigenvalue. When the demonstration eigenvalue exceeds the cold critical design-basis eigenvalue, this suggests that the calculated SDM is conservative because it indicates that the calculated eigenvalue in the strongest rod out configuration is over predicted.

In addition to the uncertainties discussed above, GEH has historically applied a design margin of 1 percent Δk/k SDM to ensure that the 0.38 percent Δk/k TS requirement is met. The design margin accounts for the following factors that affect the prediction of the SDM:
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The staff finds that the design-basis SDM of 1 percent Δk/k provides adequate margin to account for the uncertainties that affect prediction of the cold critical behavior of the reactor core at EPU conditions. However, the applicability of the 1 percent Δk/k SDM is based on operational data for which experience provides a high degree of assurance that the predictable

cold eigenvalue bias is quantified and applicable to future cycle analyses. The staff determined that this approach is acceptable. The initial core design, however, requires the selection of design basis eigenvalues without direct ESBWR operating experience. The selection of the initial core design basis eigenvalues is outside the scope of the current review and is addressed in the staff's review of the Initial Core Licensing Topical Report (IC LTR) (Ref. 17). Nonetheless, the review documented herein is applicable to the design certification analysis with respect to the equilibrium cycle and potential reload applications.

PANAC11 has several internal options for calculating control rod worth and, therefore, SDM. The staff requested additional information in RAI 4.4-53 regarding the use of PANAC11 to calculate the SDM. GEH provided these details in its response to RAI 4.4-53 (for additional information regarding the staff's technical review, see Item B.19 in Appendix B to the SE). Individual control rod worth can be calculated using two different procedures. The first method employs a three-dimensional core model. The core reactivity with a control rod fully inserted is compared to the core reactivity with that same rod fully withdrawn, and the control rod worth is determined based on the difference between the two full-core reactivities. This calculation is repeated for each control rod to determine the highest worth control rod for SDM calculations. PANAC11 also includes an option for estimating the control rod worth based on self-adjoint weighting and the integrated two-dimensional radial flux profile. The staff accepts the strongest rod determination based on the full three-dimensional modeling to account for any axial effects that may affect control blade worth. Therefore, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following condition:

PANAC11 Three-Dimensional Shutdown Margin Condition[13]

The ESBWR SDM calculation must be performed using three-dimensional methods. The two-dimensional rod worth estimation technique is not approved for licensing analysis.

When calculating SDM, PANAC11 includes an option for deactivating the Doppler reactivity worth model. When the model is deactivated, PANAC11 SDM includes the change in reactivity as a result of cooling the reactor from hot full-power conditions (including temperature) to cold conditions. This misrepresents the reactivity worth of the control blades. Therefore, the staff accepts the determination of the SDM when the Doppler reactivity effect is accounted for independently of the scram worth. Therefore, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following condition:

Scram Reactivity Calculation Condition[14]

The scram reactivity calculated using the PANAC11 neutronic solver must be calculated with Doppler reactivity feedback modeling activated to accurately determine the reactivity effect of the blades without including the Doppler reactivity in the scram reactivity.

For the boron worth determination, the lattice parameters are also calculated based on boron concentration branch cases. Therefore, the analysis explicitly treats the effect of the boron injection of the core eigenvalue. Accordingly, the staff finds that the boron model is qualified

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13 The staff added this condition for clarification. It reflects the applicant's response to RAI 4.4-53 and Section 4.8 of NEDC-33239P Rev 4 (Ref. 1).

14 The LTR incorporates this condition through the applicant's response to RAI 7.2-71.

over the range of application to the ESBWR. Additionally, the boron libraries were generated to (1) be within the range of the TGBLA06 qualification, (2) bound the cold shutdown concentrations for the ESBWR, and (3) remain sufficiently similar so as not to invalidate the accuracy of the linear interpolation between the two values. Therefore, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following limitation:

> **Boron Branch Limitation**[15]
>
> For the standby liquid control system shutdown analysis, the TGBLA06 borated libraries must be generated with lattice boron inventories between 600 parts per million (ppm) and 1,000 ppm natural boron equivalent.

In the case of the xenon concentration, the model calculations treat the xenon effects separately from all other fission products, allowing the core simulator to calculate the core eigenvalue during steady-state operation while also allowing the code to set the xenon concentration to zero for SDM calculations. The staff has reviewed the xenon worth model incorporated into the core simulator and finds the approach acceptable for determining the effect of xenon on reactivity and power distribution. The staff has previously reviewed and approved this model (Ref. 6). Therefore, the SDM calculation is a specific use of the acceptable xenon worth model where the concentration is fixed at zero.

On the basis of the PANAC11 cold temperature models and associated TGBLA06 input and design-basis eigenvalue selection, the staff finds that the PANAC11 eigenvalue calculations are sufficiently accurate to predict the core reactivity. In addition, GEH has adequately justified the design margin to demonstrate compliance with the ESBWR proposed design bases with respect to GDC 25, 26, and 27.

3.2.2.5 General Design Criterion 28

Compliance with GDC 28 is demonstrated, in part, by analysis of the consequences of a postulated control rod drop accident (CRDA). The analysis methodology is reported in the response to RAI 4.6-23 S02 (Ref. 24). The staff noted some conservatism in the analysis. In particular, the adiabatic assumption precludes any void formation (which would insert negative reactivity during the accident). Also, the calculations assumed that the worth of the dropped rod, regardless of its position during the startup withdrawal sequence, is added to a critical reactor.

The analysis appropriately assumes that the control rod is dropped from its full inserted position to the position of the drive and accounts for the effects of exposure explicitly.

The staff noted that the calculation included neither operator error nor calculational biases and uncertainties. The staff, however, has reviewed the applicability of PANAC11 to evaluating nuclear characteristics for the ESBWR. The staff finds that PANAC11 is suitable for calculations of blade worth for the ESBWR (see Section 3.2.2.4 of this SE). The staff has approved previous versions of PANACEA to provide control blade worth and control rod drop shape information to downstream transient evaluations (Refs. 18, 19, 20, and 21). Therefore, the staff finds that the calculations are indicative of the expected ESBWR behavior; however, the staff does not find that the brief description of the reload licensing methodology is adequate

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[15] This condition is consistent with Section 4.9 of NEDC-33239P (Ref. 1) and incorporated through response to RAI 7.2-71.

to determine its generic application to all ESBWR reload licensing evaluations because this method description does not address modeling biases, uncertainty or operator error.

Therefore, the staff's acceptance of the analytical results for the initial and equilibrium core designs does not constitute staff approval of the generic reload licensing methodology outlined in the response to RAI 4.6-23 S02. With respect to the cycle-specific CRDA evaluations, DCD Section 4.6.2.1.5 states that cycle-specific confirmatory evaluations will be performed based on an NRC-approved or NRC-accepted method for reload cores to ensure that all current and emerging requirements pertaining to a postulated CRDA are met.

The staff finds that the low enthalpy rises are a result of low blade worth (less than 80 cents in all cases). Therefore, the staff finds that the calculational results indicating large margin are expected. The staff finds that consideration of modeling biases, uncertainty, and operator error would not result in changes to the analytic result on the order of magnitude of the available margin because the available margin is approximately 1000%. The large margins to cladding failure for the ESBWR assure that, for the core design described in the DCD, the radiological consequences are bounded by the DCD analyses and that barrier integrity has been demonstrated.

3.2.3 Interface with TRACG04 for Transient Calculations

The nuclear design codes TGBLA06 and PANAC11 perform steady-state evaluations, but also provide nuclear data to TRACG04 for transient evaluations. The TRACG04 three-dimensional kinetics solver is identical to the PANAC11 neutronic solver. Therefore, the staff review of the neutronic methods considered the efficacy of these codes to generate acceptable nuclear data for downstream transient analyses.

Nuclear data generated by PANAC11 is passed to TRACG04 through the PANACEA wrapup file. The staff requested additional information regarding the PANACEA wrapup file in RAI 21.6-85. The response to RAI 21.6-85 describes the contents of the file. The staff reviewed the response and finds that the data are sufficient to fully characterize the kinetic parameters in the core for subsequent transient analyses (for additional information regarding the staff's technical review, see Item B.56 of Appendix B of this SE). Accordingly, RAI 21.6-85 is resolved.

Transient analyses are performed to demonstrate compliance with GDC 10, 20, and 12. In the case of GDC 10 and GDC 20, transient analyses demonstrate that the reactor protection system is capable of shutting down the reactor before the fuel exceeds any SAFDLs. In the case of GDC 12, perturbation analyses are performed with TRACG04 to calculate various decay ratios. The staff reviewed the applicability of PANAC11 to provide upstream data for these analyses, as documented in Sections 3.2.3.1 and 3.2.3.2 of this document.

3.2.3.1 General Design Criteria 10 and 20

Nuclear data generated from TGBLA06 and PANAC11 are used to perform transient analyses by providing input to the TRACG transient reactor analysis code. Therefore, this section of the SE addresses the adequacy of the PANACEA-generated nuclear data for performing transient analyses. The staff has previously reviewed the use of PANAC11 for this purpose in its review of the Migration LTR (Refs. 12 and 13) for AOO and anticipated transient without scram (ATWS) overpressure transient analyses for the operating fleet.

In its review of the Migration LTR, the staff identified two primary technical concerns: (1) void reactivity coefficient biases and uncertainties based on uniform void history and (2) nodal void fraction mismatch affecting transient analyses.

In the conduct of its review of the nuclear design methods for use in ESBWR AOO and infrequent event (IE) analyses, the staff considered the applicability of the staff's findings from the Migration LTR as well as additional concerns regarding the use of historically determined gadolinia bias information for the ESBWR fuel design.

3.2.3.1.1 Void Reactivity Coefficient Biases and Uncertainties

In RAI 21.6-111, the staff requested that GEH revise the void reactivity coefficient biases and uncertainties to more accurately account for nodal void history. The response to the RAI provides the updated model description and the results of TGBLA06 and Monte Carlo N Particle Transport Code (MCNP) comparisons (for additional information regarding the staff's technical review, see Item B.61 in Appendix B to the SE).

The staff has found that the TGBLA06 to MCNP comparisons were adequate to determine the differences in eigenvalue trends with void fraction as a function of void fraction, void history, and exposure to adequately capture these effects on the transient nodal response during AOO or IE simulation with TRACG04 (Ref. 13). On this basis, the staff finds that the model is acceptable and allows extension of the methodology to ESBWR conditions by explicitly modeling the hard spectrum exposure conditions typical of the ESBWR design.

In its review of the void reactivity coefficient correction model, the staff determined that TRACG04 is not acceptable for AOO and ATWS overpressure transient analysis for the ESBWR unless this correction model is activated (Ref. 13).

Furthermore, the void coefficient correction model is based on specific lattice calculations performed using TGBLA06 and MCNP. Lattice designs vary with fuel bundle design; therefore, a set of lattices may not be representative of all future fuel designs. The current lattice set is based on representative modern fuel designs (10 X 10 rod arrays). Applicants or licensees referencing NEDC-33239P (Ref. 1) should either (1) confirm that the void coefficient correction model includes lattice information that is representative of the licensee's fuel or (2) update the void reactivity coefficient correction model lattice database for consistency and evaluate the uncertainties and biases. Therefore, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following void exposure history bias condition:

> **Void Exposure History Bias Condition**[16]
>
> Use of PANAC11-generated nuclear data for ESBWR reload transient analyses (AOO, stability, or ATWS) requires that TRACG utilize the void reactivity coefficient correction model described in NEDE-32906P-A, Supplement 3. The fuel lattices input to the model must represent the cycle-specific fuel loading.

3.2.3.1.2 Void Fraction Mismatch during Initialization

In its review of the Migration LTR, the staff found that differences in the PANAC11 standalone thermal hydraulic model and the TRACG04 thermal hydraulic model led to differences in the

[16] This condition is incorporated in the LTR through response to RAIs 21.6-111 and 7.2-71.

predicted void fraction. The staff requested additional information, in RAI 32 of the Migration LTR review, regarding the effect of the void fraction mismatch. The staff SE for the Migration LTR states the following (Ref. 13):

> The NRC staff's conclusions here are predicated on consideration of those transients that are typically limiting transients in reload licensing analyses. The staff considered those potentially limiting events for the operating fleet of BWR/2-6 reactors. Therefore, the NRC staff's findings in this matter may not be applicable to other BWR designs.

The ESBWR is generally limited by cool down or cold water injection transients as opposed to pressurization transients. Therefore, in RAI 21.6-111 S01, the staff requested additional information regarding the effects of the void fraction mismatch on transient calculations.

RAI 21.6-111 was issued to address the void exposure history effect on void reactivity biases and uncertainties and is substantially similar to RAI 30 relating to the staff review of the Migration LTR. Because the subject of RAI 21.6-111 S01 is TRACG thermal hydraulic modeling, the staff review of the response to the RAI is outside the scope of the current review. The staff was able to complete its review of the neutronic model on the basis of the response to RAI 21.6-111 (for additional information regarding the staff's technical review, see Item B.61 in Appendix B of the SE). The staff's review of the response to RAI 21.6-111 S01 is documented in the SEs for the TRACG04 application to the ESBWR (NEDE-33083P and its supplements (e.g., Ref. 22)).

3.2.3.1.3 Gadolinia Biases

TRACG also accounts for gadolinia biases. As described in its response to RAI 4.4-35, the applicant explained that gadolinia biases are captured based on an operating experience database (for additional information regarding the staff's technical review, see Item B.7 in Appendix B to the SE). The void dependence of the gadolinia bias, however, is sufficiently small that modifying the gadolinia bias treatment for the ESBWR-specific operating conditions would result in a negligible change in predicted dynamic response. Therefore, the staff finds the method for including this bias through TRACG analysis to be acceptable for the ESBWR application. Accordingly, RAI 4.4-35 is resolved.

3.2.3.2 General Design Criterion 12

To demonstrate compliance with GDC 12, GEH performed analyses using the TRACG04 transient methodology. The purpose of these calculations is to determine the channel, core-wide, and regional mode decay ratios. Acceptance criteria are established for the decay ratios based on TRACG04 method and uncertainties. The staff has reviewed and approved the TRACG04 stability methodology (Ref. 22).

In its SE for NEDE-33083P Supplement 1 the staff stated that the design certification review of the ESBWR will address uncertainties in the physics parameters. The staff did not consider the methods employed for generating cross-sections for TRACG as part of the scope of the review of NEDE-33083P, Supplement 1 (Ref. 22), in which the void coefficient is a primary factor in determining core stability.

The staff has reviewed the efficacy of the nuclear design methods to provide cross-section data to TRACG04. The response to RAI 21.6-111 provides a revised methodology for incorporating void reactivity coefficient biases and uncertainties (for additional information regarding the staff's

technical review, see Item B.61 in Appendix B to the SE). As with analysis of transients, the staff finds that PANAC11 is suitable for providing nuclear data for the TRACG04 stability analyses when this revised methodology is used. Accordingly, the staff approves the use of PANAC11 in this respect subject to the condition documented in Section 3.2.3.1.1.

Aside from supplying nuclear data to TRACG04, PANAC11 is also used to predict the first harmonic power shape for use in TRACG nodalization and to determine the radial symmetry plane for the purpose of perturbing the core conditions to excite the regional mode oscillation.

The harmonics module is designed to calculate the flux shape for higher harmonic modes of the flux. The power module calculates the fundamental mode flux distribution. The harmonics module uses the same solution technique, but at each step in the iteration subtracts the fundamental, as well as all lower, harmonic mode solutions from the flux. Therefore, the solution iterates to convergence on increasingly higher harmonic modes. The power technique also calculates the core eigenvalue. The difference in these eigenvalues gives the eigenvalue separation between the modes.

It should be noted that the harmonics module can only be used for a full-core model, otherwise there is no way to capture asymmetries in the neutron flux.

Flux Harmonic Condition[17]

The regional mode stability analysis must be performed using a radial nodalization in TRACG04 based on the PANAC11-generated first harmonic mode. The harmonic calculation performed by PANAC11 must use a full-core representation.

The staff finds that the solution technique for the higher harmonic flux shapes is well supported by the theoretical application and the qualification of the model. The staff has previously reviewed and approved the application of the harmonic flux shape in the determination process for the TRACG nodalization and perturbation for the ESBWR stability methodology described in Reference 22. Therefore, the staff finds that PANAC11 is suitable for providing input to inform the TRACG nodalization for ESBWR stability evaluations.

3.2.4 Code Documentation, Maintenance, and Quality Assurance

The nuclear design codes support the transient and accident analyses provided in DCD Chapter 15. Additionally, PANAC11 forms the kinetic methodology in the TRACG transient code. Therefore, the staff conducted its review of the calculational methods in accordance with SRP Section 15.0.2. This SRP section directs the staff to review the complete code documentation. This documentation discusses (1) the evaluation model, (2) the accident scenario identification process, (3) the code assessment, (4) the uncertainty analysis, (5) a theory manual, (6) a user manual, and (7) the quality assurance program.

The staff conducted an audit of the nuclear design codes to review these code documents. Because the nuclear design codes play a supporting role in the transient and accident analysis methods, the staff did not review the accident scenario identification process as part of the subject review.

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[17] This condition reflects the staff's safety evaluation report for NEDE-33083P, Supplement 1 (Ref. 22), and incorporated in the LTR through response to RAI 7.2-71.

The staff reviewed the accident scenario identification process in its review of NEDE-33083 (Ref. 22) and its supplements.

Reference 23 documents the staff audit. The staff reviewed the complete code documentation. The staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to following condition on the approval of the nuclear design codes:

Code Usage Condition[18]

The limitations on TGBLA06 and PANAC11 code usage, as described in the user manuals, are a condition of the acceptance of these methodologies for the ESBWR. Changes to the manuals that are made in accordance with the quality assurance procedures audited by the staff, as documented in the applicable reference, do not require NRC review and approval. However, if used in the safety analysis, the cycle-specific supplemental reload licensing report (SRLR) must document these changes.

In the conduct of its audit, the staff reviewed the quality assurance program guiding code updates and code error corrections. The staff finds that the program addressed the SRP Section 15.0.2 criteria of design control, document control, software configuration control and testing, and error identification and corrective actions (Reference 23). The staff identified one open item regarding code drift. “Code drift” refers to changes between revisions that fall within acceptance criteria during each revision, but occur subsequently such that a consistent trend in these changes may compound. This is a particular concern because the internal code revision procedures and guidance do not call for comparisons of code revisions to either the originally approved code or to any data in the original qualification.

The staff identified an open item as part of the audit and requested in RAI 4.3-4 that GEH assess whether TGBLA06/PANAC11 code revisions have resulted in code drift. The staff reviewed the response to RAI 4.3-4 and determined that code drift has not occurred during the maintenance of these codes since NRC approval (for additional information regarding the staff’s technical review, see Item B.5 in Appendix B to the SE). Reference 23 documents the closure of the open item.

The staff noted that code maintenance activities in certain instances result in code modifications or updates. Specifically, 10 CFR 52.98, “Finality of Combined Licenses; Information Requests,” outlines the regulatory change processes that may apply to address the potential for future code updates. Guidelines for prior NRC review and approval of future code updates are consistent with the definition of a methodology change in 10 CFR 50.59(a)(1) and the criteria of 10 CFR 50.59(c)(2)(viii) to ensure that the methodology is not adversely impacted for reload licensing or core monitoring purposes. In this vein, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following code change limitation:

Code Change Limitation[19]

- The NRC staff considers modifications to the models described in NEDC-33239P-A or MFN 098-96 to constitute a departure from a method of evaluation in the safety

[18] This condition is incorporated in the LTR through response to RAI 7.2-71.
[19] This condition is incorporated in the LTR through response to RAI 7.2-71.

analysis, and they may not be used for licensing calculations without prior NRC review and approval.

- The NRC considers modifications to the TGBLA06/PANAC11 codes that result in inconsistency with the NEDC-33239P-A LTR to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval of the necessary revisions to the LTR.

- The NRC staff considers modifications to the TGBLA06/PANAC11 codes or the GT CMS software that result in inconsistency with the NEDE-33197P-A LTR to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval of the necessary revisions to the LTR.

- The NRC staff does not consider updates to the PANAC11 nuclear methods to ensure compatibility with other NRC-approved methods to constitute a departure from a method of evaluation in the safety analysis. These updates may be used for licensing calculations without prior NRC review and approval so long as the predicted ESBWR equilibrium cycle MLHGR or the downstream ΔCPR/ICPR for the potentially limiting transients (calculated by TRACG04) show less than a 1-standard-deviation difference.

- The NRC staff does not consider increases in the spatial or energy resolution in the TGBLA06 lattice physics method to constitute a departure from a method of evaluation in the safety analysis. These updates may be used for licensing calculations without prior NRC review and approval so long as the uncertainties in the lattice parameters do not increase as a result. In all cases, the cycle-specific SRLR, if used in the safety analysis, must document modifications or updates done without prior NRC review and approval.

- The NRC staff does not consider changes in the numerical methods to improve code convergence to constitute a departure from a method of evaluation in the safety analysis, and they may be used in the licensing calculations without prior NRC review and approval.

3.3 Gamma Thermometer Core Monitoring System

To meet the requirements of GDC 13, the ESBWR incorporates a GT-based CMS. The GT replace TIPs used in the operating fleet of reactors to perform the functions of LPRM calibration and axial power shape (APS) adaption. The GT LTR (Ref. 2) describes the GT devices, the principle of their operation, and their interface with the 3D MONICORE core simulator. The staff reviewed the GT system and the GT CMS to ensure that these instruments were sufficiently capable to monitor the fission process. The staff also reviewed the GT system to determine how data collected from these devices are used in the CMS to monitor thermal margin.

3.3.1 Design Description

As shown in Figure 1-3 of the GT LTR (Ref. 2), the LPRM/GT assembly consists of a GT rod with seven GT sensors and the normal compliment of four LPRMs. One GT sensor is

positioned adjacent to each LPRM and one midway between each pair of LPRMs. In addition to GT sensors, a cable pack is placed in the center of the GT rod. This cable pack (see Figure 1 in Ref. 31) is a tightly compacted system consisting of the central heater cable and nine thermocouple locations, seven of which are utilized in the ESBWR design. The central heater cable provides a means of calibrating the thermometers.

3.3.2 Principle of Operation

Figure 1-2 of the GT LTR (Ref. 2) depicts the principle of GT operation. GT sensors are intended to determine the local power distribution by measuring the relative intensity of the gamma flux at the detector location. The instrument is formed by a metallic sensing region that is insulated from the core bypass by an argon fill gas chamber.

During reactor operation, prompt and delayed gamma energy released from the fuel is deposited in the metallic heating section. The heat is then conducted primarily through the metallic section to the noninsulated portion of the device and removed by the bypass cooling water. The insulated and noninsulated sections include hot and cold thermocouple junctions, respectively. The thermocouple is used to measure the temperature difference between the insulated and noninsulated sections. The measured temperature difference is related to the amount of heat deposited in the sensing section, which in turn is related to the local gamma flux. The local fission power is also related to the local gamma flux.

3.3.3 Equipment Design and Experimental Qualification

The staff conducted a review of the GT design described in the GT LTR (Ref. 2) to ensure that the device would operate as intended. The staff requested detailed design information in RAI 7.2-5 regarding the makeup of the GT components. The response provides the materials used to construct the device (for additional information regarding the staff's technical review, see Item B.22 in Appendix B to the SE). The staff was primarily concerned about the conductivity and electrical insulation properties of these materials. In the case of the fill gas, the staff was concerned about the fill gas pressure and the thermal insulation properties of the insulating fill gas region. The staff reviewed the response and finds that the design choices are appropriate to allow the device to perform as intended.

In addition, the staff requested additional information regarding the performance of the thermoelement. In particular, the staff requested additional information in RAIs 7.2-6 and 7.2-8 regarding the thermocouple signal response. The response to RAI 7.2-6 provides additional clarification regarding the influence of the calibration heat wire current on the thermocouple signal and demonstrates sufficient insulation to prevent erroneous thermocouple indication during calibration (for additional information regarding the staff's technical review, see Item B.23 in Appendix B to the SE). The staff requested information in RAI 7.2-8 regarding any influence of dissimilar metal interfaces on thermoelement signals.

The response provides justification that the design of the device does not result in erroneous signal output caused by any voltages induced by dissimilar metal interfaces (for additional information regarding the staff's technical review, see Item B.25 in Appendix B to the SE).

As described above, the staff reviewed the information provided in the GT LTR and the responses to RAIs 7.2-5, 7.2-6, and 7.2-8. On the basis of this information, the staff finds that the equipment design is appropriate to perform its intended function. Accordingly, RAIs 7.2-5, 7.2-6, and 7.2-8 are resolved.

The efficacy of the GT is further demonstrated by in-plant tests performed at Limerick 2, Tokai 2, and Kashiwazaki-Kariwa Unit 5 (K5). These test data include comparisons between GT and TIP measurements and provide reasonable assurance that the GT provides an acceptably accurate measurement of the local power. In the conduct of its review, the staff requested additional information to justify the applicability of the in-plant test data to the ESBWR. The response to RAIs 4.2-12 S02-12 and 4.3-2 S02-C-1 justify the in-plant test data applicability and clarify the use of the in-plant test data as part of the overall GT CMS qualification and uncertainty analysis. The staff reviewed these responses and finds that the data were used in a limited scope commensurate with the degree of the detail provided by these tests (for additional information regarding the staff's technical review, see Items B.1.2.5 and B.3.6.3 in Appendix B to the SE).

On the basis of the in-plant qualification and detailed design information evaluated above, the staff finds that the GT design is acceptable.

3.3.4 Signal to Power

The GT CMS utilizes the voltage readings from the GT thermocouples to infer the local reactor power. To perform this function, the signal must be analyzed using various models to relate the voltage to the gamma heating and, in turn, the gamma heating to the local fission power. This process requires GT CMS models of the GT instrument and gamma transport models.

The staff reviewed these models and their bases to evaluate the efficacy of the GT CMS to infer the local reactor power distribution. Section 3.3.9 of this SE discusses the staff's review of the capability of the GT CMS to determine the entire reactor power distribution based on the local power information inferred from the GT .

3.3.4.1 [[ ]] Method

The GT CMS relies on the [[ ]] method to correlate the GT voltage to the gamma heating. As described in Section 3.3.2, the GT CMS operates by relating the temperature difference across the GT hot and cold junctions to the gamma heat deposition in the insulated section. However, this relationship is not linear. Effects such as variation in the GT thermal conductivity at various temperatures mean that the sensitivity of the instrument changes at various temperatures. To address this effect, the GT CMS relies on capturing the nonlinear effects using a correction to the relationship between the thermocouple voltage and the local gamma heat deposition using a [[ ]] method.

During factory calibration, the GT signal is related to the [[ ]]. Equation 4.1-4 of Reference 2 describes the [[ ]] method. Section 4.2 of Reference 2 describes the factory tests performed for each GT to determine the GT-specific [[ ]]. In RAI 7.2-19, the staff requested additional information regarding the validity of the factory calibration method because in-reactor conditions differ from the factory calibration conditions.

In the response to RAI 7.2-19, the applicant analyzed any signal biases introduced as a function of the changes in material properties between the factory test conditions and the reactor conditions. The response adequately justifies the approach based on the negligible error that is introduced to the relative power shape measured by the instruments (for additional information regarding the staff's technical review, see Item B.36 in Appendix B to the SE). Therefore, the

staff finds that the model is sufficiently robust to address the nonlinearity of the GT signal and that the factory calibration is adequate to compensate for the nonlinearity. Accordingly, RAI 7.2-19 is resolved.

In response to RAI 7.2-10, the applicant describes process improvements in the GT CMS based on early in-plant tests performed at Limerick (for additional information regarding the staff's technical review, see Item B.27 in Appendix B to the SE). One of the improvements is to utilize the specific GT [[ ]] for each specific GT in the reactor. This leads to an improvement in the local prediction of reactor power relative to using an average [[ ]] for each GT instrument. In response to RAI 7.2-18, the applicant stated that the [[ ]] method using [[ ]] value for each GT sensor has to be applied for the GT in-plant calibration (for additional information regarding the staff's technical review, see Item B.35 in Appendix B to the SE). Therefore, the GT CMS should track these individual GT-specific [[ ]] values. Accordingly, RAIs 7.2-10 and 7.2-18 are resolved. The staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following condition:

GT-Specific [[ ]] Values Condition[20]

The CMS must track individual [[ ]] values for each GT in the core.

3.3.4.2 J-Factor Method

The [[ ]] method is used to correlate the GT thermocouple voltage to the local gamma heating. GEH utilizes a detector response model based on [[ ]].

The J-factor methodology has previously been applied to [[ ]]. The staff requested additional information regarding the detector response model in RAIs 7.2-9 and 21.6-89. The response to RAI 4.2-12 effectively supersedes these responses. In RAI 4.2-12, the applicant provides the details of the revised J-factors and how these factors are used in GT CMS (for additional information regarding the staff's technical review, see Item B.1 in Appendix B to the SE).

The response to RAI 4.2-12 describes the MCNP calculations performed to determine the [[ ]] The results indicate an acceptable degree of agreement. Based on this comparison, the staff finds that the revised correlation parameters are appropriate to capture design differences between gamma TIPs and the GT and to address the fuel-specific transport properties (for additional information regarding the staff's technical review, see Item B.1 in Appendix B to the SE). Therefore, the staff finds that the use of the J-factor methodology is appropriate and acceptable. Accordingly, RAI 4.2-12 is resolved.

[20] This condition is incorporated in the GT LTR appendix through response to RAI 7.2-18S2.

However, as stated above, these calculations are performed using detailed two-dimensional nuclear simulations. Section 3.3.4.3 of this SE discusses the staff's review of the effects of local axial geometry variation.

3.3.4.3 Local Geometry Effects

Unlike TIP instruments, the GT are arrayed axially through the core at discrete axial locations. Therefore, axial geometry variations have an impact on the GT signal that cannot be neglected. In the case of TIPs, measurements of the power are made at every inch, and these detailed axial data are averaged to determine the nodal powers. Therefore, the effect of a signal anomaly caused by a spacer or other geometry variation is negligible because of the averaging process.

In RAIs 4.2-12, 7.2-20, and 7.2-72, the staff requested that GEH evaluate the influence of local axial geometry on the GT signal and develop model refinements to the detector response model to account for any biases introduced as a result of axial geometry effects. The response to RAI 4.2-12 refers to RAI 7.2-20 in terms of the model for the spacer effect. The response to RAI 7.2-72, however, is more comprehensive and addresses a variety of axial geometry effects (for additional information regarding the staff's technical review, see Items B.1, B.37, and B.54 in Appendix B to the SE).

The response to RAI 7.2-72 incorporates the response to RAI 7.2-20 in terms of quantifying the spacer effect and addresses nodal hybridization. For a specific fuel product, the applicant performed detailed nuclear simulations using Monte Carlo calculations to determine GT signal biases resulting from (1) the presence of a fuel spacer near the GT and (2) axial fuel geometric variation within a node (hybrid nodes) (for additional information regarding the staff's technical review, see Item B.54 in Appendix B to the SE).

The fuel spacers are metallic and thus contribute to gamma shielding in the vicinity of the GT instrument. If this shielding effect is not accounted for, then the GT measurement of the local nodal powers will be biased. The response to RAI 7.2-72 describes the analyses that the applicant performed to evaluate the relationship between the signal bias and the relative location of a fuel spacer to the GT instrument. The staff has reviewed this model and finds that the detailed nuclear simulations are sufficient on a fuel-product basis to quantify and model the spacer bias because they explicitly treat the fuel-product specific geometry with sophisticated and highly accurate transport methods.

The PANAC11 engine in 3D MONICORE for the operating fleet utilizes nodal-averaged J-factors in evaluating instrument response. However, the GT are arrayed at specific axial elevations within the core. It is possible for specific fuel product designs to include geometry variations within a subnode. For example, it is possible for PANAC11 to model a single node that includes the transition from the dominant zone to the plenum zone. This is referred to as a hybrid node. In RAI 7.2-72, the staff requested that GEH develop a refinement to PANAC11 in the GT CMS to account for any signal biases that may be introduced when the GT is adjacent to a particular geometry; however, the balance of the node may have a significant variation in gamma transport properties caused by a geometry change above the instrument.

In response to RAI 7.2-72, GEH described a methodology for averaging lattice transport properties with an appropriate weighting function to account for the sensitivity of the GT instrument to the local axial geometry. The staff reviewed the response and found that, in particular instances, the local geometry refinement models may improve the GT response

model by approximately 5 percent. The methodology explicitly accounts for impact of the geometry on the gamma transport characteristics (for additional information regarding the staff's technical review, see Item B.54 in Appendix B to the SE). Accordingly, RAI 7.2-72 is resolved.

The staff noted that these refinement models are based on detailed Monte Carlo calculations performed for specific fuel products. The staff finds the basis for the models to be acceptable. The staff agrees that the refinement model parameters must be determined on a fuel-product-specific basis and that these parameters must be utilized in the GT CMS on a cycle-specific basis to reflect the fuel loading. Therefore, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to a condition that the refinement model be utilized consistent with the RAI response and the LTR revision.

> **Local Geometry Refinement Model Condition**[21]
>
> The parameters used to compensate for biases introduced in the GT sensor signal by the proximity to spacers or fuel type changes or both will be determined only when a new bundle design (i.e., new axial lattice composition) or a new spacer design (i.e., material) is applied to a particular ESBWR core loading, as described in Section 8.6 of NEDE-33197P-A. The parameters will be incorporated into the GT-based monitoring system in a cycle-specific basis, as required.

The language of the condition clarifies that if subsequent cycle core designs include the same fuel product that the fuel-product-specific factors do not have to be recalculated using the methodology.

3.3.5 Gamma Thermometer Calibration

During operation, the GT instruments must be periodically calibrated to account for sensitivity changes. Section 3.3.6 of this SE describes several mechanisms that affect the sensitivity of the GT. Section 4.2 of Reference 2 describes the process for performing the in-plant calibration. To calibrate the GT, a known current is passed through the in-line heater wire. This deposits ohmic heating to the insulating section of the GT. Equation 4.3-3 shows how the GT sensitivity can be calculated based on the nominal GT reading and the reading during in-line heating.

The staff has reviewed the calibration procedure and finds the method described in Section 4.3 to be acceptable as these methods utilize sufficient data to determine the GT sensitivity. However, the staff noted that additional consideration must be given to conditions during the calibration. In particular, the GT response time to the ohmic heat deposition is lagged because of the thermal time constant of the instrument. The staff requested additional information in RAI 7.2-20 regarding the conditions for GT calibration (for additional information regarding the staff's technical review, see Item B.37.12 in Appendix B to the SE). The response states that the thermal time constant is established during factory testing. GEH provided the results of several factory tests that show that the thermal time constant is expected to be on the order of [[ ]]. The response states that during calibration, the current in the in-line heater wire must be held constant for a duration of at least five thermal time constants. After five thermal time constants the signal has essentially achieved its steady-state value. The staff finds that this will ensure that the GT signal is constant during the calibration. Therefore, the staff

[21] This condition is incorporated in the GT LTR through revision based on the response to RAI 7.2-72.

approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following condition on the GT calibration procedure:

GT In-Line Heater Wire Current Hold Time Condition[22]

GT calibration must be performed using a current hold time of at least five GT thermal time constants per current magnitude per string.

3.3.6 Irradiation Effects

The staff reviewed the effect of irradiation on the GT performance. In particular the staff reviewed the sensitivity changes with irradiation and any changes in the physical properties affecting the instrument performance.

3.3.6.1 Heater Wire Resistance

The staff requested additional information in RAI 7.2-7 regarding the effect of irradiation on the GT heater wire resistance. In particular, the staff requested that GEH evaluate the potential for changes in the heater wire material properties on the ability to calibrate the GT instrument. The staff was concerned that irradiation damage in the heater wire, and subsequent changes to the conductivity, would result in a systematic calibration error in high flux regions of the reactor. The response to RAI 7.2-7 provides analyses predicting the change in heater wire resistance under conditions of irradiation. The analyses provide assurance that the changes in the heater wire resistance are negligible compared to other sources of uncertainty in the GT CMS (for additional information regarding the staff's technical review, see Item B.24 in Appendix B to the SE).

The response to RAI 7.2-7, however, outlines a methodology for correcting the analytical models to account for changes in heater wire resistance based on periodic measurement of the resistance. Such a model is not necessary at this stage, but, if GEH were to seek approval for such a model to improve GT CMS performance, the model would require NRC review and approval because the downstream impact of such a model on the overall accuracy of the CMS has not been determined. Therefore, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following condition:

GT Heater Wire Exposure-Dependent Resistance Condition[23]

If changes are deemed appropriate to improve accuracy by accounting for GT heater wire resistance changes during irradiation, implementation of the proposed method outlined in the applicant's response to RAI 7.2-7 requires revision to NEDE-33197P. The NRC staff considers this to constitute a departure from the method of evaluation that has not been reviewed by the NRC for the intended application.

3.3.6.2 Sensitivity Decrease Model

The original NEDE-33197P revision (Ref. 3) included a description of a sensitivity decrease model. The purpose of the model was to predict the GT sensitivity between GT calibrations.

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22 This condition is consistent with Section 4.3 of the GT LTR.

23 This condition reflects the response to RAI 7.2-7.

During irradiation, the GT sensitivity during the in-plant tests was shown to change, particularly during early irradiation. The LTR attributed the sensitivity change to [[
]] (Ref. 2).

The LTR states that [[
]]. In RAI 7.2-12, the staff requested that GEH evaluate the potential of [[
]]. In response to RAI 7.2-12, GEH evaluated [[

]] (for additional information regarding the staff's technical review, see Item B.29 in Appendix B to the SE).

The staff observed trends consistent with decreasing sensitivity during early irradiation, but identified other incidences of sensitivity increase during early irradiation. The staff requested additional information regarding the sensitivity decrease model in RAIs 7.2-11 and 7.2-63.

In RAI 7.2-11, the staff requested that the applicant address the ramifications of sensitivity decrease modeling for the purpose of extended durations between calibrations given that the model may misrepresent the actual change in GT sensitivity. The response stated that irradiation damage to the materials may cause changes in the electrical resistance, which in turn may lead to an increase or decrease in sensitivity during the initial stages of operation (for additional information regarding the staff's technical review, see Item B.28 in Appendix B to the SE). This explanation is inconsistent with the evaluation of resistance changes under irradiation provided in response to RAI 7.2-7 (for additional information regarding the staff's technical review, see Item B.24 in Appendix B to the SE).

Therefore, the staff does not agree with the basis for the mechanistic model to predict the GT sensitivity and does not approve the use of the sensitivity decrease model. The GT may not be used for in-core instrumentation for the ESBWR unless GT sensitivity is established through calibrations using the in-line heaters before adaption or LPRM calibration.

Additional information provided in the response to RAI 7.2-63 states that the sensitivity decrease model is not required because the GT are calibrated before use (for additional information regarding the staff's technical review, see Item B.49 in Appendix B to the SE). In response to the staff's RAIs, GEH removed the sensitivity decrease model from the LTR. For clarification, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following condition regarding the sensitivity decrease model:

> **Sensitivity Decrease Model Restriction**[24]
>
> The sensitivity decrease model is not approved. Therefore, when GT are used for the purpose of LPRM calibration or adaption, they must be calibrated beforehand using in-line heater calibration to determine the sensitivity.

The staff noted that the sensitivity of the GT cannot be predicted using analytical methods and must be measured before use. Therefore, should a GT in-line heater fail, there is no means for establishing the GT sensitivity. If the sensitivity cannot be established, the GT sensor indication is suspect. A licensee would then be compelled to declare the affected GT inoperable.

[24] This condition is incorporated through GT LTR revision, and reflects the RAI 7.2-63 response.

Accordingly, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following condition:

> **GT Operability Condition**[25]
>
> The failure of a GT heater is considered a loss of calibration capability of the full GT string (all sensors). Therefore, in case of failure of a GT heater, the GT CMS will declare the GT string as inoperable.

3.3.6.3 Gamma Thermometer Lifetime

In RAI 7.2-61, the staff requested that the applicant determine the GT lifetime and the anticipated replacement schedule. The applicant stated that the GT will be replaced on the same schedule as the LPRMs. The applicant stated that the GT will be designed to last at least as long as the LPRMs. The analyses provided by the applicant to assess the irradiation damage and [[ ]] indicated acceptable performance for up to eight effective full-power years. Eight effective full-power years is consistent with the LPRM lifetime (for additional information regarding the staff's technical review, see Item B.47 in Appendix B to the SE). Therefore, the staff finds the GT replacement schedule acceptable.

3.3.7 Environmental Effects

The staff considered several environmental effects within the bypass and considered the potential for these effects to impact the performance of the GT. The staff considered thermal hydraulic conditions within the bypass, particularly in light of the feedwater temperature power operating domain described in Reference 14. The staff noted that the GT sensitivity is likely to be a function of the heat transfer characteristics to the bypass water through the non-insulated sections of the GT tube. Similarly, the staff considered the N-lattice and the potential for gamma streaming through the interassembly bypass area.

3.3.7.1 Bypass Thermal Hydraulics

In RAIs 7.2-13 and 4.3-2, the staff requested that GEH evaluate the effects of bypass void formation on GT performance. In response to RAI 7.2-13, the applicant estimated the effect of the thermal hydraulic conditions on the instrument sensitivity. The response evaluates the expected change in sensitivity with changes in temperature consistent with changing bypass conditions (for additional information regarding the staff's technical review, see Item B.30 in RAIs 7.2-13 and 4.3-2). The staff noted that changes in sensitivity with changes in temperature provide an additional basis for the conclusion that the GT must be calibrated before use given that the bypass conditions vary widely in the expanded operating domain described in Reference 14.

In response to RAI 4.3-2 S02-C-2, GEH provided the results of a full-scale experiment conducted at the Multi-Use Safety Experimental facility. The test results confirm that the GT sensitivity is not a strong function of the bypass void fraction up to 55 percent (for additional information regarding the staff's technical review, see Item B.3.6.4 in Appendix B to the SE). NEDC-33239P is not approved for use in an ESBWR unless bypass void conditions are less than 5 percent at the highest LPRM and GT location (see Section 3.2.2.1.4 of this SE).

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[25] This condition is incorporated in GT LTR through response to RAI 7.2-71.

Therefore, the staff finds that void formation within the bypass does not adversely affect the GT performance for the ESBWR.

The staff also considered conditions of operation where the bypass is highly subcooled, such as operation at SP1M[26] in the expanded operating domain. The requirements to calibrate the GT before use address the concerns regarding increased GT sensitivity under conditions of high bypass subcooling, thereby inherently accounting for the change in sensitivity caused by the thermal hydraulic conditions in the bypass at high inlet subcooling conditions. Section 4.4.2 of the GT LTR (Ref. 2) addresses the effects of high inlet subcooling.

The staff finds that the GT performance is acceptable over the entire range of normal operating conditions because (1) the bypass thermal hydraulic conditions and their effect on the GT sensitivity are inherently captured during the in-line heater calibration and (2) the GT sensitivity is not a function of the bypass void fraction over the anticipated range for the ESBWR. Transient conditions are addressed in Section 3.3.8 of this SE.

3.3.7.2 Gamma Streaming

The staff noted that the N-lattice design of the assembly spacing for the ESBWR results in a slightly wider water gap between the assemblies as compared to the plants in the in-plant qualification dataset. In RAI 7.2-57, the staff requested that GEH evaluate the potential for interassembly gamma streaming to introduce biases in the local gamma flux indication in the GT.

In the response to RAI 7.2-57, the applicant stated that, while there will be gamma streaming in the bypass, this contribution is expected to be very small (for additional information regarding the staff's technical review, see Item B.44 in Appendix B of this SE). The staff agrees because the length of the fuel bundle would effectively collimate the cross-bundle gamma sources and thus result in a very low gamma flux contribution. Additionally, the applicant stated that this cross-bundle effect is likely to exist for all GT in the core to a certain extent, and the normalization of the signals to determine the axial power shape would effectively normalize out any cross-bundle gamma transport effects (for additional information regarding the staff's technical review, see Item B.44 in Appendix B to the SE). The staff agrees with the applicant's assessment and finds that any additional uncertainty as a result of cross-bundle gamma transport through the bypass would have a negligible effect on the overall uncertainty assessment and would not preclude the GT from producing an indication representative of the local four bundle power. Based on the foregoing staff evaluation, RAI 7.2-57 is resolved.

3.3.8 Transient Effects

The GT LTR includes a model for delayed gamma compensation. The staff considered the effect of operational and anticipated transients on the GT instruments and the GT indications.

3.3.8.1 Delayed Gamma Compensation Model

The GT instruments are slow to respond to changing reactor power caused by delayed gamma radiation. Therefore, while the LPRM instruments will respond promptly to changes in the

[26] SP1M refers to the operating point in the feedwater temperature power operating domain along the licensed thermal power line at the minimum feedwater temperature allowed based on stability considerations.

neutron flux, the GT will lag the LPRM response because of delayed gammas. Section 4.5 of the GT LTR describes the delayed gamma compensation model, and Section 7 includes several in-plant transient tests comparing the uncompensated and compensated GT signals to measured LPRM signals (Ref. 2).

The staff noted that the delayed gamma compensation model also compensates for the GT thermal inertia. Table 4-1 provides a mode 0 time constant of approximately [[ ]] to account for the GT thermal time constant. In RAI 7.2-59, the staff requested that, should approval be sought for the delayed gamma compensation model, its uncertainties should be factored into the overall uncertainty analysis (this includes a consideration of the range of GT thermal time constants). The response to RAI 7.2-59 S02 states that the GT are only used under conditions of steady-state operation (for additional information regarding the staff's technical review, see Item B.46 in Appendix B to the SE). Therefore, the delayed gamma compensation model is not needed to perform those functions described in the GT LTR. The staff did not review the delayed gamma compensation model, and the staff approval of the GT LTR does not constitute approval of the delayed gamma compensation model.

Should future approval be sought for the delayed gamma compensation model, GEH should address inherent limiting assumptions regarding the model. In particular, the staff noted that the ratio of gammas released per fission may be a function of the control state of the nearby bundles. Therefore, compensation may result in GT instrument biases relative to the LPRMs when the control state of an instrumented node changes during a plant maneuver. The staff reviewed transient test data during an onsite audit that indicated that control blade movement introduces GT biases (Ref. 25). The current methodology does not appear to capture this effect. Additionally, maneuvering in the feedwater temperature power operating domain may result in changes in GT sensitivity as the bypass temperature changes. The delayed gamma compensation model does not appear to account for transient bypass conditions.

The description of the model does not provide sufficient information to understand the operation of the GT during in-line heater calibration if the delayed gamma compensation model is activated. The compensation may result in unintended errors in GT-compensated response during calibration. The GT LTR does not present this information in sufficient detail for the staff to review the use of the delayed gamma compensation model. Finally, the staff noted that the GT CMS uncertainties do not quantify or include the delayed gamma compensation model uncertainties.

While some of the transient in-plant test data indicate that compensating the GT signals yield greater agreement between the LPRM and GT response, the staff does not have sufficient information regarding the performance of the compensation model for feedwater maneuvers, control blade movements, or relatively fast transients.

On these bases, the staff has restricted the use of the delayed gamma compensation model and does not approve application of the topical unless the reactor is in an appropriate steady-state condition before GT use. The staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following condition:

Delayed Gamma Compensation Model Restriction[27]

The delayed gamma compensation model is not approved. Therefore, GT calibration and the use of GT for LPRM calibration or adaption purposes may only be performed during a steady-state condition of operation that meets the core requirements described in Section 4.5 of the GT LTR.

3.3.8.2 Functionality Following a Transient

The staff requested additional information regarding the GT instruments in terms of performance during AOOs. The staff noted that GT need not provide data during AOOs nor are they used to actuate automated plant responses. In RAI 7.2-65, the staff requested that the applicant address the potential to damage a GT during a reactor transient. The applicant evaluated the expected heat deposition in the GT core region during anticipated transients and determined the specific energy deposition to be approximately half of the saturation specific energy deposition (for additional information regarding the staff's technical review, see Item B.51 in Appendix B to the SE). Therefore, the staff finds that the GT instruments will function properly following an anticipated transient condition. Based on the foregoing evaluation, RAI 7.2-65 is resolved.

For the reasons set forth above, the staff finds that the GT instruments will allow a reactor to continue to meet the requirements of GDC 13 following a transient during cycle operation.

3.3.9 Power Shape Adaption

Power shape adaption refers to the process in which the GT CMS adjusts nodal nuclear parameters to adapt the core power distribution to the measured nodal power distribution. In the operating fleet of plants, the adaption is based on nodal TIP measurements. In the ESBWR, the GT instruments are axially arrayed at discrete axial locations; therefore, the adaption technique must include a means for interpolating between the discrete GT signals to determine a continuous power shape. The continuous power shape is used to adjust the reactor power distribution. This power distribution is used to calibrate the LPRMs.

The response to RAI 4.2-12 states that the adaptive technique is based on axial power shape adaption. Shape adaption refers to an adaptive method by which the nodal parameters are adjusted such that the reactor axial power distribution matches the measured continuous axial power shape and the radial power distribution is unadjusted (for additional information regarding the staff's technical review, see Item B.1 in Appendix B to the SE).

The staff noted that the use of an interpolated power shape necessarily results in increases in the uncertainty in the monitoring axial power shape because the power shape is determined from fewer axial measurements for the ESBWR relative to the operating fleet. Therefore, the staff focused its review of the interpolation method to ensure that the increase in the nodal power distribution uncertainties was limited to acceptably low values.

The staff requested information in RAI 4.2-12 regarding the interpolation technique and associated uncertainties. The response to RAI 4.2-12 provided several interpolation techniques. The response to RAI 7.2-18 S02 superseded portions of the RAI 4.2-12 response to specify a revised interpolation technique (for additional information regarding the staff's technical review, see Items B.1.2 and B.35 in Appendix B to the SE).

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[27] This condition is incorporated in Section 4.5 of the GT LTR through response to RAI 7.2-59S2.

[[

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[[

]]

The staff considers the Plant E MOC9 axial power shape to represent a particularly difficult power shape to adapt to with discrete GT signals based on the high degree of complexity in the shape. The axial power shape includes two distinct axial power peaks. These challenging power shapes are not expected for the ESBWR based on the shorter core height; however, qualification against these TIP data provides a reasonable basis for testing the efficacy of the interpolation technique.

The interpolation technique utilized in the GT CMS for power shape adaption directly affects the capability of the GT CMS to monitor the peak nodal powers and axial power shape and will impact the ability of the GT CMS to adequately monitor thermal margin. The staff reviewed the determination of the component uncertainties in the OLMCPR and MLHGR limit in Section 3.5 of this report. The staff noted, however, that the thermal limits are based on the adequacy of the GT CMS software; therefore, the uncertainties may not be acceptable for different interpolation and adaptive methods. Thus, the staff limits its approval to [[ ]] and considers changes to the adaption technique to be a departure from a method of evaluation in the safety analysis, as described in the following condition:

> **Adaption Method Condition**[28]
>
> The NRC staff considers modifications to the adaption technique in the PANAC11-based GT CMS, described in NEDE-33197P-A, to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval.

On the basis of qualification against challenging reactor power shapes, as discussed above, and the determination of the interpolation uncertainty as documented in Section 3.5 of this report, the staff finds that the adaption and interpolation techniques are acceptable.

3.3.10 Local Power Range Monitor Calibration

Section 5.1 of the GT LTR (Ref. 2) describes the LPRM calibration process. The staff has reviewed this section and finds that the process is essentially identical to the process employed for the operating fleet of plants. The primary difference is that the LPRMs are calibrated to an adapted power shape based on the GT readings as opposed to TIP readings. The staff reviewed the efficacy of the GT CMS to adapt the core power shape in Section 3.3.9 of this SE

[28] This condition is consistent with Section 8.3.2 of the GT LTR, incorporated through response to RAI 7.2-71.

and finds the approach acceptable. The response to RAI 7.2-18 S02 provides updated uncertainty analyses to capture the effects of any additional uncertainty introduced into the CMS from the adaptive technique (for additional information regarding the staff's technical review, see Item B.35 in Appendix B to the SE). Therefore, the staff finds that the process remains applicable and that the applicant appropriately tabulated the uncertainties (see Section 3.5 of this report) for the safety and operating limits. Therefore, the staff finds the methodology acceptable. Accordingly, RAI 7.2-18 is resolved.

3.3.11 Minimum Instrumentation Configuration and Statistical Control Method

The staff noted that the GT instruments are subject to failure during normal operation. The failure of GT instruments, therefore, must be detected and the instruments declared inoperable. The staff noted that it has separately addressed instrument failure caused by (1) the inability of the in-line heater to calibrate the GT and (2) reactor transients in Sections 3.3.5 and 3.3.8.2 of this report, respectively.

Additionally, the staff determined that if a sufficient number of the instruments failed, it would impair the ability of the GT CMS to perform its adaption and calibration functions. Accordingly, the staff requested additional information in RAIs 7.2-55 and 7.2-66 regarding instrumentation failures. The response to RAI 7.2-55 provides descriptive details of the statistical control methodology (for additional information regarding the staff's technical review, see Item B.42 in Appendix B to the SE). The response to RAI 7.2-66 describes the minimum instrumentation configuration (for additional information regarding the staff's technical review, see Item B.52 in Appendix B to the SE).

The staff accepts the statistical control methodology to identify failed GT instruments on the basis that it is essentially equivalent to the statistical control methodology currently approved and in use in the operating fleet of plants with TIP instruments. Based on the foregoing evaluation, RAIs 7.2-55 and 7.2-66 are resolved.

The staff accepts the determination of the minimum instrumentation configuration on the basis that the uncertainty analysis adequately determined and incorporated the uncertainty associated with the minimum instrumentation configuration, as documented in the response to RAI 7.2-18 S02 (for additional information regarding the staff's technical review, see Item B.35 in Appendix B to the SE).

The minimum instrumentation configuration forms part of the basis for the acceptance of the power distribution uncertainties. The power distribution uncertainties need to account for failed instruments, since increased instrument failures result in increased uncertainties in the CMS-predicted power distribution. The staff noted that the SAFDLs are established before cycle operation based on a limiting instrumentation configuration and the associated increase in power distribution uncertainties. Therefore, to meet the requirements of GDC 10 during normal operation, the capability to monitor the core power distribution must remain within the limits assumed in the safety and operating limits analysis. As discussed in Section 3.5 of this report, the staff reviewed the terms in the uncertainty analysis accounting for failed instruments. To ensure that the requirements of GDC 10 are met, the staff approves the NEDE-33197P and NEDC-33239P LTRs in this regard subject to the following limitation on the minimum instrumentation configuration:

Minimum Instrumentation Configuration Limitation[29]

The staff acceptance of the power distribution uncertainties in the OLMCPR analysis (see Section 3.5.2) and the MLHGR analysis (see Section 3.5.1) is limited to those conditions that meet the minimum instrumentation configuration described in the applicant's response to RAI 7.2-66.

3.3.12 Summary Regarding Gamma Thermometer Core Monitoring System

For the reasons set forth above, the staff finds that the combination of the in-plant qualification, the description of the core monitoring methods, and the conditions applied through this section provide an adequate basis to demonstrate that the GT CMS is capable of meeting the requirements of GDC 13 and GDC 10 in terms of core monitoring when appropriate uncertainty parameters are developed for unique aspects of the system and accounted for in the thermal margin assessment. The staff has reviewed these uncertainties in Section 3.5 of this report.

3.4 Summary of NRC Staff Confirmatory Calculations

The staff has performed independent confirmatory calculations to validate the core monitoring methods. Appendix A describes these calculations in greater detail. The staff performed two series of calculations to support the subject review.

The staff performed confirmatory calculations using MONTEBURNS (a code that couples the MCNP transport code with the ORIGEN depletion code). These calculations allowed the staff to compare the predictive capabilities of the TGBLA06 lattice physics code against higher order transport methods. The staff found that the TGBLA06 calculations of the infinite reactivity and void reactivity coefficient very closely agreed with the predicted MONTEBURNS results. The level of agreement provides additional assurance that the TGBLA06 code accurately models the detailed neutronic phenomena for the ESBWR fuel design over the anticipated range of exposure and void fraction (for additional information regarding the staff's calculations, see Section A.1 of Appendix A to this SE).

The staff's second set of confirmatory calculations was intended to assist the staff in its review of the use of GT to monitor the core power shape. In these calculations, the staff relied on MCNP simulation of a nuclear fuel bundle, control blade, and GT instrument tube. The calculation considered axial variation in bundle geometry and power shape (using various control blade insertion depths). The staff concluded that the GT is expected to yield an accurate measurement of the local power distribution. However, in the vicinity of the control blade, the tilt in the radial power shape leads to differences in the axial power shape and the trace of GT indications. The GT measurements, however, are adjusted by the J-factor to infer the local power distribution. The staff noted that in this process the core monitoring calculation inherently captures the radial power tilt induced by the control blade. The calculations, however, prompted the staff to assess the local geometry correction factor methodology for partially controlled nodes because of the use of fine motion control rod drives (FMCRDs) in the ESBWR design. On the basis of its review of the local geometry correction methodology, the staff finds that the modeling capabilities are sufficiently robust in terms of the phenomena that are taken into account to preclude any biases in GT indications arising from partially controlled nodes (for additional information regarding the staff's calculations, see Section A.2 of Appendix A to this SE).

[29] This condition is incorporated in Section 9.3.3 of the GT LTR through response to RAI 7.2-66.

Therefore, the staff finds that the confirmatory calculations provide additional assurance of the adequate performance of the core monitoring system calculational methods.

3.5 Safety and Operating Limit Uncertainties

GDC 10 requires that the reactor core and coolant, control, and protections systems be designed to assure that SAFDLs are not exceeded. The analysis methods and core monitoring hardware and software are used to monitor the margin to safety and operating limits to ensure that SAFDLs are not exceeded. The limits are established with consideration of any sources of uncertainty in the CMS to ensure adequate thermal margin during cycle operation. Of interest in the subject review are the MLHGR and OLMCPR. To determine the margin to these limits, the CMS calculates the power distribution and thermal hydraulic condition of the core.

The staff has reviewed the bases of these limits to ensure that the uncertainties in the analysis methods and CMS software are appropriate.

3.5.1 Linear Heat Generation Rate Uncertainty

The GT CMS monitors the LHGR on a nodal level. Periodically, [[ ]]. The nodal LHGR is compared to the exposure-dependent MLHGR limit to ensure that the thermal mechanical criteria are met during normal operation. When compared to the operating fleet CMS, the ESBWR MLHGR limit must account for aspects unique to the GT CMS.

The staff noted that constructing the APS from discrete signals, as opposed to continuous TIP measurements, lead to an increase in the uncertainty in the nodal LHGR. The staff compared the component uncertainties in the LHGR based on the values reported in Table 2-14 of the IMLTR (Ref. 9). In certain instances, equivalent uncertainty parameters were developed to account for the GT CMS. The LHGR uncertainty also incorporated a parameter to account for the introduction of additional uncertainty resulting from the interpolation scheme. The staff requested additional information regarding these uncertainty components in RAIs 4.2-12, 4.3-2, and 7.2-18 (for additional information regarding the staff's technical review, see Items 0, 0, and Appendix BB.35 in Appendix B to the SE, respectively).

The affected component uncertainties are the infinite lattice pin power peaking factor uncertainty, the [[ ]] (which replaces the TIP random uncertainty), the [[ ]], the [[ ]], and the [[ ]].

The response to RAI 4.3-2 provides the infinite lattice pin power peaking factor uncertainty. The applicant developed this uncertainty consistent with the 95UTL statistical method approved in Reference 10. This statistical method was found to adequately address potentially increased local power distribution uncertainties for plants operating at EPU conditions. The staff's review for the applicability to EPU conditions is documented in Reference 10. As discussed in Section 3.2.2.1 of this SE, the staff finds that this approach is also appropriate for the ESBWR.

The response to RAI 4.2-12 provides information regarding the GT uncertainties attributed to the interpolation. In large part, the response to RAI 7.2-18, which updates the interpolation scheme, supersedes the response to RAI 4.2-12. However, RAI 4.2-12 provides the basis for the [[ ]]. This uncertainty is based on a combination of the historical TIP

uncertainty with an additional term based on the GT to TIP comparisons performed during the Limerick in-plant test. The staff finds this approach acceptable to capture the [[ ]] (for additional information regarding the staff's technical review, see Item B.1 in Appendix B of the SE).

The response to RAI 7.2-18 S02 provides the [[ ]], and [[ ]] for the LHGR. The staff reviewed these bases and finds them to be consistent with the previously approved approach for a TIP-based CMS, as described in References 27 and 28, with appropriate modifications to account for the unique ESBWR GT CMS. The GT interpolation uncertainty is an additional uncertainty component to account for lost information relative to the TIP system. The basis for this uncertainty incorporates qualification against operating plant data and challenging power shapes. The response compares calculated power shapes based on discrete measurements to detailed axial TIP data. The staff finds the basis for this uncertainty component is sufficiently comprehensive and relevant and therefore acceptable (for additional information regarding the staff's technical review, see Item B.35 in Appendix B of the SE).

In its review of the [[ ]], the staff considered the appropriateness of the LPRM calibration frequency. The response to RAI 7.2-18 S02 provides for a reduced LPRM calibration interval from 1,000 megawatt-day per metric tonne (MWD/T) to 750 MWD/T to reduce this component uncertainty. The staff finds that reducing the calibration interval is an acceptable means for reducing this uncertainty component and is consistent with the open item associated with RAI 7.2-18 from the staff audit of the GT CMS described in Reference 25. The following condition will ensure that the LPRM calibration interval is consistent with the reduced interval provided in the response to RAI 7.2-18 S02 to justify the current uncertainty analysis.

LPRM Calibration Interval Limitation[30]

The LPRM calibration interval cannot exceed 750 MWD/T.

The staff clarifies that its findings regarding acceptability of the values of the uncertainties documented in NEDE-33197P for the GT CMS are valid only for an LPRM calibration interval that does not exceed 750 MWD/T.

Table 3-1 summarizes the LHGR uncertainty components. When combined[31], the total LHGR uncertainty is [[ ]]. This is essentially the same as the [[ ]] assumed in the analysis of the MLHGR limit. Therefore, the staff finds that the uncertainty used in the MLHGR limit is appropriate to account for the GT CMS uncertainties.

[30] This condition is consistent with Section 9.3.3 of the GT LTR and DCD Chapter 16 SR 3.3.1.4.4 and 3.3.1.4.5

[31] [[ ]]

Table 3-1 Component Uncertainties in the Linear Heat Generation Rate

LHGR Uncertainty for MLHGR Limit	*Units of Percent*	*Reference for Technical Basis*
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	

3.5.2 Operating Limit Minimum Critical Power Ratio Uncertainties

Reference 16 documents the OLMCPR determination process. The staff noted that this methodology differs from the previously approved OLMCPR determination process, which relies on the calculation of the SLMCPR a priority. However, the revised determination process must consider the same component uncertainties to ensure adequate thermal margin. Table 2-1 of the IMLTR (Ref. 9) describes component uncertainties considered in developing the SLMCPR for EPU reactors. Of these uncertainties, two are subject to the calculational performance of the GT CMS: the bundle power uncertainty and the R-factor uncertainty.

The CMS calculates the bundle power and the R-factors which are used to determine the critical power ratio of the bundles during cycle operation. The individual bundle powers and the local rod peaking factors used to develop the bundle R-factors are subject to calculational inaccuracies introduced by the CMS. Therefore, the staff reviewed the applicability of historically determined component uncertainties of the bundle power uncertainty and the R-factor uncertainty. The staff also reviewed any revised components unique to the GT CMS.

While the staff relied on previous review experience, many of its previous review findings were not directly applicable based on the revised OLMCPR determination process. Therefore, as documented in Section 3.2.2.1 of this SE, the staff modified certain conditions and limitations previously imposed with respect to SLMCPR uncertainties to ensure compatibility with the revised uncertainty determination process.

3.5.2.1 Bundle Power Uncertainty

The staff reviewed the component uncertainties contributing to the overall measured bundle power uncertainty used in the downstream OLMCPR determination process. In its review, the staff considered equivalent component uncertainties similar to the TIP CMS uncertainties quoted in Reference 28. In response to RAI 7.2-18 S02 (for additional information regarding the staff's technical review, see Item B.35 in Appendix B to the SE), GEH provided an assessment of equivalent component uncertainties by performing analyses with a GT-based CMS model. In the case of the ESBWR, the two components affected by the introduction of the GT CMS are the [[ ]] and the [[ ]]. The response to RAI 7.2-18 S02 provides analyses of these uncertainty components that are analogous to the analyses performed to assess the equivalent TIP CMS-based uncertainties. The staff finds the approach used to be acceptable. The [[ ]] is treated in both the TIP CMS and GT CMS. Because the LPRM system is essentially unchanged relative to the operating fleet, the staff accepts that the [[ ]] component is also applicable to the ESBWR.

The staff noted that the [[ ]] referenced in the bundle power uncertainty is identical to the value approved by the staff for the operating fleet of plants operating at EPU conditions (Refs. 9 and 10) and is therefore acceptable. The [[ ]] used in the analysis is consistent with the staff condition regarding the [[ ]] documented in Section 3.2.2.1.1 of this SE.

Table 3-2 summarizes the bundle power uncertainty and its components. The staff finds that use of the bundle power uncertainty provided in Table 3-2 of this SE is acceptable for generating the OLMCPR because it accounts for the GT CMS and ESBWR specific component uncertainty parameters.

Table 3-2 Component Uncertainties in the Bundle Power Uncertainty

Bundle Power Uncertainty for OLMCPR	*Units of Percent*	*Reference for Technical Basis*
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	

3.5.2.2 R-Factor Uncertainty

The second component uncertainty affecting the OLMCPR determination process relevant to the power distribution monitoring performed by the GT CMS is the R-factor uncertainty. The R-factor uncertainty in the IMLTR is the same as the R-factor uncertainty quoted in References 27 and 28. However, the applicant performed sensitivity analyses to determine the SLMCPR effect of an increased R-factor uncertainty consistent with an infinite lattice power peaking factor uncertainty determined using the [[ ]]. In Reference 10, the staff conditioned its approval of IMLTR on an increase in the SLMCPR to account for potentially increased uncertainty in the local power peaking. Such a condition imposed on the ESBWR would not provide additional thermal margin because the OLMCPR would be determined using the historical uncertainties and the SLMCPR would be back calculated and then artificially increased without a commensurate increase in the actual thermal margin.

The staff, in Section 3.2.2.1.1 of this SE, conditioned its approval of NEDC-33239P in regard to the determination of the R-factor component uncertainty. This condition provides that the [[ ]] be applied at the onset of the OLMCPR calculation to ensure that the operating thermal margins are appropriate. The R-factor component uncertainty is conservatively set at [[ ]] for the ESBWR. This value provides significant conservatism relative to the R-factor uncertainty quoted in the IMLTR (Ref. 9). Therefore, the generic [[ ]] for the ESBWR is conservative relative to an R-factor uncertainty generated using the revised infinite lattice pin power peaking factor uncertainty, and is acceptable.

4. CONDITIONS, LIMITATIONS, AND RESTRICTIONS

This section of the SE provides a consolidated listing of the conditions, limitations, and restrictions documented in the body of this SE.

4.1 [[ ]] Condition (Section 3.2.2.1.1)

The [[ ]] is a component of the linear heat generation rate (LHGR) and OLMCPR calculation uncertainties. Its value is determined using a [[ ]] statistical analysis on gamma scan data. NEDC 33173P-A reports the value determined using this approach as [[ ]].

The applicability of this condition is dictated by the [[ ]] approved by the NRC in NEDC-33173P. Should the NRC approve an alternative approach for establishing the aforementioned uncertainties in subsequent supplements to or revisions of the NEDC 33173P LTR, the approved, alternative approach may be adopted in NEDE 33197P-A in lieu of this condition without separate NRC review and approval.

[[ ]] of the uncertainty value must be submitted to the NRC before the change is incorporated into any safety analysis basis.

4.2 Peaking Factor Uncertainty for MLHGR Condition (Section 3.2.2.1.1)

The LHGR infinite lattice peaking factor uncertainty value is determined as the [[ ]] using the statistical analysis of the population of peak power as a function of exposure. The GE14E-specific LHGR infinite lattice peaking factor uncertainty determined using this approach is [[ ]]. This uncertainty will be determined whenever a new fuel product is applied to a particular ESBWR core loading.

The applicability of this condition is dictated by the [[ ]] approved by the NRC in NEDC-33173P. Should the NRC approve an alternative approach for establishing the aforementioned uncertainties in subsequent supplements to or revisions of the NEDC 33173P LTR, the approved, alternative approach may be adopted in NEDE 33197P-A in lieu of this condition without separate NRC review and approval.

Any reduction of the uncertainty value must be submitted to the NRC before the change is incorporated into any safety analysis basis.

4.3 Peaking Factor Uncertainty for OLMCPR Condition (Section 3.2.2.1.1)

NEDC-32601P-A describes the method for calculating the R-factor uncertainty. When determining the R-factor uncertainty for ESBWR analyses, the infinite lattice peaking model uncertainty value will be assumed as equal to/or more conservative than, the LHGR infinite lattice peaking factor uncertainty value for a particular ESBWR core loading.

Any reduction of the uncertainty value must be submitted to the NRC before the change is incorporated into any safety analysis basis.

4.4 Peaking Factor Uncertainty and Fuel Exposure Condition (Section 3.2.2.1.1)

The LHGR infinite lattice pin power uncertainty must represent the full range of fuel lattice exposure values. The calculated peak pellet exposure must be confirmed by GEH or the licensee referencing the LTR to comply with the corresponding licensing limit approved by the NRC. The design analysis described in NEDC-33242P establishes the licensing limit for GE14E (Ref. 11).

4.5 R-Factor Condition (Section 3.2.2.1.2)

The bundle R-factor must be calculated using representative lattice pin power distributions and axial void and power profiles.

4.6 TGBLA06 8-Weight-Percent Gadolinia Restriction (Section 3.2.2.1.3)

TGBLA06 is not approved to analyze fuel lattices with gadolinia burnable poison loadings in excess of 8 weight percent gadolinia until the NRC staff quantifies and reviews the gadolinia bias.

4.7 Steady-State 5-Percent Bypass Voiding Limitation (Section 3.2.2.1.4)

The bypass voiding will be evaluated on a cycle-specific basis to confirm that the void fraction remains below 5 percent at all LPRM levels when operating at steady-state conditions at the upper boundary of the allowable operating domain.

4.8 Mixed Oxide Fuel Restriction (Section 3.2.2.1.5)

TGBLA06 is not approved to analyze mixed oxide fuel until the methods are qualified for this application and the uncertainties are reassessed and submitted to the NRC for review and approval.

4.9 Bundle Isotopic Tracking Model Restriction (Section 3.2.2.1.6)

The staff did not perform a review of the bundle isotopic tracking model. Therefore, staff approval of NEDC-33239P does not constitute approval of the bundle isotopic tracking model.

4.10 Bypass Flow Lookup Table Condition (Section 3.2.2.2.2)

Licensing evaluations performed with PANAC11 must use bypass flow fractions consistent with all core operating states, as determined by TRACG04, and input in the core simulator to accurately determine the bypass flow. Bypass flow tables or explicit modeling of data from TRACG04 can be used for PANAC11 input values.

4.11 PANAC11 Three-Dimensional Shutdown Margin Condition (Section 3.2.2.4)

The ESBWR SDM must be performed using three-dimensional methods. The two-dimensional rod worth estimation technique is not approved for licensing analysis other than as a scoping tool to identify potentially limiting control blades or control blade pairs.

4.12 Scram Reactivity Calculation Condition (Section 3.2.2.4)

The scram reactivity calculated using the PANAC11 neutronic solver must be calculated with Doppler reactivity feedback modeling activated to accurately determine the reactivity effect of the blades without including the Doppler reactivity in the scram reactivity.

4.13 Boron Branch Limitation (Section 3.2.2.4)

For the standby liquid control system shutdown analysis, the TGBLA06 borated libraries must be generated with lattice boron inventories between 600 ppm and 1,000 ppm natural boron equivalent.

4.14 Void Exposure History Bias Condition (Section 3.2.3.1.1)

Use of PANAC11-generated nuclear data for ESBWR reload transient analyses (AOO, stability, or ATWS) requires that TRACG utilize the void reactivity coefficient correction model described in NEDE-32906P-A, Supplement 3. The fuel lattices input to the model must represent the cycle-specific fuel loading.

4.15 Flux Harmonic Condition (Section 3.2.3.2)

The regional mode stability analysis must be performed using a radial nodalization in TRACG04 based on the PANAC11-generated first harmonic mode. The harmonic calculation performed by PANAC11 must use a full-core representation.

4.16 Code Usage Condition (Section 3.2.4)

The limitations on TGBLA06 and PANAC11 code usage, as described in the user manuals, are a condition of the acceptance of these methodologies for the ESBWR. Changes to the manuals that are made in accordance with the quality assurance procedures audited by the staff, as documented in the applicable reference, do not require NRC review and approval. However, if used in the safety analysis, the cycle-specific SRLR must document these changes.

4.17 Code Change Limitation (Section 3.2.4)

- The NRC staff considers modifications to the models described in NEDC-33239P-A or MFN 098-96 to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval.

- The NRC staff considers modifications to the TGBLA06/PANAC11 codes that result in inconsistency with the NEDC-33239P-A LTR to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval of the necessary revisions to the LTR.

- The NRC staff considers modifications to the TGBLA06/PANAC11 codes or the GT CMS software that result in inconsistency with the NEDE-33197P-A LTR to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval of the necessary revisions to the LTR.

- The NRC staff does not consider updates to the PANAC11 nuclear methods to ensure compatibility with other NRC-approved methods to constitute a departure from a method of evaluation in the safety analysis. These updates may be used for licensing calculations without prior NRC review and approval so long as the predicted ESBWR

equilibrium cycle MLHGR or the downstream ΔCPR/ICPR for the potentially limiting transients (calculated by TRACG04) show less than a 1-standard-deviation difference.

- The NRC staff does not consider increases in the spatial or energy resolution in the TGBLA06 lattice physics method to constitute a departure from a method of evaluation in the safety analysis. These updates may be used for licensing calculations without prior NRC review and approval so long as the uncertainties in the lattice parameters do not increase as a result. In all cases, the cycle-specific SRLR, if used in the safety analysis, must document modifications or updates done without prior NRC review and approval.

- The NRC staff does not consider changes in the numerical methods to improve code convergence to constitute a departure from a method of evaluation in the safety analysis, and they may be used in the licensing calculations without prior NRC review and approval.

4.18 GT-Specific [[ ]] Values Condition (Section 3.3.4.1)

The CMS must track individual [[ ]] values for each GT in the core.

4.19 Local Geometry Refinement Model Condition (Section 3.3.4.3)

The parameters used to compensate for biases introduced in the GT sensor signal by the proximity to spacers or fuel type changes or both will be determined only when a new bundle design (i.e., new axial lattice composition) or a new spacer design (i.e., material) is applied to a particular ESBWR core loading, as described in Section 8.6 of NEDE-33197P-A. The parameters will be incorporated into the GT-based monitoring system on a cycle-specific basis, as required.

4.20 GT In-Line Heater Wire Current Hold Time Condition (Section 3.3.5)

GT calibration must be performed using a current hold time of at least five GT thermal time constants per current magnitude per string.

4.21 GT Heater Wire Exposure-Dependent Resistance Condition (Section 3.3.6.1)

If changes are deemed appropriate to improve accuracy by accounting for GT heater wire resistance changes during irradiation, implementation of the proposed method outlined in the applicant's response to RAI 7.2-7 requires revision to NEDE-33197P. The NRC staff considers this to constitute a departure from the method of evaluation that has not been reviewed by the NRC for the intended application.

4.22 Sensitivity Decrease Model Restriction (Section 3.3.6.2)

The sensitivity decrease model is not approved. Therefore, when GT are used for the purpose of LPRM calibration or adaption, they must be calibrated beforehand using in-line heater calibration to determine the sensitivity.

4.23 GT Operability Condition (Section 3.3.6.2)

The failure of a GT heater is considered a loss of calibration capability of the full GT string (all sensors). Therefore, in case of failure of a GT heater, the GT CMS will declare the GT string as inoperable.

4.24 Delayed Gamma Compensation Model Restriction (Section 3.3.8.1)

The delayed gamma compensation model is not approved. Therefore, GT calibration and the use of GT for LPRM calibration or adaption purposes may only be performed during a steady-state condition of operation that meets the core requirement described in Section 4.5 of the GT LTR.

4.25 Adaption Method Condition (Section 3.3.9)

The NRC staff considers modifications to the adaption technique in the PANAC11-based GT CMS, described in NEDE-33197P-A, to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval.

4.26 Minimum Instrumentation Configuration Limitation (Section 3.3.11)

The staff acceptance of the power distribution uncertainties in the OLMCPR analysis and the MLHGR analysis is limited to those conditions that meet the minimum instrumentation configuration described in the applicant's response to RAI 7.2-66.

4.27 LPRM Calibration Interval Limitation (Section 3.5.1)

The LPRM calibration interval cannot exceed 750 MWD/T.

5. CONCLUSIONS

The staff has completed its review of the NEDC-33239P and NEDE-33197P LTRs. For the reasons set forth throughout this SE, the staff finds the core monitoring methods described in these LTRs to be acceptable subject to the conditions listed in section 4 of this report. The staff's conclusions are based on a detailed review of the applicant's methods and experimental qualification programs, and the staff's independent confirmatory calculations.

Because the NRC staff has reviewed the subject LTRs, it does not intend to review the associated LTRs when referenced in licensing evaluations. However, the staff only finds the methods applicable when exercised in accordance with the limitations and conditions described in Section 4 of this SE. Further, for the reasons set forth in this SE, GEH has demonstrated that, when exercised appropriately, the methods documented in References 1 and 2 are adequate to ensure that the design complies with the applicable GDC, with respect to the ESBWR design basis as documented in the DCD. Subject to the identified conditions, the nuclear methods are acceptable to perform those calculations aimed at assessing the safety of the nuclear design of the core and the GT CMS is acceptable to performing core simulation and thermal limits monitoring.

6. REFERENCES

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APPENDIX A
CONFIRMATORY CALCULATIONS PERFORMED BY U.S. NUCLEAR REGULATORY COMMISSION STAFF

A.1 Lattice Physics Confirmatory Calculations

The U.S. Nuclear Regulatory Commission (NRC) staff performed confirmatory nuclear calculations for the economic simplified boiling-water reactor (ESBWR). The purpose of these calculations was to confirm (1) the extrapolation of nuclear data calculations made by GE-Hitachi Nuclear Energy Americas LLC (GEH or the applicant) to high void fraction (90 percent), (2) the efficacy of the GEH code's depletion capability to determine reactivity trends at high void fraction, and (3) the accuracy of GEH's calculation of the void reactivity coefficient.

The staff performed its calculations using MONTEBURNS. MONTEBURNS is a code that couples the Monte Carlo N Particle (MCNP) transport code with the ORIGEN depletion code. The confirmatory analysis considered various lattices encompassing those lattices reported in Reference 1. The MONTEBURNS calculations were only capable of depleting 49 simultaneous material regions. To address this limitation in the calculations, the staff performed several sensitivity studies to verify the results. While a typical fuel lattice includes 92 fuel rod locations, symmetry arguments allow for detailed representation of the depletion. This limitation, however, presents particular difficulty in the simulation of gadolinia-bearing rod depletion given the steep thermal flux gradients across the pellets and the subsequent "onion skin" effect where the gadolinia depletes more rapidly in the outer radial regions of the fuel pellet before the gadolinia in the central region of the pellet depletes.

To simplify the depletion calculation, the staff made various approximations, while ensuring a level of accuracy commensurate with the intent to independently verify the nature of the depletion solution in the GEH analytical methods. To this end, the staff evaluated the approximate depletion arrangements using a number of sensitivity studies to determine an optimized approximation. Due to limitations in MONTEBURNS the staff was limited in the number of available depletion zones in the lattice calculation. The staff noted that the approximations may lead to differences in the staff's and GEH calculations, particularly at higher exposure. Noting the approximations for the depletion regions in the fuel geometry, the staff determined that the calculation method provides a robust transport-based flux calculation while allowing an independent means to gauge the capability of the TGBLA06 code to calculate the nuclear data for the ESBWR with an independent depletion solution. Given the staff's approximations in the depletion, the staff did not expect a high degree of agreement between the two methods, however, the staff used the results to verify expected trends.

This appendix includes representative cases from the staff's calculations. Figure A.1.1 compares the lattice infinite multiplication factor calculated using the "BNL [Brookhaven National Laboratory] standard model" and the TGBLA06 code. The case presented reflects a 40 percent depletion history and the dominant (DOM) zone lattice. This lattice includes a fully rodded (92 fuel rods) configuration, with 13 gadolinia-bearing rods. For this mid-void depletion, the TGBLA06 and staff results very closely agree at the beginning of life. The BNL standard model and the TGBLA06 code predict the peak reactivity and the exposure of the peak reactivity very closely. The staff does not expect the two cases to exactly predict the peak reactivity exposure based on limitations of the BNL standard model in terms of the depletion zones. However, the staff considers the close agreement between the two cases an indication that the TGBLA06

calculations are performing adequately in terms of the calculation of the lattice reactivity and depletion.

Figure A.1.2 is based on Reference 3 and depicts the results of the staff's and GEH calculations of the DOM lattice depletion at very high void fraction (90 percent).

The comparison for the DOM lattice in this case is very similar to the 40 percent void fraction depletion history. Thus, the staff has confidence that the solution techniques in TGBLA06 are robust over this void range. The reported results include several more cases for many lattices that characterize the entire fuel bundle geometry over a range of void fraction from 0 percent to 90 percent. The results presented in Figures A.1.1 and A.1.2 are typical of the results depicted for the other lattices.

The staff also compared the calculation of the void reactivity coefficient to independent confirmatory calculations. The void reactivity coefficient is calculated by perturbing the lattice calculations at any exposure point instantaneously by changing the void fraction and then calculating the void reactivity coefficient based on the differences in the infinite lattice multiplication factor. The staff's calculation used the mean gradient about the point of interest to predict the void reactivity coefficient. That is, while the GEH results for the void reactivity coefficient are derived from a polynomial fit to lattice branch calculations, the staff's calculations are performed by averaging two linear gradients about the void fraction of interest.

The analysis compared the void reactivity coefficient as a function of exposure. Figure A.1.3 compares the void reactivity coefficient predicted by MCNP and TGBLA06 for the DOM lattice at high void (70 percent). The calculations indicate that the TGBLA06 and MONTEBURNS codes predict a void reactivity coefficient of similar magnitude at various exposure points. For this particular case, the staff's code predicts a more negative void reactivity coefficient at high burnup; however, for lower burnups, the results agree very well. The staff also noted that there is good agreement between the codes in terms of the change in void reactivity coefficient with burnup. As illustrated in Figure A.1.3, the TGBLA06 and the staff's curves are very similar in character and magnitude with some differences in the exposure point of the curve deflections. Some differences of this type and magnitude are expected based on inherent approximations in the BNL standard model approach for gadolina-bearing fuel. These trends are characteristic for the fueled regions of the bundle.

Figure A.1.4 depicts the staff's results for the natural uranium blanket zone (NAT) at the bottom of the fuel bundles (without gadolinia-bearing fuel). The results provided indicate very close agreement between the two methods in terms of the magnitude of the void coefficient and the evolution of the void reactivity coefficient with exposure. This provides the staff with assurance that, under conditions in which the fuel depletion may be more explicitly modeled in the staff's model, the two methods are in very good agreement.

The staff opted to incorporate Figures A.1.3 and A.1.4 because these results are representative of the figures in Reference 3.

The staff's analyses largely confirm the GEH results. The staff considers the agreement between theirs and GEH's results particularly significant given that they were derived from two completely distinct methods. The MCNP transport solution is a very detailed solution based on Monte Carlo transport, whereas the TGBLA06 solution is based on a simpler collision probability transport method.

The variation in multiplication factor with exposure using either code was in good agreement, confirming the ability of the TGBLA06 code to extrapolate to high void fractions and to solve the eigenvalue equations at these conditions.

The trends in both the infinite multiplication factor and the void reactivity coefficient with exposure were in very good agreement, providing simultaneous confirmation of the TGBLA06 calculation of the depletion and the reactivity feedback at various void conditions, including high void fraction.

The staff's confirmatory calculations provide additional reasonable assurance that TGBLA06 provides acceptable nuclear data results for ESBWR calculations.
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Figure A.1.1 Infinite lattice multiplication factor for representative DOM lattice at 40-percent void

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Figure A.1.2 Infinite lattice multiplication factor for representative DOM lattice at 90-percent void

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Figure A.1.3 Void coefficient comparison for representative DOM lattice at 70-percent void

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Figure A.1.4 Void coefficient comparison for representative NAT lattice at 40 percent void

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A.2 Gamma Transport Confirmatory Calculations

The NRC staff performed confirmatory calculations of the expected gamma thermometer (GT) performance using detailed MCNP simulations. The purpose of performing these calculations was to (1) confirm the heat deposition in the GT insulated section and (2) predict the expected GT response to axial power perturbations induced by control blade insertion.

The staff conducted the confirmatory analysis in two stages. The first stage considered a two-dimensional simulation of a color set. A color set refers to a two-dimensional nuclear simulation of four fuel assemblies. The color-set model considers a reflected geometry with four lattices surrounding a central GT assembly. The staff performed the analysis using a coupled neutron-photon MCNP simulation. The analysis included tallies for the GT insulated section regions to determine the expected heat deposition in this section resulting from the gamma and neutron flux.

The analysis confirmed the expected heat deposition rate in the insulated section consistent with the specifications developed by the applicant and with NEDE-33197P, "Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring" (Ref. 1) (hereafter referred to as the GT LTR). The analysis also predicts that as much as 10 percent of the heat deposited in the insulated section may come from neutron heating. This is predominantly attributed to neutron heating of the alumina insulator. The two-dimensional model treats the cable pack as being entirely insulator, which results in an artificially high fraction of heating coming from neutron heating.

According to the design information in the GT LTR, [[ ]]. The cable pack material is made from [[ ]]. Since neutrons heat alumina to a greater extent than [[ ]] due to the high concentration of lighter nuclei (oxygen in this case), this leads to an overestimate of the total heat deposition in the instrument from neutron irradiation.

Therefore, the staff finds that, at most, 10 percent of the heat deposited in the instrument will be from neutron irradiation. Considering that the instrument model includes a significant amount of alumina relative to the design information, the staff expects that the actual neutron heating will be a smaller fraction. The GT LTR describes the design as containing [[ ]] which is very small relative to the cable pack. The alumina present in the GT design is approximately 10 percent of the alumina represented in the staff's MCNP analysis. Because neutrons do not efficiently heat the stainless steel, the expected neutron heating fraction could be estimated at 10 percent of 10 percent, or 1 percent. This approximation demonstrates that the neutron heating is negligible.

However, even considering the artificially increased neutron contribution, the staff finds that neutron heating on the order of 10 percent or less is not expected to introduce significant biases in the GT local gamma flux indication. While the neutron and gamma transport characteristics across the bundles will be different, the local combination of the gamma and neutron fluxes should scale together because both fluxes are essentially proportional to power.

The potential to introduce instrument error arises because the neutron and gamma transport characteristics across the bundle differ. For instance, the neutron penetration is highly dependent on the absorption cross-section, while the gamma penetration is highly dependent on the electron density. Therefore, assuming that the heat is entirely deposited by gamma heat

affects the translation of the measured signal to bundle power. However, in either case, the instrument is most sensitive to the power in the pins nearest the instrument (this is true for the case of neutron and gamma sources). Because the pin neutron and gamma sources scale directly with fission power, the translation of the heat deposition through gamma transport factors to the pin powers (or vice versa) will only introduce a bias insofar as the relative pin transport factors differ for either neutron or gamma radiation.

For the degree of neutron heat (less than 10 percent), and noting that the gamma flux and neutron flux scale together and the transport characteristics are similar for fuel bundle geometries, the staff does not find that this effect introduces a discernable bias. The staff's conclusion is further supported by the in-plant qualification data demonstrating the GT performance and capability to approximate the measured axial power shape. As the influence of neutron heating would be discernable in the measurements performed for any BWR, the staff finds that these results are applicable to the review of the GT CMS for the ESBWR.

In the second stage of the analysis, the staff modeled a controlled bundle with GT instruments using three-dimensional MCNP calculations. The three-dimensional model was radially reflected, thus modeling an infinite array of fuel bundles while allowing axial neutron leakage. The analysis considered variation in axial void fraction and power using stacks of lattices. The purpose of this analysis was not necessarily to accurately model the ESBWR axial shape, but to introduce a "realistic" axial variation in neutronic and transport properties to test the efficacy of the GT instruments to measure variation in axial power. The axial power shape was perturbed externally using a control blade that could be inserted or withdrawn opposite the instrument corner in the model.

The staff performed the three-dimensional calculations for various control blade insertions. The staff was particularly interested in the potential for the control blade insertion to introduce biases in the GT instrument reading. The study included analyses for six control blade insertion depths. Figures 5 through 9 of Reference 2 illustrate the results.

For the shallow insertion depth, the axial power shape is mid-peaked. The results show that the GT instrument reading and the nodal axial power shape are in very close agreement above the control blade. For the nodes below the control blade tip, there is an apparent bias in the GT reading. The staff, however, noted that the plots compare the axial fission power and the axial indications of GT heat deposition. Figure A.2.1 depicts the shallow insertion calculation results.

The following figures illustrate the differences between the axial power shape and the GT readings below the control blade tip. In certain instances, the difference between the GT heat deposition and the power is substantial (50 percent). The staff considered the GT instrument biases and attributes the differences to radial power tilting in the assembly due to the presence of the control blade.

When the control blade is inserted in a node, the assembly power will radially tilt away from the blade and result in high local peaking in the instrument corner. Overall, the nodal power will decrease due to the blade; however, the instrument corner is least affected by the blade, and the power in this corner will remain high relative to the balance of the node. Therefore, even as the blade reduces the nodal power, the GT will still respond to power generated in the pins closest to the instrument corner.

The staff concluded that the GT instruments are expected to perform adequately when the control blades are withdrawn during normal operation.

The staff, however, noted that the ESBWR reactivity control is accomplished through a mixture of temperature variation and control blade insertion. During cycle operation, various rods are inserted to various depths depending on the exposure to maintain criticality.

While the staff's independent confirmatory calculations have characterized the potential for the control blades to introduce biases in the GT readings, the staff noted that the J-factor methodology [[ ]] to determine the instrument response. The J-factor, therefore, accounts for radial power peaking introduced by the explicit introduction of the control blade.

In other words, the staff does not necessarily expect the GT reading, established by the heat deposition in the sensing region, to match exactly the assembly power. The close agreement between the GT reading and the axial power shape above the control blade tip provides additional assurance that the GT will accurately predict the local power. However, this is most likely an artifact of the radial reflection assumptions in the analysis which reduce the potential radial power tilting in the assembly due to effects, such as buckling, that result from the specific core design and fuel loading.

The bias introduced in the GT reading by the presence of the control blade is a function of the radial power shift in the assembly. The staff has reviewed GEH's core monitoring software and the J-factor methodology to confirm that, when the GT readings are translated into an indication of the local power, this methodology captures the effect of the radial pin power distribution explicitly. The staff's calculations confirm the expected performance of the GT instruments for controlled conditions and these calculations verify that need for the J-factor methodology [[ ]].

For partially controlled nodes, the J-factors are combined according to the local geometry correction methodology provided in the response to Request for Information 7.2-72 (for additional information regarding the staff's technical review, see Item B.54 in Appendix B to this safety evaluation). This methodology accounts for the potential for the fine motion control rod drive to position a control blade partially through a node near a GT sensor.

Overall, the staff's confirmatory calculations provide additional assurance that the applicant's characterization of the GT in-plant operation is consistent with the expected heating of the instrument. Similarly, the confirmatory calculations indicate that the J-factor approach considers the appropriate phenomena to account for the axial power variation introduced by the control blades. This is further confirmed by Limerick 2, Tokai 2, and K5 in-plant tests showing good agreement between the GT "measured" local power and the thermal traversing in-core probe "measured" local power distributions in the GT LTR. In this case, the staff uses the term "measured" to mean that these distributions are the reactor power distributions inferred from the instrument readings.

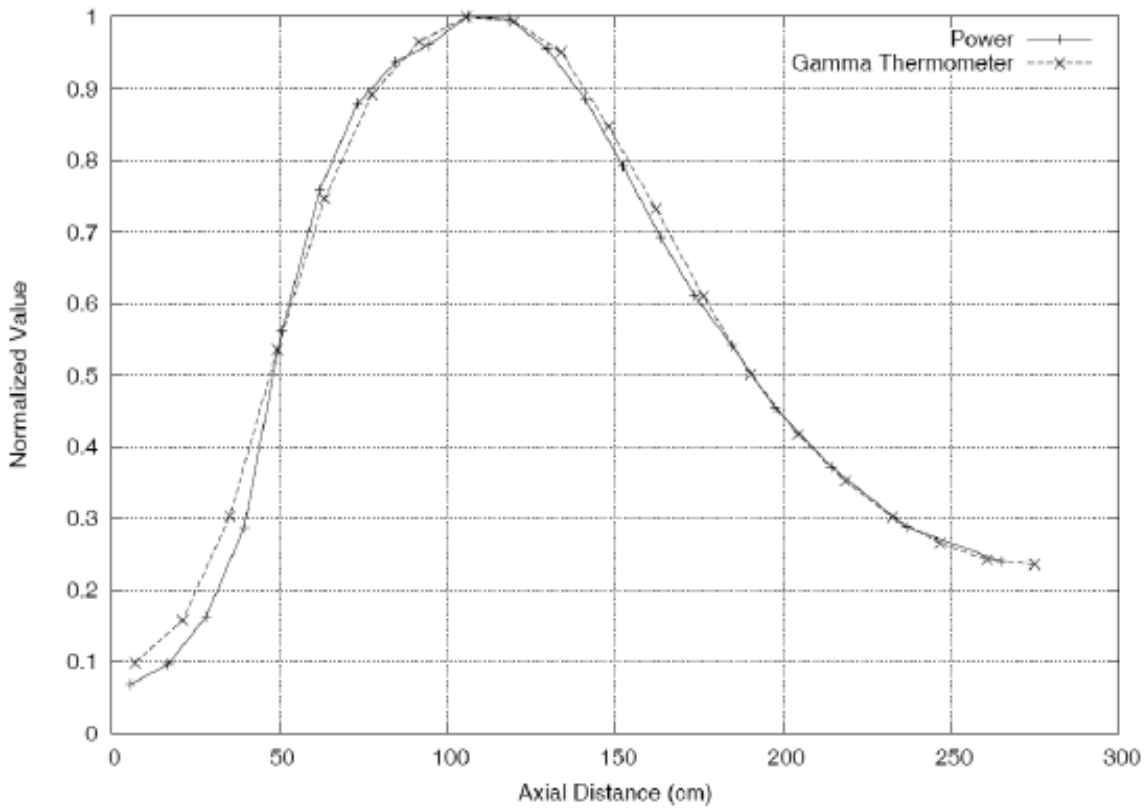

Figure A.2.1 Normalized GT response and power shape (control blade at 45.11 cm)

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APPENDIX B
STAFF REVIEW OF APPLICANT RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

This appendix provides the detailed findings of the U.S. Nuclear Regulatory Commission (NRC) staff review of the GE- Hitachi Nuclear Energy Americas LLC (GEH or the applicant) responses to the staff's requests for additional information (RAIs) regarding the Economic Simplified Boiling Water Reactor (ESBWR). In many cases, initial RAI responses did not provide sufficient detail for the staff to complete its review. This appendix also discusses supplemental RAIs and describes the final closure of all open items. The staff noted that GEH implemented significant methodology changes during the course of the NRC review of the gamma thermometer (GT) core monitoring system (CMS). In certain cases, these methodology changes rendered previous RAI responses irrelevant. This appendix summarizes these cases as a means of documenting the closure of all RAIs.

The staff noted that the following RAIs were rendered irrelevant, partially superseded, or fully superseded by information provided in other RAI responses: 4.3-1, 4.2-12, 7.2-14, 7.2-16, 7.2-17, 7.2-51, 7.2-52, 7.2-53, 7.2-54, 7.2-55, 7.2-56, 7.2-58, 7.2-60, 7.2-62, and 7.2-64.

The Appendix B sections are divided into sections for each RAI or in some cases for multiple RAIs that address the same general topical area. For RAIs with multiple parts or supplements, the section is subdivided. Each Appendix B section contains an individual reference section. In most cases the reference provides the RAI and the response. If other documents were used in the review of a particular response, then each section provides the appropriate references and citations.

B.1 RAI 4.2-12

The staff requested that GEH describe those factors in the linear heat generation rate uncertainty analysis that account for the uncertainties based on the GT CMS. The staff found the response to RAI 4.2-12 (Ref. 4.2-12.1) to be insufficient. Based on the response to RAI 4.2-12, the staff requested additional information in RAI 4.2-12S1.

B.1.1 RAI 4.2-12S1

The response to RAI 4.2-12S1 is provided in Reference 4.2-12.2. The response individually addresses the 26 parts of RAI 4.2-12S1.

B.1.1.1 RAI 4.2-12S1-1

In RAI 4.2-12S1-1 the staff requested additional information regarding the R-factor uncertainty. The response stated that the applicant would provide this information under RAI 4.3-2S1.

B.1.1.2 RAI 4.2-12S1-2

In RAI 4.2-12S1-2 the staff requested additional information regarding the bundle power calculational uncertainty. The response states that this information will be provided under RAI 4.3-2S1.

B.1.1.3 RAI 4.2-12S1-3

RAI 4.2-12S1-3 pertains to the [[ ]]. GEH and the staff agree that a value of [[ ]] is appropriate because it is consistent with NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains."

B.1.1.4 RAI 4.2-12S1-4

In RAI 4.2-12S1-4 the staff requested additional information regarding the barium calculation performed as part of a report used by GEH from the Kashiwazaki-Kariwa Unit 5 (K5) gamma scan campaign. The response clarifies that the PANAC11 bundle isotopic model was not used for this purpose. GEH calculated the concentrations using a separate methodology, integrating the power distributions over the last several months of operation. This approach is an industry standard practice. The calculation is based on a simple and straightforward relationship between integrated power and concentration and is acceptable to the staff for the purpose of predicting the barium concentration for gamma scan validation purposes.

B.1.1.5 RAI 4.2-12S1-5

In RAI 4.2-12S1-5 the staff requested additional clarification of the barium calculation. In its response, GEH stated that it determined [[ ]]. FLN-2005-034 (Ref. 4.2-12.9) describes the methodology. As stated above, the calculation is simple and straightforward and this process is acceptable to the staff.

B.1.1.6 RAI 4.2-12S1-6

In RAI 4.2-12S1-6 the staff requested additional information regarding the K5 scanned fuel discharge exposure. The response states that the discharge exposure was not part of the report GEH used. Toshiba performed the gamma scans on non-GEH/Global Nuclear Fuel (GNF) fuel and did not provide additional details regarding the bundle exposure.

The staff requested additional information in RAI 4.2-12S2-6.

B.1.1.7 RAI 4.2-12S1-7

In RAI 4.2-12S1-7 the staff requested additional information regarding the standard deviation in the nodal power provided in Table 7A-4 of the design control document (DCD). [[ ]]

The staff determined this acceptable [[ ]].

B.1.1.8 RAI 4.2-12S1-8

In RAI 4.2-12S1-8 the staff requested [[ ]]. The staff again requested additional information in RAI 4.2-12S2-8.

B.1.1.9 RAI 4.2-12S1-9

In RAI 4.2-12S1-9 the staff requested additional information regarding the K5 core loading before the gamma scan campaign. The response states that the fuel is 8X8 and 9X9 fuel. The fuel gadolinia loading ranged between 3.5 weight percent (w/o) and 5.5 w/o. The 9X9 fuel

included part-length rods. The 9X9 fuel also includes diagonal gadolinia-bearing rods that are face-adjacent to vanished rods.

The staff has requested additional information prompting the evaluation of the uncertainties based on a combination of historical data, analytical studies, and the qualification data from all of the tests. Therefore, this information is not required for the staff to determine directly the applicability of the K5 gamma scan uncertainties.

B.1.1.10 RAI 4.2-12S1-10

In RAI 4.2-12S1-10 the staff requested analytical qualification of the GT CMS methodology based on simulated results and power shapes from Plant E [[ ]] of the qualification database for Cycle 9 or 10. The response to RAI 4.2-12S1-10 states that the applicant was not performing this calculation. The staff requested additional information in RAI 4.2-12S2-10.

B.1.1.11 RAI 4.2-12S1-11

In RAI 4.2-12S1-11 the staff requested that the uncertainty analysis considers the trends in uncertainty with power-to-flow ratio. The response states that no commitment is made in regard to this sensitivity as it may apply to the ESBWR. The staff requested additional information in RAI-4.2-12S2-11.

B.1.1.12 RAI 4.2-12S1-12

In RAI 4.2-12S1-12 the staff requested the detailed power/flow operating history for K5 before the gamma scan. The response states that these data were not available to GEH. The staff requested additional information in RAI 4.2-12S2-12.

B.1.1.13 RAI 4.2-12S1-13

In RAI 4.2-12S1-13 the staff requested details regarding the K5 GT CMS adaption technique. The response states that these details were not available to GEH.

The staff requested additional information prompting the evaluation of the uncertainties based on a combination of historical data, analytical studies, and the qualification data from all of the tests. Therefore, this information is not required for the staff to determine the applicability of the K5 gamma scan uncertainties directly.

B.1.1.14 RAI 4.2-12S1-14

In RAI 4.2-12S1-14 the staff requested the verification of the K5 gamma scan measurements compared to PANAC11/TGBLA06 methods. The response states that these codes in the K5 qualification were not part of the report provided to the applicant.

The staff has requested additional information prompting the evaluation of the uncertainties based on a combination of historical data, analytical studies, and the qualification data from all of the tests. Therefore, this information is not required for the staff to determine the applicability of the K5 gamma scan uncertainties directly.

B.1.1.15 RAI 4.2-12S1-15

In RAI 4.2-12S1-15 the staff requested that Figure 7.2-8 in the DCD be revised to indicate the axial elevation of the GT instruments. The response states that Figure 7.2-8 in the next revision of the DCD will indicate the axial elevation of the GT instruments.

B.1.1.16 RAI 4.2-12S1-16

In RAI 4.2-12S1-16 the staff requested that GEH verify that the power shape adaption is performed based on the GT signals and not the local power range monitor (LPRM) signals. The response states that the final decision has not been made, and the RAI cannot be resolved. The staff requested additional information in RAI 4.2-12S2-16.

B.1.1.17 RAI 4.2-12S1-17

In RAI 4.2-12S1-17 the staff requested that GEH update the DCD with the specific adaption technique and the specific means for LPRM calibration. The response states that these techniques are under development and cannot be incorporated. The staff requested additional information in RAI 4.2 12S2-17.

B.1.1.18 RAI 4.2-12S1-18

In RAI 4.2-12S1-18 the staff requested justification of the [[ ]] uncertainty adder for a seven-GT arrangement relative to a nine-GT arrangement. The response states that this adder is determined based on an analysis of the Tokai 2 data. The nodal power uncertainty increased from [[ ]] to [[ ]] when seven GT were considered relative to the nine GT used in the test. The statistical combination was used to determine a factor of [[ ]] to account for the increased nodal uncertainty.

The staff has requested additional information prompting the evaluation of the uncertainties based on a combination of historical data, analytical studies, and the qualification data from all of the tests. Therefore, this information is not required for the staff to determine directly the applicability of the K5 gamma scan uncertainties.

B.1.1.19 RAI 4.2-12S1-19

In RAI 4.2-12S1-19 the staff requested information regarding the sufficiency of the number of instruments. Particularly, the staff asked that GEH verify that the number of GT is sufficient to perform adaption with extrapolation. The response states that the final technique is under development, and the question cannot be resolved. The staff requested additional information in RAI 4.2 12S2-19.

B.1.1.20 RAI 4.2-12S1-20

In RAI 4.2-12S1-20 the staff requested additional information regarding the introduction of higher uncertainties for scenarios in which the axial power shape has several local axial peaks (i.e., double-humped power shapes). The response states that the adaption technique is under development, and the question cannot be resolved. The staff requested additional information in RAI 4.2-12S2-20.

B.1.1.21 RAI 4.2-12S1-21

In RAI 4.2-12S1-21 the staff requested additional information regarding any update to the PCGEN methodology to enable its use in a GT-based CMS. The response states that no modifications have been made. The response to RAI 4.2-12S2-22 provides the updates that have been made to PCGEN. Therefore, the response to RAI 4.2-12S2-22 supersedes the response to RAI 4.2-12S2-21.

B.1.1.22 RAI 4.2-12S1-22

In RAI 4.2-12S1-22 the staff requested additional information regarding the use of [[ ]] to calculate J-factor parameters. The response states that (1) [[ ]] calculations have been performed, (2) 3D MONICORE will be updated, and (3) corroborative Monte Carlo N Particle Transport Code (MCNP) calculations will be performed. The staff requested additional information in RAI 4.2-12S2-22.

B.1.1.23 RAI 4.2-12S1-23

In RAI 4.2-12S1-23 the staff requested additional information regarding the calibration accuracy. The response states that the data acquisition system and calibration frequency are components of the site calibration and that the uncertainty is considered part of the overall bundle power model uncertainty.

The response states that this uncertainty is inherently accounted for in the [[ ]] uncertainty value based on the K5 gamma scan results. The response to RAI 7.2-18S2 revises the GT CMS uncertainty analysis and the staff did not consider the K5 bundle power uncertainty in its review of NEDE-33197P.

B.1.1.24 RAI 4.2-12S1-24

In RAI 4.2-12S1-24 the staff requested justification of the update uncertainty of [[ ]]. The response states that the [[ ]] update contribution is in addition to the [[ ]]. The response to RAI 7.2-18S2 justifies the continued applicability of the update uncertainty.

B.1.1.25 RAI 4.2-12S1-25

In RAI 4.2-12S1-25 the staff requested revision of NEDE-33197P, “Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring.” The response states that revision is not required.

B.1.1.26 RAI 4.2-12S1-26

In RAI 4.2-12S1-26 the staff requested revision of NEDC-33242P “GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report” (Ref. 4.2-12.10). The response states that revision is not required. The staff requested additional information in RAI 4.2-12S2 regarding the component uncertainties in the linear heat generation rate (LHGR) uncertainty. Based on an audit conducted regarding the GT CMS, the staff identified additional information needed for it to complete its review. The staff documented this in an audit report open item associated with RAI 7.2-18S2 (Ref. 4.2-12.7). The open item stated that, should the LHGR uncertainty exceed [[ ]], the staff would require additional information regarding the continued applicability

of NEDC-33242P without revision or a revision of NEDC-33242P. Because the response to RAI 7.2-18S2 indicates that the decreased calibration interval maintains the LHGR uncertainty below [[ ]], the staff agrees that revision of NEDC-33242P is not necessary.

B.1.2 RAI 4.2-12S2

In its review of the response to RAI 4.2-12S1, the staff found that several elements of the methodology were under development, and in several cases, sufficient detail could not be provided regarding the qualification of the K5 GT CMS to allow the staff to reach a conclusion regarding the applicability of the gamma scan data as a basis for the uncertainties associated with the ESBWR. Therefore, the staff issued a supplemental request for additional information. The response to RAI 4.2-12S2 is provided in Reference 4.2-12.3.

B.1.2.1 RAIs 4.2-12S2-1 and 4.2-12S2-2

The staff communicated that these RAI items will not be closed until the staff makes a final determination on the acceptability of the response to RAI 4.3-2S1. As described below, the staff determined that that the response to RAI 4.3-2S2 is acceptable. Therefore, the staff has closed the open items associated with RAIs 4.2-12S2-1 and 4.2-12S2-2.

RAI 4.2-12S2-6

In RAI 4.2-12S2-6 the staff requested a comparison of the K5 reactor to the ESBWR. K5 is a 1,100-megawatt-electric (MWe) class BWR/5. K5 is substantially similar to Tokai 2 in core size and thermal power. K5, Tokai 2, and Limerick 2 are all 764-fuel-bundle cores operating between 3,292 and 3,460 megawatt thermal (MWt) with 43 instrument strings. These are large cores relative to the operating fleet of boiling-water reactors (BWRs). The core thermal power density for these plants is on the order of 50 kilowatts per liter (kW/l), which is similar to the ESBWR power density of 54 kW/l.

The K5 gamma scans were performed for 8X8 and 9X9 fuel bundles; the response to RAI 4.2-12S2-6 (Ref. 4.2-12.3) states that the discharge exposure for the 8X8 fuel bundles is 40 gigawatt-days per metric tonne (GWD/mT) (or 36 gigawatt-days per short ton (GWD/ST)), which is largely similar to the ESBWR fuel discharge exposure of 44 GWD/mT (40 GWD/ST). The K5 plant is a high power density with a maximum linear heat generation rate (MLHGR) on the order of 35–40 kilowatt per meter (kW/m) (10.7–12.2 kilowatt per foot (kW/ft)), which is similar to the ESBWR MLHGR. The similarity in average bundle exposure and power density ensures comparable core isotopic inventories between the plants. The staff noted that, while the ESBWR core is larger (1,132 bundles) relative to those considered in the qualification (764 bundles), these plants are among the largest cores in the operating fleet.

The staff considers the K5 gamma scan data to apply to the qualification of a GT-based CMS for the ESBWR considering the core power density, discharge exposure, and large size of the core. The staff did not consider the applicability of the data to demonstrate the capabilities of the PANAC11 core simulator software for ESBWR-specific geometry, however. The uncertainty analysis considers the GT CMS and nuclear methods uncertainty, as described in response to RAI 4.2-12S2-11.

The intent of the qualification data is to enumerate those uncertainties specific to the GT instrumentation. Considering the prototypic application in large reactors at comparable power density and application to measurements of axial power shapes for high exposure bundles, the

staff determined that that the qualification data collected over these series of plants is adequate to determine instrument uncertainties when considered with parallel measurements using gamma scans and traversing in-core probe (TIP) traces. Therefore, the staff determined that this response acceptable.

B.1.2.2 RAI 4.2-12S2-8

In RAI 4.2-12S2-8 the staff requested additional information regarding the interim methods. In its response to RAI 4.2-12S2-8, GEH clarified that the ESBWR uncertainty analysis considers portions of the interim methodology for expanded operating domain reactors. This interim methodology uses information from the historical qualification of the nuclear design methods to TIP measurements. The staff determined that this interim approach acceptable for the ESBWR when specific adjustments are made to the uncertainty analysis to account for the GT-based CMS.

B.1.2.3 RAI 4.2-12S2-10

The staff requested that GEH perform an interpolation study using simulated GT signals to establish the uncertainty impact of monitoring and adaption based on discrete, as opposed to continuous, axial power measurement.

The applicant performed and documented an adaption study for Plant E. The response (Ref. 4.2-12.3) provides the results of the adaption study. The response describes the adaption techniques and GT arrangements considered in the adaption study. This study evaluates several different adaption options. [[ ]] to generate a data array that is the same dimension as the [[ ]] (measured traversing in-core probe) array currently used in PANAC11 to adapt the power shape. The response to RAI 4.2-12S2-10 specifies [[ ]] as the ESBWR adaption technique.

When the adaption technique was revised from [[ ]] to [[ ]], the response to RAI 7.2-18S2 superseded the response to RAI 4.2-12S2-10. [[

]]The response to RAI 7.2-18S2 provides the revised methodology and revised uncertainty analysis.

B.1.2.4 RAI 4.2-12S2-11

The staff requested additional information regarding the uncertainty analysis to determine the acceptability of the design to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded.

The response to RAI 7.2-18S2 supersedes the response to RAI 4.2-12S2-11. The response to RAI 7.2-18S2 addresses all of the component uncertainties and includes the uncertainty analysis for the critical power ratio (CPR) as well as the LHGR limits. The staff determined that the modified methodology and the associated uncertainty analyses are appropriate and acceptable.

B.1.2.5 RAI 4.2-12S2-12

In RAI 4.2-12S2-12 the staff requested that GEH evaluate the qualification plant tests against the conditions of the ESBWR. GEH provided the reactor thermal power and power-to-flow ratio

window for Limerick 2 and Tokai 2 in response to RAI 4.2-12S2-12. In each case, the power-to-flow ratios were substantially smaller than the ESBWR range ([23.5, 42.5][32] megawatt-thermal per million pounds mass per hour (MWt/Mlbm/h) for Limerick, [24.1, 36.1] MWt/Mlbm/h for Tokai, and [53.6, 62.8] MWt/Mlbm/h for ESBWR).

While the ESBWR power-to-flow ratio is much greater than those plants considered in the test, the plants considered are large plants (764 bundles) with a large number of instrument strings (43). The intent of the qualification data presented in NEDE-33197P is to qualify the use of a GT-based CMS; it is not to independently develop the uncertainty parameters for ESBWR uncertainty analysis or operating limit determination. The staff determined that the conditions of these tests, in core power level, core size, and lattice, are similar enough to ESBWR conditions that these tests may provide the qualification of the instrumentation system to be used in parallel with historical data and interim methods to develop the ESBWR-specific bundle power and LHGR uncertainties.

B.1.2.6 RAI 4.2-12S2-16

In RAI 4.2-12S2-16 the staff requested that GEH perform an uncertainty analysis for each adaptive technique. This was performed as part of the adaption study. The response states that the applicant used the [[ ]] to evaluate the Tokai 2 test results; however, in the application of the GT CMS for ESBWR, GEH developed specific J-factors to account for the gamma field measurement capability of the GT instruments. The staff agrees that the comparison of the gamma-sensitive measurements to the neutron-sensitive instruments requires some assessment of the difference in the measured field (neutron or gamma). The response states that the GT are gamma-sensitive instruments and that modern J-factors are being developed specifically for the ESBWR and GT geometries. The staff reviewed the J-factors provided in the response to RAI 4.2-12S2-22. The staff agrees that further consideration of the [[ ]] is not necessary to develop the ESBWR-specific bundle power and LHGR uncertainties, but may be used as a tool for the qualification of the instruments during in-plant testing.

B.1.2.7 RAI 4.2-12S2-17

In RAI 4.2-12S2-17 the staff requested that GEH revise the licensing topical report (LTR) to include the adaption technique summary and the results of the uncertainty analyses. The response states that GEH will revise the LTR accordingly to close this open item (Ref. 4.2-12.3). Responses to RAIs 4.2-12S2-10 and 4.2-12S2-11 describe the adaption methodology and evaluate the analytic uncertainty associated with the technique. The response to RAI 7.2-18S2 supersedes the response to RAIs 4.2-12S2-10 and 4.2-12S2-11. However, the LTR revision provided in the response to RAI 7.2-18S2 is sufficient to address the specification of the adaption technique and the determination of the uncertainties.

B.1.2.8 RAI 4.2-12S2-19

RAI 4.2-12S2-19 stipulates that, if a single adaption technique is selected, the information requested in RAIs 4.2-12S2-16 and 4.2-12S2-17 be provided for only that technique. The response states that the adaptive technique specifies [[ ]], the response to RAI 4.2-12S2-11 provides the uncertainty analysis, and the applicant will revise the LTR

[32] Bracketed values here denote the minimum and maximum power to flow ratios allowed at the licensed thermal power for the flow control window.

accordingly, as described in response to RAI 4.2-12S2-16. The staff determined that this information is sufficient to close the open item associated with RAI 4.2-12S2-16.

The response to RAI 7.2-18S2 supersedes the response to RAI 4.2-12S2-19. The adaption methodology is based on [[ ]] and the corresponding uncertainties. The response to the RAI and the revised LTR provide these uncertainties. The response to RAI 7.2-18S2 is sufficient to address RAIs 4.2-12S2-16, 4.2-12S2-17, and 5.2-12S2-19 because it provides the information sought by these RAIs.

B.1.2.9 RAI 4.2-12S2-20

The staff requested additional information regarding the axial power shape uncertainty when adaption is performed using discrete, as opposed to continuous, axial power measurements. The staff noted that the nodal power uncertainties were shown to increase with decreasing axial measurements based on the Tokai 2 qualification.

The response to RAI 7.2-18S2 provides the revised interpolation and technique and the associated uncertainty analyses. The response includes consideration of the failure of GT sensors. The response refers to the response to RAI 7.2-66, which specifies the minimum acceptable instrumentation configuration. The response to RAI 7.2-18S2 also provides the details of the quantification of the [[ ]] that are consistent with the minimum set of instrumentation listed in RAI 7.2-66. The staff determined that the response to RAI 7.2-18S2 is adequate to resolve the staff's concerns expressed in RAI 4.2-12S2-20. The response further states that the next revision of the LTR will include the relevant information. The staff determined this is acceptable.

B.1.2.10 RAI 4.2-12S2-22

The staff requested that GEH provide the results of the [[ ]] and MCNP calculations for the J-factor and associated PCGEN inputs. The staff also requested that GEH provide at least one case that considered the effect of the fuel spacer.

The response to RAI 4.2-12S2-22 provides the results of MCNP calculations of the gamma transport factors used in the J-factor methodology. The response provides qualification of the J-factor methodology in PCGEN against MCNP using revised constants in the model. The results indicate agreement within [[ ]]. The staff concluded this acceptable. The response to RAI 4.2-12S2-22 supersedes the response to RAI 4.2-12S1-22.

The response provides the revised PCGEN parameters and their qualification against MCNP. The staff determined that the use of MCNP for this purpose is acceptable. The staff further noted that the response did not include the spacer effect. The applicant explained that the response to RAI 7.2-20S1-B considered the spacer effect. The staff has reviewed this response, as documented below, and concluded that the revised methodology adequately captures the spacer effect. Therefore, the staff determined that the response is acceptable.

B.1.2.11 RAI 4.2-12S2-25

The staff requested that NEDE-33197P be revised. As discussed in RAIs 4.2-12S2-10, 4.2-12S2-11, 4.2-12S2-16, 4.2-12S2-17, 4.2-12S2-19, 4.2-12S2-22, 4.2-12S2-25, and 7.2-18S2, the applicant revised the LTR to account for the finalized methodology and the associated uncertainty analysis.

References

4.2-12.1 MFN 06-492, General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application DCD Chapter 4 and GNF Topical Reports—RAI Numbers 4.2-11 and 4.2-12," December 1, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML063460397, ML063460395).

4.2-12.2 MFN 06-492, Supplement 1, General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 S01," June 20, 2007 (ADAMS Accession Nos. ML071930094, ML071930093).

4.2-12.3 MFN 08-293, General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 106 Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 S02 and 4.3-2 S02," April 3, 2008 (ADAMS Accession Nos. ML080990616, ML080990615).

4.2-12.4 MFN 08-293, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 106 Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 Supplement 2 and 4.3-2 Supplement 2," July 3, 2008 (ADAMS Accession No. ML081930310).

4.2-12.5 MFN-07-544, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Number 7.2-18 Supplement 2," August 18, 2008 (ADAMS Accession No. ML082350337).

4.2-12.6 NEDE-33197P, Rev. 2, "Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring," General Electric Hitachi Nuclear Energy America, LLC, August 2008 (ADAMS Accession Nos. ML082460975, ML082460976, ML082460974, ML082460977).

4.2-12.7 U.S. Nuclear Regulatory Commission, Audit Results Summary Report, "Gamma Thermometers for the ESBWR," August 2008 (ADAMS Accession No. ML082810409).

4.2-12.8 U.S. Nuclear Regulatory Commission, Audit Summary, "Final Audit Summary Including Phase 4 for the ESBWR Gamma Thermometer July 2008," November 5, 2008 (ADAMS Accession No. ML082940529).

4.2-12.9 FLN-2005-034, A. A. Lingenfelter to NRC, "Recent Experimental Thermal Hydraulics and GNF2 Licensing Meeting, October 26-27, 2005," December 15, 2005 (ADAMS Accession No. ML060050548).

4.2-12.10 NEDC-33242P, "GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report," Global Nuclear fuel, January 2006 (ADAMS Accession Nos. ML060370033, ML060370030, ML060370032).

B.2 RAI 4.3-1

The staff requested that GEH provide the results of TGBLA06 calculations to determine the void reactivity coefficient. In a supplemental request for information, the staff asked that GEH describe the process by which it used TGBLA06 lattice calculations to determine the void reactivity coefficient biases and uncertainties used in TRACG04.

The original response provided the detailed lattice physics calculational results for the GE14E fuel lattices used in the ESBWR equilibrium core design (Ref. 4.3-1.1). The results were compared against independent calculations performed by the staff using MCNP/MONTEBURNS. A contractor report details the results of the BNL calculations and comparisons (Ref. 4.3-1.3). The staff reviewed these calculations and the contractor report and found that the TGBLA06 calculations indicate good agreement with more sophisticated transport methods. The staff therefore determined that the lattice physics code acceptably models the lattice design, including the N-lattice fuel bundle arrangement.

In the response to the supplemental request for information, GEH provided a detailed description of the means for calculating the void reactivity coefficient biases and uncertainties using MCNP (Ref. 4.3-1.2). Although the process outlined is unacceptable because it (1) relies on only one void history (40 percent) and (2) does not account for modern fuel lattice designs referenced in the ESBWR core design, the response to RAI 21.6-111 supersedes the response to RAI 4.3-1S1.

References

4.3-1.1 MFN-06-291, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Letter No. 21 Related to ESBWR Design Certification Application—Nuclear Design—RAI Number 4.3-1," August 22, 2006 (ADAMS Accession Nos. ML062480414, ML062480406).

4.3-1.2 MFN-06-291, Supplement 1, "Response to Portion of NRC Request for Additional Information Letter No. 21 Related to ESBWR Design Certification Application—DCD Chapter 4 and GNF Topical Reports—RAI Number 4.3-1," November 13, 2006 (ADAMS Accession Nos. ML063390082, ML063390081).

4.3-1.3 Technical Evaluation Report (TER) for the MCNP Code Package Validation of the Analysis Presented in Chapter 4.3 of the ESBWR Design Application, Brookhaven National Laboratory (ADAMS Accession No. ML092660173)

B.3 RAI 4.3-2

B.3.1 RAI 4.3-2(a)

The staff requested that GEH demonstrate quantitatively and qualitatively that the current lattice and simulator code suite have been validated in regions characteristic of ESBWR operation, such as low mass flow rate and high void fractions. The response provides a comparison of void fraction and flow rates for the ESBWR to the updated experience database (Ref. 4.3-2.1).

B.3.2 RAI 4.3-2(b)

The staff requested that GEH demonstrate quantitatively and qualitatively that the lattice code and associated uncertainties and biases established remain valid for the neutronic and thermal hydraulic conditions predicted for ESBWR operation.

The response states that the coefficients and biases remain valid for ESBWR operation (Ref. 4.3-2.1). The staff did not find the method for quantifying the biases and uncertainties for downstream transient analyses to be acceptable. Therefore, the staff requested in RAI 21.6-111 that GEH update the void reactivity coefficient bias and uncertainty model to (1) incorporate void history and (2) incorporate modern 10X10 lattices indicative of the ESBWR fuel design. The response to RAI 21.6-111 is sufficient to resolve this concern. Therefore, while the staff does not agree that the previous methodology for establishing the void reactivity coefficient and biases is acceptable for ESBWR operation at high power and low flow, the staff determined that the response to RAI 21.6-111 adequately resolves the technical concern. Therefore, the staff has closed RAI 4.3-2(b).

B.3.3 RAI 4.3-2(c)

The staff requested additional information regarding the lattice physics code capabilities to model isotopic depletion for prolonged hard spectrum exposure under high void conditions. The response states that the similarities between the ESBWR and the updated experience database in terms of channel void fractions indicate sufficient similarity in the neutron spectra that conclusions drawn regarding the code applicability to the updated experience database are equally applicable to the ESBWR (Ref. 4.3-2.1). The staff reviewed the updated experience database and found that the database provides sufficient justification of the capability of the method. The staff agrees that the neutron spectra are sufficiently similar to support the conclusions drawn in the RAI response. Therefore, the staff determined that the response provided adequate justification and is acceptable.

B.3.4 RAI 4.3-2(d)

The staff requested validation data of the GEH neutronic methodology predictions by comparison to gamma scan data and TIP data. The staff also requested core follow benchmarking on present fuel design, operating strategies, and core conditions similar to those strategies and core conditions expected for the ESBWR.

This request pertains to any recent fuel (e.g., GE14), particularly for first- and second-cycle operation.

The response states that the updated experience database included in the LTR is based on BWR core follow data at extended power uprate (EPU) operation with GE14 fuel (Ref. 4.3-2.1). The response does not include any data specific to first- and second-cycle operation. The staff has deferred the review of the first-cycle operation to the review of the IC LTR.

B.3.5 RAI 4.3-2S1

Part (a) of the response states that the in-channel void fraction expected for the ESBWR is similar to the in-channel void fractions of those plants included in NEDC-33239P, “GE14 for ESBWR Nuclear Design Report,” as an update to the experience database. Those in-channel

void fraction ranges, depicted for the ESBWR, are within the same range as those experienced for high power density fuels in EPU plants, labeled A through E in NEDC-33239P.

These plants form the experience database for validation of the lattice depletion and core simulator codes, as applied to the ESBWR. Part (d) of the response indicates that core follow data from plants A through E are applicable to the expected conditions for the ESBWR core and fuel design. Part (b) of the response indicates that the associated biases and uncertainties remain valid for the ESBWR.

The uncertainty analyses applied in NEDC-33237P is based on NRC-approved methodologies in NEDC-32694P-A. The staff does not find this methodology acceptable for application to EPU plants or plants with normal conditions of operation similar to currently operating BWRs with expanded operating domains. Therefore, the staff does not find that the response adequately justifies the current uncertainty analyses based on the database referenced.

In accordance with the conditions of NEDC-32601P-A, the following actions must be taken to apply the approved methodology for power distribution uncertainties to determine the safety limit minimum critical power ratio (SLMCPR):

- Verify the TGBLA fuel rod power calculational uncertainty when applied to new fuel designs.
- Reevaluate the effect of the correlation of rod power calculation uncertainties to ensure the accuracy of the R-factor uncertainty when the methodology is applied to a new fuel lattice.
- Verify the 3D MONICORE bundle power calculational uncertainty when applied to new fuel and core designs.

The uncertainty analysis in NEDC-33237P references a power peaking uncertainty of [[ ]] (NEDC-33239P). This value is inconsistent with the value of [[ ]] referenced in NEDC-33173P, based on the [[ ]].

Therefore, the staff issued a supplemental request for additional information (RAI 4.3-2S1) which requested that GEH revise the uncertainty analysis.

B.3.5.1 RAI 4.3-2S1-1

The staff requested that GEH explain the inconsistency and provide the value for local pin peaking factor uncertainty based on the MCNP and TGBLA06 calculations provided in NEDC-33239P using the [[ ]], as described in Section 2.2.1.2 of NEDC-33173P (Ref. 4.3-2.7). The response states that the generic value of [[ ]] bounds the ESBWR-specific value of [[ ]] (Ref. 4.3-2.2). However, the response states that if the [[ ]] were to be adopted for the ESBWR-specific lattices, the resultant uncertainty is [[ ]]. The staff determined that a value of [[ ]] is acceptable for use in the uncertainty analysis because it bounds all of the ESBWR lattices.

B.3.5.2 RAI 4.3-2S1-2

FLN 2001-017, dated October 1, 2001, details the applicability of the R-factor methodology in NEDC-32505P-A to GE14 fuel lattices (Refs. 4.3-2.5 and 4.3-2.6). The staff requested that GEH explain the applicability of the methodology for the same lattice with reduced flow conditions relative to currently operating BWRs with GE14 fuel. The staff requested that GEH evaluate the R-factor uncertainty based on the local (pin) power peaking uncertainty calculated based on the [[ ]].

The response states that consideration of higher void conditions in the local pin power peaking uncertainty based on MCNP calculations using the [[ ]] results in a reduction in the lattice peaking uncertainty from [[ ]] to [[ ]] (Ref. 4.3-2.2). Since the TGBLA06 uncertainties are reduced for higher in-channel void conditions, the staff determined that the use of the value based on the standard production lattice in-channel void fractions confers a small degree of conservatism given the large ESBWR core average void fraction. This is acceptable to justify the use of the pin peaking uncertainty for downstream R-factor uncertainty evaluations at high power-to-flow ratios.

The response further calculates the total random uncertainty based on the [[ ]] based pin power peaking uncertainty yielding [[ ]] (Ref. 4.3-2.2). The [[ ]] value was compared with the [[ ]] value used to evaluate the interim methods R-factor uncertainty. All other parameters being equal ([[ ]]), the staff determined that the R-factor uncertainty calculated using the higher lattice uncertainty of [[ ]] is less than [[ ]]. The ESBWR generic R-factor uncertainty of [[ ]] is therefore conservative and acceptable for use in the operating limit minimum critical power ratio (OLMCPR) determination.

B.3.5.3 RAI 4.3-2S1-3

The staff requested that GEH justify the applicability of a bundle power distribution uncertainty. The bundle power calculational uncertainty in NEDC-33237P is based strictly on the value quoted in NEDC-32601P-A (Refs. 4.3-2.8 and 4.3-2.9). The lower calculated bundle uncertainties from NEDE-33197P justify this uncertainty. Table 9-14 in NEDE-33179P cites a bundle power uncertainty of [[ ]] for the GT configuration proposed for the ESBWR core (Ref. 4.3-2.12).

The response provides an average of the Tokai 2 and K5 estimated total uncertainty per GT string. The resultant uncertainty is lower than the [[ ]] value provided in NEDC-32601P-A. The response states that GEH intends to maintain the historical parameter based on a larger dataset and generic application. The staff finds that the historical parameter has been technically justified based on a larger dataset and the staff has concluded that this historical parameter is acceptable.

B.3.5.4 RAI 4.3-2S1-4

The staff requested that GEH describe the component uncertainties in the [[ ]] bundle power uncertainty provided in NEDE-33197P (Ref. 4.3-2.12). The response states that the bundle power uncertainty is a combination of the [[ ]], the [[

]], the [[ ]], and the [[ ]] for Tokai 2 (Ref. 4.3-2.2). For K5, the gamma scan results are used to determine the bundle powers for uncertainty determination. Doing so explicitly accounts for the [[ ]]. However, based on the response to RAI 7.2-18S2, these data are not used in determining the nuclear methodology power distribution uncertainties. Therefore, the staff did not rely on this information in the conduct of its review.

B.3.5.5 RAI 4.3-2S1-5

The staff requested that GEH describe the determination of the [[ ]] uncertainty. The response states that the uncertainty is based on the differences in the measured GT and n TIP readings above [[ ]] thermal power at Tokai 2 (Ref. 4.3-2.2). The staff determined that this explanation is sufficient for the staff to understand the basis for this uncertainty.

B.3.5.6 RAI 4.3-2S1-6

The staff requested additional justification of the bundle [[ ]]. As discussed above, the staff determined that the ESBWR operating conditions are similar to EPU operating conditions. The staff requested that GEH specifically provide an analysis showing the bundle power calculational uncertainty applying the [[ ]] for the bundle [[ ]]. The value of the bundle [[ ]] in NEDC-33173P (Ref. 4.3-2.7) is inconsistent with the value of [[ ]] shown in Table 9-14 of NEDE-33197P (Ref. 4.3-2.12).

The response states that utilizing the [[ ]] increases the bundle power uncertainty based on the Tokai 2 test data to [[ ]], although these gamma scan data do not factor into the K5 test data (Ref. 4.3-2.2). The staff agrees that the [[ ]] does not affect the K5 bundle power uncertainty because the gamma scans allow specific determination of the bundle powers for direct comparison.

The response combines the K5 and Tokai 2 data and provides an average bundle power uncertainty that is less than the [[ ]] value included in NEDC-32601P-A (Ref. 4.3-2.8).

B.3.5.7 RAI 4.3-2S1-7

The staff requested that GEH update the OLMCPR. The response states that an update is not required (Ref. 4.3-2.2). This RAI and its response were rendered irrelevant by subsequent changes in the uncertainty analysis as described in the response to RAI 7.2-18S2. Therefore, the staff did not consider the information provided in this response in its review.

B.3.5.8 RAI 4.3-2S1-8

The staff requested that GEH update the MLHGR. The response states that the [[ ]] uncertainty assumed in the MLHGR limit remains conservative (Ref. 4.3-2.2). This RAI and its response were rendered irrelevant by subsequent changes in the uncertainty analysis as described in the response to RAI 7.2-18S2. Therefore, the staff did not consider the information provided in this response in its review.

B.3.6 RAI 4.3-2S2

The staff found that the basis for the bundle power distribution uncertainty was not well founded. In particular, the staff found insufficient data to justify the applicability of the K5 gamma scan to ESBWR operating conditions.

Further, the qualification data are predicated on an increased number of GT instruments relative to the ESBWR design (nine instruments per string as opposed to seven instruments per string for the ESBWR).

The staff requested additional information in RAI 4.3-2S2 to justify the bundle power distribution uncertainty. This RAI addressed concerns regarding instrumentation performance, core operating conditions, and the data used in quantifying the uncertainty.

B.3.6.1. RAI 4.3-2S2-A

The staff requested that GEH confirm that the [[ ]] local pin power peaking uncertainty bounds the ESBWR equilibrium and initial core lattices. The response verifies that this value is bounding and therefore acceptable.

The staff's approval in this regard is subject to a condition on the LTR that other fuel designs may be used only if the infinite lattice pin power peaking uncertainty to be reassessed using the approach described in the response to RAI 4.3-2S2-A or a subsequently approved method as described in the most recently reviewed and approved revision of or supplement to NEDC-33173P. This condition is captured in the response to RAI 7.2-71.

B.3.6.2 RAI 4.3-2S2-B

The response to RAI 4.3-2S1 indicates that GEH performed a SLMCPR analysis for the ESBWR. The response to RAI 4.3-2S2-B clarifies that this analysis differs from the approach used in the operating fleet and indicates that the response to RAI 15.0-16S1 describes this methodology. The staff requested the analysis to understand the uncertainty components factored in the uncertainty analysis. The staff determined that the information provided in the response to RAI 4.3-2S2-D and NEDE-33197P, Revision 2, is sufficient to explain the uncertainty components in the bundle power distribution uncertainty.

B.3.6.3 RAI 4.3-2S2-C-1

RAI 4.3-2S2-C-1 requested that GEH address the increased ESBWR power-to-flow ratio relative to operating reactors in the assessment of the power distribution uncertainties. The response specifies that the ESBWR core inlet enthalpy is substantially smaller than operating reactor core inlet enthalpy and compares the ESBWR value with a BWR/6 and the advanced boiling-water reactor (ABWR). The lower core inlet enthalpy (and hence higher inlet subcooling), and in conjunction with the shorter core height, results in similar core void fractions. The staff agrees that these features of the ESBWR design will result in lower core average void fractions than suggested by the trends presented in the updated experience database documented in MFN-05-029.

The ESBWR core average void fraction, when considered with the N-lattice, results in a neutron spectrum that is similar when compared to the ABWR and potentially softer relative to EPU conditions for a BWR/6 plant.

Based on the response, the staff concludes that, while the power-to-flow ratio for the ESBWR exceeds the extremes of the updated experience database, this power-to-flow ratio does not indicate extrapolation in terms of spectral conditions relative to EPU conditions. The staff therefore determined that interim methods are sufficient to resolve concerns regarding potentially increased bundle and pin uncertainties at the power-to-flow conditions for the ESBWR.

The response to RAI 4.3-2S2-C-1 refers to the response to RAIs 4.2-12S2-10 and 4.2-12S2-11. GEH described the adaptive process in these RAI responses. The applicant performed specific qualification against Plant E in the updated experience database and against a BWR/5 where recent GT qualification data were collected. The response to RAI 7.2-18S2 updates the information provided in the response to RAIs 4.2-12S2-10 and 4.2-12S2-11. Nonetheless, the [[ ]] is no longer used as a basis in the uncertainty analysis and is instead used as a quantity to interpret the GT qualification data collected at Limerick, Tokai 2, and K5. The staff determined that the parameter is useful for this purpose and that the revised basis for the power distribution uncertainties is acceptable.

The staff has reviewed the power distribution uncertainties and concluded that RAI 7.2-18S2 provides an adequate basis for the GT CMS instrumentation-specific uncertainties and that the use of the [[ ]] is acceptable.

B.3.6.4 RAI 4.3-2S2-C-2

RAI 4.3-2S2-C-2 requested an evaluation of the GT performance and its sensitivity to bypass conditions, such as voiding. The applicant provided the results of the Multi-Use Safety Environmental Facility test and demonstrated that for high levels of voiding the GT sensitivity remains unchanged. This is likely because of the constant temperature and consistency in the heat transfer coefficient for a wide range in the nucleate boiling regime. The staff considered the possibility of low bypass temperature impacting the instrument sensitivity; however, the staff noted that the GT instruments are calibrated at steady-state conditions before LPRM calibration in accordance with the response to RAI 7.2-59S2. Therefore, the calibration process captures the sensitivity changes caused by potentially colder bypass conditions during feedwater temperature maneuvers. Thus, the staff determined that this response is acceptable based on the additional information provided in the response to RAI 7.2-59S2.

B.3.6.5 RAI 4.3-2S2-D

RAI 4.3-2S2-D requested that GEH explain why the uncertainty determination did not include Limerick 2 data. Based on the audit, the staff identified several aspects of the test experience that differ substantially from the ESBWR design. This includes the [[ ]] approach, the GT string assembly, and the number of GT instruments per string.

The response to RAI 7.2-18S2 clarifies that the ESBWR GT CMS is based on the GT-specific [[ ]] approach as opposed to the single fixed-alpha approach used in the Limerick test.

The response to RAI 4.3-2S2-D states that the applicant will update the uncertainty analysis in the LTR to remove consideration of the comparison of simulated GT measurements to nTIP

measurements and reference the adaption study results. The imposition of interim methods uncertainties (see response to RAI 4.2-12S2-8), when considered with the ESBWR spectral conditions (see response to RAI 4.3-2S2-C-1), allows for the limited application of historically determined uncertainties based on TIP measurements to the ESBWR conditions.

The adaption study and uncertainty analysis presented in response to RAIs 4.2-12S2-10 and 4.2-12S2-11, as updated by the response to RAI 7.2-18S2, form the basis for the ESBWR-specific uncertainty analysis.

The response to RAI 7.2-18S2 provides the details of the specific uncertainties. Limerick 2 and Tokai 2 test data were used to determine the [[ ]] in a bounding sense.

The inclusion of limited Limerick 2 data is conservative considering improvements in GT operating practices following the test that were adopted in Tokai 2 and preserved for the ESBWR application.

Based on the above, the staff determined that this approach is acceptable to address concerns regarding the applicability of the data used in the uncertainty analysis.

B.3.6.6 RAI 4.3-2S2-E

RAI 4.3-2S2-E requested that GEH explain how it weighted the Tokai 2 and K5 uncertainties. The response is sufficient insofar as it explains how the uncertainties were weighted. These uncertainties, however, are not used in the uncertainty determination per the response to RAI 7.2-18S2. Therefore, the response has been rendered irrelevant to the staff's review and this response was not considered.

B.3.6.7 RAI 4.3-2S2-F

RAI 4.3-2S2-F requested that the power distribution uncertainty analysis include a term to address the number of GT instruments per string. The adaptive study provides a quantitative basis for determining the additional uncertainty. GEH used the results of the study to determine this uncertainty parameter and include this uncertainty in the additional bundle uncertainty in the subject analysis and update the LTR. GEH provided the qualification of the adaptive technique to the staff in response to RAI 4.2-12S2-10. The applicant revised the uncertainty analysis to include the [[ ]], as determined by the qualification, and provided it to the staff in response to RAI 4.2-12S2-11. The staff determined that the information provided by the response to RAIs 4.2-12S2-10 and 4.2-12S2-11 is sufficient to address its technical concern. In response to RAI 4.3-2S2-F, GEH stated in the response that it will revise the topical report accordingly. Subsequently GEH submitted the revised LTR and the revision made.

The response to RAI 7.2-18S2 supersedes the responses to RAIs 4.2-12S2-10 and 4.2-12S2-11. The revision, however, is to the adaptive methodology. The response to RAI 7.2-18S2 describes the revised adaption methodology and provides the recalculated [[ ]]. The response to RAI 7.2-18S2 states that the applicant will revise the LTR incorporating the updated methodology and uncertainty analysis. Therefore, the staff determined that the information provided in response to RAI 7.2-18S2 is sufficient to resolve the technical concern associated with RAI 4.3-2S2-F. Subsequently, GEH submitted another revision to the LTR which is consistent with the more recent response to RAI 7.2-18S2.

B.3.6.8 RAI 4.3-2S2-G

The response states that the [[ ]] ESBWR generic R-factor uncertainty is based on the [[ ]] uncertainty reported in NEDC-33239P. The staff did not find that the response addressed its concern that the R-factor uncertainty must be consistent with or conservative relative to the pin power peaking uncertainty. Nonetheless, the staff has reviewed the [[ ]] value and found this value to be conservative relative to the accepted pin power peaking uncertainty of [[ ]] using the [[ ]]. The R-factor uncertainty calculated using either [[ ]] or [[ ]] is less than [[ ]]. The [[ ]] assumed for the ESBWR was selected to be conservative. The staff's review of the combination of the uncertainties confirmed that [[ ]] is conservative.

As discussed in the revised LTR, the R-factor uncertainty is consistent with or conservative relative to the pin power peaking uncertainty determined using either the [[ ]] or an alternative approach described in the most recently reviewed and approved revision or supplement to NEDC-33173P.

References

4.3-2.1 MFN-06-350, General Electric, letter to U.S. Nuclear Regulatory Commission, "Partial Response to RAI Letter No. 53 Related to ESBWR Design Certification Application DCD Chapter 4 and GNF Topical Reports—RAI Number 4.3-2, 4.3-5, 4.4-25, 4.4-30, 4.4-35, 4.4-39, 4.4-51," September 29, 2006 (ADAMS Accession Nos. ML062890048, ML062890047).

4.3-2.2 MFN-06-350 Supplement 3, General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application RAI Numbers 4.3-2 S01 and 4.4-39 S01," June 15, 2007 (ADAMS Accession Nos. ML071930245, ML071930240).

4.3-2.3 MFN-08-293, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Letter No. 106—Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 Supplement 2 and 4.3-2 Supplement 2," April 3, 2008 (ADAMS Accession Nos. ML080990615, ML080990616).

4.3-2.4 MFN-08-293 Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 106—Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 Supplement 2 and 4.3-2 Supplement 2," July 3, 2008 (ADAMS Accession Nos. ML081930310, ML081930311).

4.3-2.5 FLN 2001-017, Watford, G., Global Nuclear Fuel, letter to U.S. Nuclear Regulatory Commission, "Confirmation of the Applicability of the GEXL14 Correlation and Associated R-Factor Methodology for Calculating SLMCPR Values in Cores Containing GE14 Fuel," October 1, 2001 (ADAMS Accession No. ML012830075).

4.3-2.6 NEDC-32505P-A, Rev. 1, General Electric, Licensing Topical Report, "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel," July 31, 1999 (ADAMS Accession Nos. ML060520637, ML060520635, ML060520636).

4.3-2.7 NEDC-33173P, General Electric, Licensing Topical Report, "Applicability of GE Methods to Expanded Operating Domains," February 28, 2006 (ADAMS Accession Nos. ML060720280, ML060720281).

4.3-2.8 NEDC-32601P-A, General Electric, Licensing Topical Report, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," August 31, 1999 (ADAMS Accession Nos. ML003740145, ML003740119).

4.3-2.9 NEDC-33237P, Bentley, T., Bolger, F. and Congdon, S., Global Nuclear Fuel, Licensing Topical Report, "GE14 for ESBWR—Critical Power Correlation, Uncertainty, and OLMCPR Development," March 31, 2006 (ADAMS Accession Nos. ML060750695, ML060750687, ML060750690).

4.3-2.10 NEDC-32694P-A, General Electric, Licensing Topical Report, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," August 31, 1999 (ADAMS Accession Nos. ML003740151, ML003740119).

4.3-2.11 MFN-07-544 Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Number 7.2-18 Supplement 2," August 18, 2008 (ADAMS Accession Nos. ML082350337, ML082350338).

4.3-2.12 NEDE-33197P, Rev. 2, "Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring," General Electric Hitachi Nuclear Energy America, LLC, August 2008 (ADAMS Accession Nos. ML082460975, ML082460976, ML082460974, ML082460977).

B.4 RAI 4.3-3

The staff requested that GEH describe modifications made to the TGBLA06 lattice physics code and provide demonstration analyses of the modified and unmodified TGBLA06 code.

In response to RAI 4.3-3, the applicant demonstrated the performance of the TGBLA06AE4 code against the modified TGBLA06AE5 code. The TGBLA06AE5 code includes a modification to the treatment of a strong low-lying plutonium-240 resonance. In harder spectrum exposures (for high void fractions up to 90 percent), the current energy group structure in the production TGBLA06AE4 code overpredicts the plutonium-240 absorption cross-section. While the modification to TGBLA06AE5 includes improved resolution and modeling techniques for this resonance, it also serves to demonstrate that the extrapolation technique does not appear to significantly affect the TGBLA06AE4 predicted infinite eigenvalue at higher void fractions.

The TGBLA06AE5 modification is the treatment of the low-lying plutonium-240 resonance at 1.058 electron volts (eV) in the epithermal self-shielding model. The enhancement in the treatment of this particular resonance is performed to support qualification of the TGBLA06AE5 lattice physics code to in-channel void fractions on the order of 90 percent. This particular resonance is treated with very fine energy resolution to improve the prediction of lattice

reactivity and isotopic rate of change calculations during depletion. The supplemental RAI response describes in detail the implementation of the correction.

At 90 percent void, TGBLA06 depletion calculations overpredict the plutonium-240 resonance absorption cross-section. As a function of overpredicting the cross-section, the 90 percent void depletion calculations show that the concentration of plutonium-240 is much smaller than expected based on extrapolation from the concentrations predicted from depletions at 0 percent, 40 percent, and 70 percent void fraction. TGBLA06AE5 modified the treatment for the 1.058-eV resonance of plutonium-240. While calculations show only a small influence on the standard production depletion calculations [[ ]] void depletion lattice parameters agree very well with the lattice parameters predicted by extrapolation from the standard production depletions.

TGBLA06AE4 calculates the effective resonance integral in the epithermal energy range based on a two-region cell model. The two-region cell assumes a single-level Breit-Wigner shape in modeling the fuel resonances. The effective absorption resonance integral sums the effects of each resonance. The group-wise absorption cross-sections are then calculated by weighting the resonance integral by the approximate fraction of absorptions that occur in the group and the group flux. Chernick's equations are used to calculate the fuel and moderator fluxes based on the background cross-section and the resonance escape probability.

For cases in which resonant self-shielding is highly pronounced (e.g., the low-lying plutonium-240 absorption resonance), an improved intermediate resonance model was implemented with a fine energy integral calculation module. The refined resonance treatment calculates the plutonium-240 fission and absorption cross-sections based on an improvement in the Chernick approach by increasing the number of energy groups and including a first flight collision probability, three-region correction.

The self-shielding prediction is improved relative to the standard model employed for other resonances by incorporating a first flight collision probability approach in a three region pin-cell (fuel, cladding, and moderator) as opposed to simply solving the Chernick's equations for a two-region cell. The energy resolution is also increased for the plutonium-240 calculation. TGBLA06AE4 resolves the plutonium-240 resonance in one of the 68 epithermal energy groups. TGBLA06AE5 also uses a fine energy group structure ranging between [[ ]]. Between these energies, TGBLA06AE5 calculates the resonance contribution given the smooth background cross-section between 0.66 eV and 3.91 kiloelectron volts and incorporates directly all 200 resonance parameters in the nuclear dataset for plutonium-240.

The staff determined that the approach for the modification is merely to increase the energy and spatial resolution of the TGBLA06 calculation of the homogenized pin parameters. The modification accounts for the more detailed geometry by accounting for the cladding region and increases the number of energy groups evaluated to better capture the resonance shielding effects. Therefore, the staff determined that this modification enhances the accuracy of TGBLA06, and the applicant's RAI response demonstrated this enhancement for high void fraction depletion.

Over the range of application, the staff, however, determined that the use of either version of the code predicts consistent nuclear parameters. Therefore, the use of the TGBLA06AE4 code version to perform the design certification safety analyses is acceptable to the staff.

References

4.3-3.1 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

4.3-3.2 MFN-06-297, Supplement 1, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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 Number 4.3-3," November 8, 2006 (ADAMS Accession Nos. ML06300074, ML063400067).

B.5 RAI 4.3-4

The staff requested that GEH provide a list of code changes to PANACEA (PANAC11) since the staff review and approval of that code. The original response provided a list of all code modifications. The staff performed an onsite audit of these code changes to ensure that these changes did not affect the approved methodology executed by the code.

To resolve concerns regarding code drift, the staff requested in a supplemental request for information that GEH perform a reassessment of the current PANAC11 code against a case in the original qualification provided to the staff in Reference 4.3-4.4. The case selected was Limerick Cycle 5.

The staff reviewed the results in greater detail in Reference 4.3-4.5. The results indicate that the prediction of the nodal power distribution, MLHGR, and minimum CPR are unaffected by the suite of code changes that have been made to the PANACEA code.

Therefore, the staff agrees that the results of the safety analysis using these codes remain essentially the same.

The staff requested additional information regarding the analysis in RAI 4.3-4S2, Items 1 through 7. Specifically, the staff asked GEH to clarify the analysis methods used and the results provided in the response to RAI 4.3-4S1.

In response to Item 1, GEH corrected the water density tables provided in the response to RAI 4.3-4S1.

In response to Items 2 and 3, GEH provided the standard production power density ([[ ]]) and verified that it performed the analysis using the standard production technique.

In response to Items 4 and 5, GEH corrected an error in the calculation of the fission density RMS values provided in the response to RAI 4.3-4S1 and verified that the LTR did not include the same error.

In response to Item 6, GEH provided the results of fission density calculation comparisons to MCNP for controlled and uncontrolled cases. The results indicate that, for the ESBWR GE14E dominant zone lattice, the fission density RMS for all control cases and void history exposure cases [[ ]]. This is consistent with the historically determined accuracy for TGBLA06.

In Item 7, the staff requested that GEH evaluate the impact of the NOLMP[33] option in TGBLA06 to determine the order of magnitude impact of the lumped cross-sections on the fission density comparison with MCNP. The results of comparisons indicate that the magnitude of the lumped fission cross-sections increase with exposure as expected. The magnitude of the impact [[ ]]. Therefore, the staff determined that the basis for comparison of the fission densities is acceptable because the removal of the lumped cross-sections does not introduce a significant perturbation to the pin power distribution relative to the fission density RMS.

References

4.3-4.1 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

4.3-4.2 MFN-06-297, Supplement 2, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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 4.3-4, 4.4-48, 4.8-7," December 21, 2006 (ADAMS Accession Nos. ML070110550, ML070110131).

4.3-4.3 MFN-06-297, Supplement 8, Hinds, D., General Electric, letter to US Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application—RAI Numbers 4.3-4 S02 and 4.4-5 S01," June 21, 2007 (ADAMS Accession Nos. ML071930222, ML071930218).

4.3-4.4 MFN 098-96, Reda, R., General Electric, letter to Jones, R.C., U.S. Nuclear Regulatory Commission, "Implementation of Improved GE Steady-State Nuclear Methods," July 2, 1996 (ADAMS Accession Nos. ML070400507, ML083660032).

4.3-4.5 U.S. Nuclear Regulatory Commission, Audit Results Summary Report, "ESBWR DCD Section 4.3 Nuclear Codes," Addendum 1, January 2007 (ADAMS Accession No. ML082890853).

[33] NOLMP is a computational option in the TGBLA06 code. This option allows TGBLA06 to compute the nuclear parameters having removed all "lumped" fission product and gadolinia tails materials from the model.

4.3-4.6 U.S. Nuclear Regulatory Commission, Audit Summary, "Summary of Audit for Nuclear Design Codes October/November 2006," July 19, 2007 (ADAMS Accession No. ML071700037).

B.6 RAI 4.4-34

The staff requested additional information regarding the core simulator methodology, and, in particular, the coupling of bundles in the calculation. The response states that internodal nuclear coupling is modeled through the epithermal leakage in the 1.5 group PANACEA method. The pin power reconstruction methodology captures the effects of neighboring bundles. Accordingly he staff determined that the response is acceptable in its description of the models.

References

4.3-4.7 4.4-34.1, MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

B.7 RAI 4.4-35

In the response to RAI 4.4-35, the applicant demonstrated the sensitivity of the gadolinium bias to void fraction. GEH presented the results for both lower and higher void depletion analyses. The results indicate that the bias is [[

]]. In a request for supplemental information, the staff requested that the applicant evaluate the influence of the void dependence of the gadolinia bias on reactivity coefficients and subsequent transient calculations.

For steady-state and depletion analyses, the staff determined that these results indicate that the bias will be [[

]]. This was confirmed by direct comparison to MCNP calculations which show that the magnitude of the bias is [[

]].

TRACG also accounts for gadolinia biases. In response to RAI 4.4-35, the applicant explained that gadolinia biases are captured based on an operating experience database.

The staff reviewed the method for capturing the void dependence of the reactivity bias on transient applications in the review of the response to RAI 4.3-1. The staff determined that the methodology for capturing the biases was not acceptable for the conditions of ESBWR operation. GEH revised this methodology and provided the revised method in response to RAI 21.6-111.

The methodology described in the response to RAI 21.6-111 considers 10X10 lattices and incorporates high instantaneous void fractions and void fraction histories.

The void dependence of the gadolinia bias, however, is [[

]]. Therefore, the staff concluded that the method for including this bias through TRACG analysis is acceptable for the ESBWR application.

References

4.4-35.1 MFN-06-350, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," September 29, 2006 (ADAMS Accession Nos. ML062890047, ML062890048).

4.4-35.2 MFN-06-350, Supplement 1, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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 Number 4.4-35 S01 on March 6, 2007 (ADAMS Accession Nos. ML070720688, ML070720690).

4.4-35.3 MFN-08-504, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 147—Related to ESBWR Design Certification Application—RAI Number 21.6-111," June 24, 2008 (ADAMS Accession No. ML081780577).

4.4-35.4 MFN-06-291, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Letter No. 21 Related to ESBWR Design Certification Application—Nuclear Design—RAI Number 4.3-1," August 22, 2006 (ADAMS Accession Nos. ML062480414, ML062480406).

B.8 RAI 4.4-36

The staff requested additional information regarding the comparison of the TGBLA06 lattice physics code against critical experiments. The response states that comparison of the TGBLA06 code against critical experiments is not practical based on the two-dimensional nature of the lattice code.

The staff noted that lattice physics codes have been successfully benchmarked directly against critical experiments based on detailed leakage correction factors to account for the three dimensional effects. This correction is referred to as the experimental buckling. GEH developed such an approach which the staff accepted in its review of NEDE-20913P-A and NEDO-20939A.

However, GEH has adopted an alternative approach which uses the MCNP code as a bridging code between the three-dimensional critical experiment qualification data and TGBLA06. GEH provided the critical benchmark qualification of MCNP against a number of critical experiments. Of these experiments, several are highly relevant to the qualification of a lattice method for BWR applications.

The response references the Toshiba critical assembly testing facility where realistic BWR lattice conditions are modeled using simulated void through hollow inserts. In the GEH process, MCNP is qualified through direct comparison to the critical benchmarks using detailed three-dimensional models. The TGBLA06 qualification is then based on direct comparison against two-dimensional MCNP calculational results. The staff determined that this approach is also acceptable.

References

4.3-4.8 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

B.9 RAI 4.4-37

The staff requested that GEH provide additional information regarding the linearity of thermal hydraulic variables between nodes. The response states that the analysis assumed linear variation so as to maintain continuity between nodes. Because the solution remains continuous and the variables are tracked at the nodal level with dimensions consistent with nuclear coupling, the staff determined that this approach is acceptable.

References

4.3-4.9 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

B.10 RAI 4.4-38

The staff requested that GEH provide additional descriptive details of its methodology in regards to the modeling of crud on the fuel assemblies. The response states that crud is modeled as a uniform thickness layer on all fuel rods. The crud affects the heat transfer and pressure drop calculations for the bundles in the PANACEA code. The effect of the crud, however, is minor when evaluating the steady-state thermal hydraulics and temperature conditions as the heat flux is driven by the total heat deposition and the heat is removed via nucleate boiling.

Therefore, the surface temperature calculation is unaffected except through indicated effects from flow variation.

The crud also reduces the in-channel flow area. GEH provided a sensitivity analysis of the flow variation caused by crud and determined that this flow change is negligible. Therefore, the staff agrees with GEH that while the code takes the crud into account, such modeling has only a negligible impact on the steady-state thermal hydraulic calculations. Since taking crud into account has a negligible effect on the calculations, the staff determined that the response is acceptable.

References

4.3-4.10 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

B.11 RAI 4.4-39

The staff requested that the applicant evaluate the [[ ]] assumption in PANAC11 given the chimney arrangement for the ESBWR. The staff expressed concern in RAI 4.4-39 that the chimney would block thermal hydraulic communication above the core and result in an uneven pressure distribution. To address the staff concern, the applicant provided analyses using TRACG to demonstrate that radial flow in the predominantly liquid bypass equalizes pressure at the core outlet.

The staff questioned the validity of the assumption for the following reasons:

1. The high power density of the ESBWR core will result in bypass voiding resulting from significant direct heating below the top of active fuel.

2. The chimney partitions block thermal hydraulic communication above the top guide between super bundles.

Therefore, the staff requested that the applicant perform an analysis to determine the core outlet pressure distribution using an independent verification approach.

In RAI 4.4-39S1, the staff requested additional information regarding this issue because the TRACG04 calculations [[

]].

Therefore, the staff did not find that the TRACG04 analysis provided an independent calculational assessment of the core outlet pressure resulting from the strong nuclear coupling between bundle fluid conditions and the PANAC11 calculated powers.

In response to RAI 4.4-39S1, GEH asserted that the analyses are independent and provided the results of the TRACG calculated bypass flow patterns, including the radial and axial fluid velocity in the interassembly bypass. The staff, however, concluded that the analyses are

coupled because GEH's assertion that the user inputs the PANAC11 core bypass flow is inaccurate.

The bypass flow is calculated using coupled TRACG04/PANAC11 calculations. The staff issued RAI 4.4-39S2 requesting that GEH modify the TRACG04 initialization to allow for an independent analysis of the core outlet pressure distribution.

In response to RAI 4.4-39S2, GEH developed an approach for independently assessing the bypass pressure distribution (Ref. 4.4-39.3). The response refers to an approach developed for initializing TRACG04 in response to RAI 32 that was issued as part of the staff's review of NEDE-32906P, Supplement 3 (Ref. 4.4-39.4). The analysis is performed by bypassing the [[

]], which include the three-dimensional vessel model and the interfacial shear model to determine the nodal void fraction.

As a result of running TRACG in this manner, the [[

]]. [[

]] This indicates a very small deviation in core reactivity as a result of the inconsistent void models. GEH attributes this small deviation to the fact that the interfacial shear model and the void quality correlation share the same development basis. The staff agrees that the magnitude of the deviation is therefore expected. The staff also agrees that including [[

]] and therefore provides a valid basis for comparison.

The accuracy of the void model and the impact of the PANAC/TRACG initialization on the transient response is the subject of RAI 21.6-111S1. As the subject matter of this RAI is core outlet pressure distribution, the staff did not perform a review of the void fraction axial distribution on the TRACG04 transient response in connection with the response to RAI 4.4-39.

The figures provided in Reference 4.4-39.3 confirm that in the ESBWR, the [[

]]. Because the purpose of the current analysis is to quantify the radial core outlet pressure distribution, the slight variations are of minor importance relative to the subject matter of the RAI.

Figure 4.4-39S02-1 of the response provides the core outlet pressure distribution. The results indicate that [[ ]]. According to Figure 3.6-11 of Reference 4.4-39.5, the vessel cells are numbered from the radial inward cells outward. Therefore, the response indicates that [[

]]. This is consistent with the results provided in Table 4.4-39S02-1 of the response.

The table provides the ring-averaged pressure drops using the modified TRACG04 initialization procedure. The results confirm that the core pressure drop [[ ]].

Because TRACG04 is explicitly modeling the chimney flow blockages and bypass flows and is run in an independent manner relative to the previously coupled manner, the staff determined

that these analyses demonstrate that thermal hydraulic communication through the bypass remains sufficient to equalize the core outlet pressure within sufficiently minor deviations; therefore, the PANAC11 assumption of [[ ]] is acceptable.

References

4.4-39.1 MFN-06-350, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," September 29, 2006 (ADAMS Accession Nos. ML062890047, ML062890048).

4.4-39.2 MFN-06-350, Supplement 3, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application—RAI Number 4.3-2 S01 and 4.4-39 S01," June 15, 2007 (ADAMS Accession Nos. ML0719302452, ML071930240).

4.4-39.3 MFN-08-949, Kingston, R., General Electric Hitachi Nuclear Energy America, LLC, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 106—Related to ESBWR Design Certification Application—RAI Number 4.4-39 Supplement 2," December 15, 2008 (ADAMS Accession Nos. ML083520263, ML083520264).

4.4-39.4 MFN-08-604, Brown, R., General Electric Hitachi Nuclear Energy America, LLC, letter to U.S. Nuclear Regulatory Commission, "Transmittal of Response to NRC Request for Additional Information—NEDC-32906P, Supplement 3, 'Migration to TRACG04/PANACII from TRACG02/PANACI0 for TRACG AOO and ATWS Overpressure Transients,' (TAC No. MD2569)," July 30, 2008 (ADAMS Accession Nos. ML082140580, ML082140581).

4.4-39.5 NEDC-32956P, Rev. 0, "TRACG User's Manual," GENE, February 2000 (ADAMS Accession Nos. ML003688152, ML003688292).

B.12 RAIs 4.4-44 and 4.4-47

The staff requested that GEH clarify the basis for the qualification studies provided in the NEDC-33239P LTR. The responses state that TIP adaption is not credited in the comparison of the PANAC11 calculations to TIP measurements. However, the PANAC11 comparisons to the gamma scan data use the TIP adaptive process; therefore, the shape adaption of the core simulator is credited. The staff determined that the response is acceptable.

References

4.3-4.11 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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,"

August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

B.13 RAIs 4.4-45 and 4.4-46

The staff requested that GEH provide additional details of the hot and cold design-basis eigenvalues. In the design process, design-basis eigenvalues are determined to account for known biases in the PANAC11-predicted core eigenvalue at hot and cold conditions. The biases are input into the code such that the core is analytically treated as critical when the calculated core eigenvalue under either hot or cold conditions is equal to the design-basis value.

The responses to the RAIs provide assurance that, while the design-basis eigenvalue is determined on a plant-specific basis, the trends in these parameters over a wide range of core designs and operating strategies remain fairly consistent, thus allowing the bias to be predicted for the equilibrium ESBWR core. The staff determined that this approach is acceptable; however, the staff noted that for the initial core there are no available ESBWR-specific plant data to verify the design-basis eigenvalue.

The staff requested that GEH provide additional details on how the design-basis eigenvalues are determined. The response to RAI 4.4-46S1 states that the calculation of the eigenvalue based on plant operating data and cold critical tests during startup provide the basis for the trend data and the assurance that the cold shutdown margin is maintained. The staff determined that the detailed explanation is acceptable. The use of plant data to qualify the nuclear design bases provides direct qualification and allows for the accurate consideration of methodology biases.

The response states that modern reactor core startups will provide data for use in the prediction of the initial core eigenvalue bias. The staff requested additional information regarding the process for determining the initial core design-basis eigenvalue during its review of LTR NEDC-33326P (Refs. 4.4-45.3 and 4.4-45.4). For the purposes of evaluating the methodology to account for the eigenvalue biases, the responses provide an adequate explanation of the trends and how the applicant accounted for them.

References

4.3-4.12 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, “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,” August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

4.4-45.2 MFN-06-297, Supplement 4, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, “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 4.2-5 S01, 4.2-6 S01, 4.2-7 S01 and 4.4-46 S01—Supplement,” January 26, 2007 (ADAMS Accession Nos. ML070380108, ML070380109).

4.4-45.3 MFN-08-087, Kinsey, J., General Electric Hitachi, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 137—Related to ESBWR Design Certification Application—RAI Numbers 4.3-11 and 4.4-68," February 4, 2008 (ADAMS Accession Nos. ML080380296, ML080380299).

4.4-45.4 NEDC-33326P, Pearson, G. and Trosman, L., Global Nuclear Fuel, Licensing Topical Report, "GE14E for ESBWR Initial Core Nuclear Design Report," July 31, 2007 (ADAMS Accession Nos. ML072040144, ML072040129, ML072040221).

B.14 RAI 4.4-48

The staff requested that GEH clarify the lattice code referenced in a section of the LTR. The response states that the reference is to TGBLA06. The staff requested that GEH specify which version of TGBLA06 it used to perform the analyses in the LTR. The response to RAI 4.4-48S1 states that the applicant performed the analyses using the standard production code version at the time of the release of the LTR, which was TGBLA06AE4. The staff determined that this response is acceptable.

References

4.4-48.1 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

4.4-48.2 MFN-06-297, Supplement 2, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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 4.3-4, 4.4-48, 4.8-7," December 21, 2006 (ADAMS Accession Nos. ML070110550, ML070110131).

B.15 RAI 4.4-49

The staff requested that GEH provide additional details regarding the standard depletion cases that are run in TGBLA06. The response states that the analysis considered void histories of 0 percent, 40 percent, and 70 percent in-channel void fraction. Parallel cases were run for controlled and uncontrolled depletion. At each depletion step, the instantaneous void fraction is branched and a TGBLA06 calculation is performed. The instantaneous void fraction branches are 0 percent, 40 percent, and 70 percent in-channel void fraction. The staff determined that the response is adequate in clarifying which TGBLA06 calculations were performed.

References

4.4-49.1 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

B.16 RAI 4.4-50

The staff requested additional details regarding the bundle-naming convention to ensure that the bundles described in the LTR were consistent with their designation and the analyses performed. The response describes each designation in the bundle name and provides the staff with adequate understanding of the bundle-naming convention to verify that the bundle designs are consistent with the analysis. Therefore, the staff determined that the response to this RAI acceptable.

References

4.4-50.1 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

B.17 RAI 4.4-51

The staff requested additional information regarding radial power distribution peaking factors for the ESBWR. The response states that the bundle-peaking factors are a strong function of the core design; however, bundle-peaking factors tend to be lower for larger cores because of the larger radial buckling (hence smaller flux gradients). The pin peaking factors are driven by the combination of gross nodal flux tilt and the infinite lattice peaking. The response compares lattice peaking for N-lattice arrangements. The response indicates that N-lattice edge rod peaking tends to be within the edge rod lattice peaking factors predicted for C- and D-lattice designs. The staff determined that the response accurately characterizes those aspects of the ESBWR design affecting radial power distribution. The response is acceptable insofar as it demonstrates that the ESBWR is not expected during normal operation to experience much greater power peaking as a function of its design.

References

4.4-51.1 MFN-06-350, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," September 29, 2006 (ADAMS Accession Nos. ML062890047, ML062890048).

B.18 RAI 4.4-52

The staff requested that GEH provide comparisons of the TGBLA06-predicted lattice peaking factors, using the extrapolation technique from the 0 percent, 40 percent, and 70 percent depletion cases and branch cases, to explicit TGBLA06 calculations at higher void fraction conditions (i.e., 90 percent). The applicant provided the results of the analysis in the response to RAI 4.4-52. The analyses indicate [[
]].

The staff requested in supplemental RAI 4.4-52S1 that GEH compare the difference in the local peaking factor to the uncertainty in lattice peaking. The analysis results indicate that below [[
]] for TGBLA06.

In RAI 4.3-3 the staff requested information regarding the TGBLA06 modification to the plutonium-240 resonance treatment. The response provided code-to-code comparisons to demonstrate the impact of the modification to calculations performed at high void fraction. The comparisons demonstrate that the extrapolation to high void fractions is essentially as accurate as detailed calculations performed explicitly at high void fraction.

Therefore, the staff determined that the information provided in the response to RAI 4.4-52S1, when considered in conjunction with the information provided in the response to RAI 4.3-3, is sufficient and acceptable.

References

4.4-52.1 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, “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,” August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

4.4-52.2 MFN-06-297, Supplement 1, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, “Supplemental 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 Number 4.3-3,” November 8, 2006 (ADAMS Accession Nos. ML063400067, ML063400074).

4.4-52.3 MFN-06-297, Supplement 5, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, “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 4.4-52 S01—Supplement,” February 8, 2007 (ADAMS Accession Nos. ML070470629, ML070470638).

B.19 RAI 4.4-53

The staff requested additional information regarding the shutdown margin calculations and the curve showing the change in shutdown margin with exposure. The response states that the shutdown margin is evaluated for an all-rods-in condition with the strongest rod withdrawn. This is dependent on the exposure of the cycle because the radial power shape has an effect on the rod worth, as does the depletion of burnable absorbers. The response states that the three-dimensional PANAC11 calculations explicitly capture these exposure-dependent effects. Therefore, the staff determined that the response acceptable.

References

4.4-53.1 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

B.20 RAI 4.4-54

The staff requested that GEH define the maximum fraction of limiting power density (MFLPD) and Critical Power Ratio (CPRRAT). The response states that the MFLPD is the maximum ratio of linear heat generation rate to the maximum linear heat generation rate limit.

The ratio is tracked on a nodal level and the maximum ratio is presented as the MFLPD. The CPRRAT is the maximum ratio of the OLMCPR to the assembly CPR. The ratio is tracked on a bundle level and the maximum is presented as the CPRRAT. The staff determined that the clarification is acceptable.

The staff requested in RAI 4.4-54S1 that the LTRs referencing the MLHGR and OLMCPR be internally-consistent by stating the ESBWR OLMCPR is [[ ]]. The applicant will revise the LTR accordingly. The staff determined that the response and revisions are acceptable.

References

4.4-54.1 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, Ml062480255).

4.4-54.2 MFN-06-297, Supplement 7, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, "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 Number 4.4-2S01, 4.4-27S01, 4.4-31S01 and 4.4-54S01," April 10, 2007 (ADAMS Accession No. ML071210063, ML071210066).

B.21 RAI 4.4-55

The staff requested that GEH provide additional descriptive details of the figures provided in the LTR. The staff reviewed the response and found that the clarification was acceptable.

References

4.4-55.1 MFN-06-297, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "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," August 23, 2006 (ADAMS Accession Nos. ML062480252, ML062480254, ML062480255).

B.22 RAI 7.2-5

In response to RAI 7.2-5, the applicant provided a detailed design description of the gamma thermometer device. [[

]]

[[

]] The staff reviewed these material choices and concluded that they ensure that the GT device will operate as intended under reactor conditions.

References

7.2-5.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.23 RAI 7.2-6

The staff requested additional information in RAI 7.2-6 regarding the material interactions, specifically in regard to dissimilar metal voltages and electrical heating energy deposition. In response, the applicant provided details regarding the instrument response to both [[

]], demonstrating that, while there are some differences in the cold junction voltage, the overall instrument response is not sensitive to the nature of the energy deposition in the insulated region.

References

7.2-6.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.24 RAI 7.2-7

In RAI 7.2-7, the staff requested additional information regarding the changes in heater wire resistance caused by irradiation effects. The staff was concerned that irradiation damage in the heat wire, and subsequent changes to the conductivity, would result in a systematic calibration error in high flux regions of the reactor. The applicant provided an [[

]]. The calculations show an expected change of approximately [[ ]] in the heater wire resistance. The response also states that [[

]]. The staff determined that the influence of neutron irradiation on the heater wire does not result in resistance changes that are significant enough to invalidate the uncertainty analysis, and the analysis is therefore acceptable. However, if the method proposed in the response to RAI 7.2-7 is deemed appropriate to improve accuracy by accounting for GT heater wire resistance changes during irradiation, such changes should be submitted for NRC review.

References

7.2-7.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.25 RAI 7.2-8

The staff requested additional information regarding the potential interaction between the cable heat current and the thermocouple voltage as a source of potential error in the GT signal. In response to RAI 7.2-8, the applicant evaluated the electrical conduction through the insulating materials and demonstrated that the material interfaces will effectively electrically isolate the heater wire and thermocouple thermoelement and thus will not impact the fidelity of the thermocouple signals during either normal operation or calibration.

References

7.2-8.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.26 RAI 7.2-9

The staff requested that GEH provide additional details of the detector sensor to power ratio. The response to RAI 7.2-9 indicates that NEDC-33239P describes the model. The staff determined that the response is acceptable insofar as it specifies the detector sensor to power ratio model.

References

7.2-9.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.27 RAI 7.2-10

In response to RAI 7.2-10, the applicant stated that several improvements were made to the GT following the Limerick 2 test and that these improvements will be likewise applied to the ESBWR design. These improvements include [[

]].

References

7.2-10.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.28 RAI 7.2-11

In RAI 7.2-11, the staff requested that the applicant address the ramifications of sensitivity decrease modeling for the purpose of extended durations between calibrations given that the model may misrepresent the actual change in GT sensitivity. The response stated that irradiation damage to the materials may cause changes in the electrical resistance, which in turn may lead to an increase or decrease in sensitivity during the initial stages of operation. This explanation is inconsistent with the evaluation of resistance changes under irradiation provided in response to RAI 7.2-7.

Therefore, the staff does not agree with the basis for the mechanistic model to predict the GT sensitivity and does not approve the use of the sensitivity decrease model. Accordingly, the staff approves the use of GT for in-core instrumentation for the ESBWR provided that the GT sensitivity be established through calibrations using the in-line heaters before adaption or LPRM calibration.

References

7.2-11.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.29 RAI 7.2-12

In response to RAI 7.2-12, the applicant provided an assessment of the [[

]] The staff agrees that this degree of contamination is essentially negligible.

References

7.2-12.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.30 RAI 7.2-13

In response to RAI 7.2-13, the applicant provided a rough model to estimate the change in sensitivity of the instrument based on core bypass conditions. In general, the sensitivity of the instrument scales proportionally to the thermal conductivity of the core material, assuming that the fill gas provides near-perfect insulation. While the core bypass temperature during normal operation should remain at the saturation temperature, this may not be the case during transient or off-normal conditions. Therefore, the sensitivity response to core bypass conditions is further justification for the condition that GT sensitivity be determined before any adaption or LPRM calibration.

The prerequisites for LPRM calibration and GT adaption cited in response to RAI 7.2-59S2 are adequate to ensure that following bypass temperature changes GT calibration will account for changes in the instrument sensitivity.

In response to RAI 4.3-2S2-C-2, the applicant provided additional information in regard to the GT sensitivity change as a result of bypass void formation.

References

7.2-13.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma

Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-13.2 MFN-07-613, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Letter No. 137 Related to ESBWR Design Certification Application—RAI Number 7.2-59 Supplement 2," July 3, 2008 (ADAMS Accession Nos. ML081920699, ML081920700).

7.2-13.3 MFN-08-293, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Letter No. 106—Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 Supplement 2 and 4.3-2 Supplement 2," July 3, 2008 (ADAMS Accession No. ML081930310, ML081930311).

B.31 RAI 7.2-14

The staff requested that GEH describe how automated fixed in-core probes (AFIPs) allow for accurate axial power shape monitoring. In particular, the staff requested information regarding the capability of the AFIPs to measure power shapes with multiple local axial peaks (e.g., double-humped power shapes).

The staff did not find the original RAI response acceptable because it referenced a qualification basis that had not been provided to the staff. The response to the supplement refers to RAI 4.2-12S2. The response to RAI 4.2-12S2 describes the adaption study performed using the Plant E power shapes.

RAI 7.2-18S2 supersedes the uncertainty analysis and qualification basis presented in the response to RAI 4.2-12S2.

The responses to RAIs 7.2-55S1 (statistical rejection method), 7.2-66 (minimum instrumentation configuration), and 7.2-18S2 ([[ ]] technique) provide the necessary information for the staff to complete its review in this matter. These RAI responses describe how the adaption is performed, how many instruments are needed, and how instrument signals are screened for rejection. Therefore, the staff determined that information provided in other RAI responses supersedes the response to RAI 7.2-14S1.

References

7.2-14.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-14.2 MFN-07-162, Supplement 1, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 105 Related to ESBWR Design Certification Application—RAI Numbers 7.2-14 S01, 7.2-55 Supplement 1, 7.2-56 Supplement 1, 7.2-58 Supplement 1, 7.2-60 Supplement 1, 7.2-64 Supplement 1," April 4, 2008 (ADAMS Accession Nos. ML080990404, ML080990405).

B.32 RAI 7.2-15

The staff requested that GEH describe the relationship between the local gamma flux and the GT indication as well as the relationship between the four bundle power and the local gamma flux. The response refers to the response to RAI 7.2-9, which provides the reference to the detector sensor to power ratio methodology. The response also states that the GT indication is essentially proportional to the gamma flux. The more detailed description in the LTR describes the [[ ]] correction factor.

Because the GT are used to establish the power shape only, and not the radial power distribution (based on integrated string values), the staff concluded that it is not necessary to determine the precise value of the gamma flux, only its relative distribution. Therefore, the staff concluded that the response is acceptable.

References

7.2-15.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-15.2 MFN-05-079, GEH Nuclear Energy, Licensing Topical Report, "Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring, NEDE-33197P, Revision 0," September 30, 2005 (ADAMS Accession Nos. ML052700451, ML052700449, ML052700450).

B.33 RAI 7.2-16

The staff requested further information regarding the additional uncertainty term associated with having fewer than nine sensors. This RAI response is applicable to Revision 0 of the LTR.

The response provides the basis for determining the additional uncertainty, however, the determination of the GT-specific uncertainties provided in the response to RAI 7.2-18S2 supersedes the response. Therefore, the staff does not require the response provided to RAI 7.2-16 to complete its review.

References

7.2-16.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-16.2 MFN-07-544, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Number 7.2-18 Supplement 2," August 18, 2008 (ADAMS Accession Nos. ML082350337, ML082350338).

B.34 RAI 7.2-17

The response corrects an error in the topical report and refers to the response to RAI 4.2-12 as the basis for the LHGR uncertainty. The response is acceptable insofar as it acknowledges that the original revision of the LTR was in error and that the applicant will correct the error in the next revision.

References

7.2-17.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.35 RAI 7.2-18

RAI 7.2-18 requested information regarding LPRM calibration. The original LTR did not provide sufficient detail regarding the overall calibration process for the staff to complete its review.

In response to several RAIs, GEH has specified the different steps for performing an LPRM calibration. These steps include GT calibration, power shape adaption, and LPRM calibration. The responses to RAIs 7.2-59S2 and 7.2-66 provide information regarding the performance of the GT calibration and GT measurements. The response to RAI 7.2-18S2 provides the updated information regarding GT power shape adaption and the determination of the associated uncertainty components.

The response to RAI 7.2-18S2 provides updated information regarding the interpolation and adaption method and the uncertainty analyses for the ESBWR OLMCPR and MLHGR limits.

The response to RAI 7.2-18S2 supersedes information provided under RAIs 4.2-12S2, Parts 10, 11, 16, 17, 19, 20, and 25; 7.2-14S1; 7.2-51S1; 7.2-55S1; 7.2-56S1; 7.2-58S1; and 7.2-64S1. In many of these cases, the primary update is to specify [[ ]] as the adaption technique. In addition to revising the referenced adaption technique from [[ ]] to [[ ]], the response provides an update to the [[ ]].

The staff reviewed the basis for the adaption techniques as provided in various RAI responses and during an audit of the adaption and interpolation methods. The staff documented its findings in the review of the relevant RAI responses, particularly in the response to RAI 4.2-12S2-10.

Based on the information provided in response to RAIs 4.2-12S2-10 and 7.2-18S2, the staff determined that (1) the preliminary studies indicate that [[ ]] accuracy can rival [[ ]] accuracy if boundary conditions are improved and (2) the revised boundary conditions confer an acceptable degree of accuracy based on the uncertainty analyses presented in response to RAI 7.2-18S2.

B.35.1. Operating Limit Minimum Critical Power Ratio Uncertainties

The applicant performed specific studies to determine the GT-equivalent components of the update uncertainty using the previous generic NEDC-32694P-A methodology as a baseline.

GEH reevaluated the uncertainties associated with the [[ ]] and the [[ ]]. In both of these cases the values used for the equivalent components from NEDC-32694P-A were slightly conservative—[[ ]] in the case of the former and [[ ]] in the case of the latter. Therefore, the ESBWR OLMCPR analysis retained these uncertainties. The staff concluded that this approach is acceptable because the results are based on detailed sensitivity analyses using data collected from a GT- instrumented plant and rely on slightly conservative values.

The staff independently reviewed the other component uncertainties, including the [[ ]]. For the purpose of its review of the subject RAI response, however, the staff determined that the uncertainty components attributable to the GT CMS are appropriate for use in the OLMCPR determination.

B.35.2. Linear Heat Generation Rate Uncertainties

The uncertainties updated for the LHGR uncertainty analysis include the [[ ]], the [[ ]], and the [[ ]]. The applicant determined the [[ ]] using specific calibration data from a GT-instrumented BWR/5 with corresponding gamma TIPs. The applicant based its analyses on PANAC11 calculations, gamma TIP measurements, and offline PANAC11 calculations. The [[ ]] could be determined using the original dataset with additional data points. The [[ ]] is sensitive to the exposure interval. The applicant performed specific studies based on a linear model of the [[ ]] variation with exposure interval. Based on these studies, GEH is revising the LPRM calibration interval to reduce the [[ ]].

The staff reviewed the basis for the [[ ]] and determined that the magnitude of the uncertainty is consistent with the revised LPRM calibration interval based on the qualification dataset. The response states that GEH will revise the ESBWR technical specifications and bases to be consistent with the reduced calibration interval of 750 megawatt-days per metric tonne. The staff determined that this is acceptable. The resultant value based on the reduced interval is [[ ]].

The [[ ]] is based on a comprehensive study. The study compared the variation in LHGR with simulated GT failures consistent with the instrumentation configuration specified in the response to RAI 7.2-66. The statistical distribution of the LHGR differences is more sharply peaked than a normal distribution. In the downstream uncertainty evaluations the one standard deviation uncertainty value is conservatively taken as the [[ ]] maximum value. The staff agrees that this approach is conservative. The applicant determined the [[ ]] for LHGR to be [[ ]].

The final uncertainty is the [[ ]]. The [[ ]] is based on a combination of qualification studies performed for [[ ]] (BWR/5) and [[ ]] (BWR/6). The [[ ]] power shapes during the subject cycle were double-humped power shapes and hence were difficult to replicate with discrete fixed axial measurements and interpolation techniques. The analysis applied the [[ ]] boundary conditions, and a [[ ]] was determined using a process analogous to the methodology presented in the response to RAI 4.2-12S2. The resultant uncertainty was [[ ]]. The staff determined that the revised methodology and the uncertainty value are acceptable.

The response provides the final combination of all uncertainties. The appropriate component uncertainties were used to account for interim methods penalties in the LHGR. The resultant LHGR uncertainty for the reduced calibration interval was [[ ]]. The value of [[ ]] is consistent with the [[ ]] value assumed in the thermal mechanical limit analysis methodology and is therefore acceptable.

References

7.2-18.1 MFN-07-162, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-18.2 MFN-07-544, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 105 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-11 S01, 7.2-16 S01, and 7.2-18 S01," November 8, 2007 (ADAMS Accession Nos. ML073180085, ML073180086).

7.2-18.3 MFN-07-544, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Number 7.2-18 Supplement 2," August 18, 2008 (ADAMS Accession Nos. ML082350337, ML082350338).

7.2-18.4 MFN-08-621, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 169 Related to ESBWR Design Certification Application—RAI Number 7.2-66," August 18, 2008 (ADAMS Accession Nos. ML082330100, ML082330101).

B.36 RAI 7.2-19

In RAI 7.2-19, the staff requested that the applicant explain the validity of the factory-calibrated value of alpha when material properties change under the conditions of normal operation. Alpha is a device-specific parameter that accounts for non-linearity in the GT signal.

The applicant provided a detailed analysis of the changes in alpha with the material properties at nominal operating conditions in response to RAI 7.2-19. The response compared the change in alpha based on an analytical model and a detailed simulation. [[

]]

Since the effect on an individual GT signal is small given potential variations in the value of alpha, all of the GT in the core are subject to the same variation in properties, and the GT are used only to determine the relative power shape (following calibration to establish the sensitivity), the staff determined that the [[ ]] approach based on factory calibration is sufficient to capture the effects of nonlinearity. Furthermore, changes during normal operation in this value will not significantly impact the GT signals, thus contributing negligibly to the overall uncertainty.

References

7.2-19.1 MFN-07-321, Kinsey, J., General Electric Hitachi Nuclear Energy America, LLC, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-19, 7.2-20, 7.2-51," June 20, 2007 (ADAMS Accession Nos. ML071930536, ML071930538).

B.37 RAI 7.2-20

The staff requested additional information from the applicant regarding the basis for the power distribution uncertainties determined based on the data reported in NEDE-33197P.

B.37.1 RAI 7.2-20-A

The staff requested additional information regarding the bundle power uncertainty used in the OLMCPR analysis. The staff did not find the explanation provided in this response acceptable. The origin of the [[ ]] value is NEDC-32964P-A. The measurements that the response refers to relied on using TIPs. The ESBWR design does not include TIPs. Therefore, while the determination of the NEDC-32964P-A uncertainty included a greater number of measurements, these measurements are not indicative of the monitoring to be performed for the ESBWR. Furthermore, NEDC-32964P-A provides that the applicability of the numbers be demonstrated; specifically Item (3) provides that the 3D MONICORE bundle power calculational uncertainty be verified when applied to fuel and core designs not included in the benchmark comparisons. It is worth noting that, in developing the uncertainty, adaption to TIP measurements is credited when gamma scan measurements are used to determine the [[ ]]. Therefore, the staff does not agree with the statement that these benchmark comparisons are necessarily indicative of the ESBWR 3D MONICORE. Therefore, the staff requested supplemental information in RAI 7.2-20S1-A.

B.37.2 RAI 7.2-20S1-A

In RAI 7.2-20S1-A the staff requested additional information regarding the applicability of historically determined uncertainty parameters for the ESBWR. The response to RAI 7.2-20S1-A (Ref. 7.2-20.2) states that the applicant updated the uncertainty analysis for the ESBWR based on high power density plant data (greater than 50 kW/l) instrumentation qualification and the specific adaption methodology proposed for the ESBWR.

GEH provided details of the methodology and the specific uncertainty analysis to the staff in response to RAIs 4.2-12S2-10 and 4.2-12S2-11. GEH's response to RAI 7.2-20S1-A references GEH's response to RAI 4.2-12S2-11. The staff determined that the response to this RAI need not include a justification of the historical values as a specific uncertainty analysis has been performed in a separate RAI. The information supplied and specific reference to RAI 4.2-12S2-11 is sufficient for the staff to close RAI 7.2-20S1-A.

B.37.3 RAI 7.2-20-B

The staff requested that GEH describe the components of the power distribution uncertainty in terms of the NEDC-32964P-A component uncertainties. The response states that the applicant addressed the extrapolation in the response to RAI 7.2-9. The staff did not agree because the response to RAI 7.2-9 addresses the detector sensor to power ratio only. Furthermore, the staff

disagreed with the applicant's statement that the uncertainty is unexpected to change. The staff disagreed because a TIP trace provides direct measurement of the four bundle power at every nodal level, while the GT arrangement cannot. The staff reasons that it is counterintuitive to conclude that fewer measurements can result in the same uncertainty. Therefore, the staff requested additional information regarding this topic in RAI 7.2-20S1-B.

B.37.4 RAI 7.2-20S1-B

RAI 7.2-20S1-B requested that GEH comment specifically on the ramification of having an anomaly in one axial node that perturbs the power distribution locally and the efficacy of the GT arrangement to identify such an anomaly. The request did not specify the source of such an anomaly. GEH considers the presence of a fuel spacer at the same axial elevation as a GT to be a local axial perturbation.

GEH studied the spacer effect using three-dimensional MCNP analyses. GEH studied the [[ ]]. For spacers located near the GT sensor, the applicant performed specific MCNP analyses to determine the bias. These biases are independent of the in-channel void fraction, which is expected. The biases are determined using an acceptable methodology and, according to the LTR, are applied in the CMS. The staff determined that this methodology is acceptable to address the staff concern regarding the spacer effect on biasing the GT signal.

GEH further clarified that bypass void formation is considered an anomaly; GEH provided bypass void formation analysis of the detector response to the staff in response to RAI 4.3-2S2. The analysis considers the impact of the void formation on the gamma transport characteristics as well as the impact of voiding on the thermal characteristics of the heat transfer from the instrument tube to the bypass. The staff's review of the response to RAI 4.3-2S2 is documented in Section 0 of this SE.

B.37.5 RAI 7.2-20-C

Revisions to the power distribution uncertainty assessment rendered the response to RAI 7.2-20S1-C obsolete, as documented in the response to RAI 7.2-18S2.

B.37.6 RAI 7.2-20-D

The staff requested that GEH justify the [[ ]] used for the four bundle power and [[ ]] components of the bundle power uncertainty. The results in Table 7-18 refer to the GT core monitor study. The staff believed that a GT-simulated "readings" technique would have to be employed to perform corewide GT adaptions. Therefore, the staff requested additional information regarding this topic in RAI 7.2-20S1-D.

B.37.7 RAI 7.2-20S1-D

In RAI 7.2-20S1-D the staff requested that GEH clarify how it performed the K5 adaptions. The K5 reference report specifies that a GT CMS was run in parallel to adapt the power shape. This information is acceptable to close this RAI. The response to RAI 7.2-20S1-D provides the specific reference to material in the open literature that describes the K5 test and computational methodology. The response similarly compares the K5 gamma scan results to previous gamma scan campaign results performed using historical GEH methods (i.e., TGBLA04/PANAC10

methods). The responses state that the K5 gamma scan results were bounded by previous gamma scans performed using similar GEH core monitoring techniques and are therefore deemed applicable. The staff determined that this information adequate to close RAI 7.2-20S1-D.

As discussed in the preceding section, the GT CMS software, as described in the open literature, is substantially similar to the TGBLA04/PANAC10 methods, and the results obtained from the gamma scan are similar to those obtained using the TGBLA04/PANAC10 methods for other campaigns. The use of the more sophisticated TGBLA06/PANAC11 methods for the ESBWR assures that the K5 results are relatively conservative when known improvements in the methods are considered.

B.37.8 RAI 7.2-20-E

The staff requested that GEH provide a greater description of the [[ ]]. The staff requested clarification of the term “maximum average” in a supplemental request for information pursuant to RAI 7.2-58. The staff also asked for clarification regarding the value in Table 8-7 in terms of its relation to the data in Tables 7-3 and 7-4. The staff did not understand the applicability of these data because they are based on nine GT per string, which is not the proposed design for the ESBWR. Therefore, the staff requested additional information regarding this topic in RAI 7.2-20S1-E.

B.37.9 RAI 7.2-20S1-E

In RAI 7.2-20S1-E the staff requested that GEH justify the applicability of the K5 and Tokai 2 test data. The response (Ref. 7.2-20.2) states that the applicant included the Tokai 2 test data to determine the GT detector uncertainty. The calculational uncertainty is determined according to an adaption study documented in response to RAI 4.2-12S2-10, which accounts for the number of GT instruments per string. The response similarly explains the term “maximum average.” The applicant justified the applicability of the K5 test data in response to RAI 7.2-20S1-D and the applicability of the Tokai 2 test data in response to RAI 7.2-20S1-E by discussing the scope of the applicability of the GT – to - nTIP values to qualify the instrumentation during the test. The K5 and Tokai 2 reactors are large, high power reactors, and are therefore deemed appropriate to qualify the GT instruments for the ESBWR. The staff determined that the response is sufficient to justify the applicability of the test data in light of the scope of its use in determining the overall efficacy of the GT CMS performance.

Tokai 2 was a test case for the improvement in GT CMS accuracy based on assigning unique [[ ]] values in the core monitoring software to each individual GT. GEH will adopt this improvement for the ESBWR, further justifying the applicability of the Tokai 2 test data. GEH provided this information to the staff in response to RAI 7.2-18S2.

B.37.10 RAI 7.2-20-F

Revision to the power distribution uncertainty assessment rendered the response to RAI 7.2-20S1-F obsolete, as documented in the response to RAI 7.2-18S2.

B.37.11 RAI 7.2-20-G

The staff requested information regarding the uncertainty analysis in terms of LPRM and GT adaption. The staff determined that the response was inconsistent with the LTR and requested supplemental information in RAI 7.2-20S1-G.

B.37.12 RAI 7.2-20S1-G

In response to RAI 7.2-20S1-G, GEH provided details of the GT calibration and adaption procedure that address transient changes in plant parameters affecting GT sensitivity and responsiveness. In regard to the GT response, the staff noted signal lag attributable to thermal inertia effects. To appropriately calibrate the instrument, the ohmic heating current should be held constant for a fixed duration to allow the GT to reach a steady signal.

GEH provided information regarding the thermal time constants for the GT instruments. Factory measurements indicate that the GT thermal time constant is [[ ]]. The response to RAI 7.2-20S1-G confirms that the time constant is less than [[ ]]. The response states that the current hold time is a minimum of five thermal time constants. Therefore, the staff determined that the response and the proposed LTR revision are acceptable.

B.37.13 RAI 7.2-20-H

The staff requested that GEH describe the relationship between Table 9-13 and the power distribution uncertainties. The response states that these values in Table 9-13 were not used in establishing the bundle power uncertainty. The staff determined that this clarification is acceptable.

References

7.2-20.1 MFN-07-321, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, “Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-19, 7.2-20, 7.2-51,” June 20, 2007 (ADAMS Accession Nos. ML071930536, ML071930538).

7.2-20.2 MFN-07-321, Supplement 1, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, “Response to Portion of NRC Request for Additional Information Letter No. 105 Related to ESBWR Design Certification Application—RAI Numbers 7.2-20 Supplement 1, Parts A, D, E and 7.2-21 Supplement 1,” April 4, 2008 (ADAMS Accession Nos. ML081000251, ML081000252).

7.2-20.3 MFN-07-321, Supplement 2, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, “Response to Portion of NRC Request for Additional Information Letter No. 105 Related to ESBWR Design Certification Application—RAI Numbers 7.2-20 Supplement 1, Part B,” July 11, 2008 (ADAMS Accession Nos. ML081980193, ML081980195).

7.2-20.4 MFN-07-321, Supplement 3, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, “Response to Portion of NRC Request for Additional

Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Numbers 7.2-20 Supplement 1, Part G," August 18, 2008 (ADAMS Accession Nos. ML081980193, ML081980195).

B.38 RAI 7.2-51

The staff requested that GEH explain how the discrete GT signals are used in conjunction with interpolation techniques to determine the axial power distribution. The responses provided to RAIs 7.2-51 and 7.2-51S1 are obsolete based on the response to RAIs 4.2-12S2-10 and 7.2-18S2 which provide the interpolation technique and associated uncertainty.

References

7.2-51.1 MFN-07-321, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-19, 7.2-20, 7.2-51," June 20, 2007 (ADAMS Accession Nos. ML071930536, ML071930538).

7.2-51.2 MFN-07-321, Supplement 1, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 105 Related to ESBWR Design Certification Application—RAI Numbers 7.2-20 Supplement 1, Parts A, D, E and 7.2-21 Supplement 1," April 4, 2008 (ADAMS Accession Nos. ML081000251, ML081000252).

7.2-51.3 MFN-08-293, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 106—Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 Supplement 2 and 4.3-2 Supplement 2," April 3, 2008 (ADAMS Accession Nos. ML080990615, ML080990616).

7.2-51.4 MFN-07-544, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Number 7.2-18 Supplement 2," August 18, 2008 (ADAMS Accession Nos. ML082350337, ML082350338).

B.39 RAI 7.2-52

The staff requested additional information in terms of the influence of fuel spacers on the GT indications. The spacers would provide additional gamma shielding for GT located at axial locations adjacent to fuel spacers. In response to RAI 7.2-52, the applicant provided information regarding the relative shielding provided by the fuel spacers by plotting a gamma TIP trace and marking the small depression in the trace near fuel spacers.

The results show that the spacers produce nearly indiscernible depressions in the trace near the fuel spacers; however, the staff noted that the gamma TIP traces are performed based on a [[ ]].

Therefore, the gamma TIP nodal power sensitivity to fuel spacers would be significantly reduced relative to the GT instruments, which would have only one nodal reading to extrapolate the

nodal conditions. Thus, in RAI 4.2-12S2-22, the staff requested that the applicant perform detailed transport calculations considering the impact of fuel spacers on GT signal. GEH provided the results of these analyses in response to RAI 7.2-20S1-B. The response to RAI 7.2-20S1-B supersedes the response to RAI 7.2-52.

References

7.2-52.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-52.2 MFN-07-321, Supplement 2, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 105 Related to ESBWR Design Certification Application—RAI Numbers 7.2-20 Supplement 1, Part B," July 11, 2008 (ADAMS Accession Nos. ML081980193, ML081980195).

B.40 RAI 7.2-53

The staff requested that GEH describe how the [[ ]]. The response refers to the response to RAI 21.6-89. The response to RAI 7.2-53 is obsolete based on the response to RAI 4.2-12S2-22, which describes the gamma transport factor determination process, the J-factor methodology, and its implementation for GT instruments.

References

7.2-53.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-53.2 MFN-08-293, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 106 Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 Supplement 2 and 4.3-2 Supplement 2," July 3, 2008 (ADAMS Accession Nos. ML081930310, ML081930311).

B.41 RAI 7.2-54

The staff requested justification of the [[ ]] used to determine the bundle power uncertainty based on the K5 qualification. The response is obsolete based on a revision to the uncertainty assessment provided in the response to RAI 7.2-18S2.

References

7.2-54.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-54.2 MFN-07-544, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Number 7.2-18 Supplement 2," August 18, 2008 (ADAMS Accession Nos. ML082350337, ML082350338)

B.42 RAI 7.2-55

The staff requested that GEH provide additional information regarding the statistical control methodology. The response to RAI 7.2-55 did not sufficiently describe the methodology.

The response to RAI 7.2-55S1 states that the applicant evaluated the [[

]]

[[ ]] This approach is fully consistent with TIP adaption, and the staff determined that this approach is acceptable.

The response states that core monitoring statistical controls are also applied. [[

]] The staff determined that this approach is acceptable and will ensure that core power distribution measurements are made at steady-power conditions.

References

7.2-55.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-55.2 MFN-07-162, Supplement 1, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 105 Related to ESBWR Design Certification Application—RAI Numbers 7.2-14 S01, 7.2-55 Supplement 1, 7.2-56 Supplement 1, 7.2-58 Supplement 1, 7.2-60 Supplement 1, 7.2-64 Supplement 1," April 4, 2008 (ADAMS Accession Nos. ML080990404, ML080990405).

B.43 RAI 7.2-56

The staff requested additional information regarding the preferred technique for GT extrapolation. The response states that the detector sensor to power ratio is based on a model similar to that described in the response to RAI 21.6-89.

The staff, in its original RAI, requested that GEH describe how it would use discrete GT signals to determine the axial power shape at every nodal location. The staff requested in a supplemental request for information that GEH specify how it selected this technique. The response to RAI 7.2-56S1 references the response to RAIs 4.2-12S2-10 and 4.2-12S2-11.

These responses reference the [[ ]] technique. RAI 7.2-18S2 supersedes these responses and specifies that [[ ]] is used exclusively. Therefore, the staff determined that the information requested in RAI 7.2-56 is provided in the response to RAI 7.2-18S2.

References

7.2-56.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-56.2 MFN-07-162, Supplement 1, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 105 Related to ESBWR Design Certification Application—RAI Numbers 7.2-14 S01, 7.2-55 Supplement 1, 7.2-56 Supplement 1, 7.2-58 Supplement 1, 7.2-60 Supplement 1, 7.2-64 Supplement 1," April 4, 2008 (ADAMS Accession Nos. ML080990404, ML080990405).

7.2-56.3 MFN-08-293, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 106—Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 Supplement 2 and 4.3-2 Supplement 2," July 3, 2008 (ADAMS Accession Nos. ML081930310, ML081930311).

7.2-56.4 MFN-07-544, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Number 7.2-18 Supplement 2," August 18, 2008 (ADAMS Accession Nos. ML082350337, ML082350338).

B.44 RAI 7.2-57

The staff requested additional information in RAI 7.2-57 to address the effect of gamma streaming and the potential for cross-bundle interference in the GT indications. In response to RAI 7.2-57, the applicant stated that the primary means for communication across the bundles would be gamma streaming through the interassembly bypass region because the fuel itself would provide sufficient shielding to limit the effective signal to the nearest four bundles. The

previously approved model for gamma transport kernels for gamma TIP instruments is also based on the nearest four bundles, and the standard J-factor decreases significantly for the corners furthest from the instrument corner. The applicant stated that, while there will be neutron streaming in the bypass region and that this contribution is expected to be very small. The staff agrees as the length of the fuel bundle would effectively collimate the cross-bundle gamma sources and thus result in a very low gamma flux contribution. Additionally, the applicant stated that this cross-bundle effect is likely to exist for all GT in the core to a certain extent, and the normalization of the signals to determine the axial power shape would effectively normalize out any cross-bundle gamma transport effects.

The staff agrees with the applicant's assessment and determined that any additional uncertainty as a result of cross-bundle gamma transport through the bypass would have a negligible effect on the overall uncertainty assessment and would not preclude the GT from producing an indication representative of the local four bundle power.

References

7.2-57.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.45 RAIs 7.2-58 and 7.2-60

The staff requested additional information regarding the [[ ]]. Because revision of the method for calculating the power distribution uncertainties is reported in the response to RAI 7.2-18S2, the staff does not need additional information relating this uncertainty to the power distribution uncertainties. The [[ ]] is a tool for assessing the Tokai 2 qualification tests only.

The staff requested supplemental information in RAIs 7.2-58S1 and 7.2-60S1 regarding the exposure accrual methodology in the CMS.

In RAI 7.2-58S1 the staff requested that the applicant provide descriptive details addressing the effect of accrued exposure in the bundles surrounding a GT string. The response provided in Reference 7.2-58.2 states that the power shape adaption only determines [[ ]] correction constants to determine the axial power shape; however, it does [[ ]]. This practice is consistent with the operating fleet adaption method. The staff determined that the uncertainty analysis based on operating fleet experience is adequate to capture the effect of potential errors in [[ ]]. The staff determined that this response, in conjunction with the uncertainty analysis provided in response to RAI 4.2-12S2-11, adequately resolves its concerns.

In RAI 7.2-60S1 the staff requested that GEH consider the sensitivity of the uncertainty in nodal and bundle power to exposure. The response to RAI 7.2-60S1 (Ref. 7.2-58.2) states that the qualification provided against plant data from the Tokai 2 is not intended to qualify the interpolation methods. The response to RAI 7.2-58S1 describes the means for accruing exposure according to the PANAC11 methodology, and the response to RAI 7.2-60S1 states that a separate study was performed to determine the uncertainty attributed to the GT adaption procedure. The applicant provided the details of the adaptive method and the uncertainty

analysis in response to RAIs 4.2-12S2-10 and 4.2-12S2-11. The staff determined that the information provided in response to these RAIs is sufficient for the staff to close the open item associated with RAI 7.2-60.

The response to RAI 7.2-18S2 supersedes the responses to RAIs 4.2-12S2-10 and 4.2-12S2-11 insofar as it identifies a different adaption technique, modifies the basic [[ ]] method boundary conditions, and updates the uncertainty analysis. Therefore, the response to RAI 7.2-18S2 does not introduce changes in the core monitoring methodology relative to the treatment of exposure effects. As the treatment of exposure effects is unaffected by the methodology change described in the response to RAI 7.2-18S2 the staff did not have to perform another review of the responses to RAIs 7.2-58 and 7.2-60.

References

7.2-58.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-58.2 MFN-07-162, Supplement 1, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 105 Related to ESBWR Design Certification Application—RAI Numbers 7.2-14 S01, 7.2-55 Supplement 1, 7.2-56 Supplement 1, 7.2-58 Supplement 1, 7.2-60 Supplement 1, 7.2-64 Supplement 1," April 4, 2008 (ADAMS Accession Nos. ML080990404, ML080990405).

7.2-58.3 MFN-08-293, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 106—Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 Supplement 2 and 4.3-2 Supplement 2," July 3, 2008 (ADAMS Accession Nos. ML081930310, ML081930311).

7.2-58.4 MFN-07-544, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Number 7.2-18 Supplement 2," August 18, 2008 (ADAMS Accession Nos. ML082350337, ML082350338).

B.46 RAI 7.2-59

In RAI 7.2-59, the staff requested additional information regarding the delayed gamma compensation model. Based on its review, the staff did not find that the application contained sufficient information to permit the use of the GT system for transient monitoring. The response to RAI 7.2-59S2 states that the GT instruments are not intended for transient monitoring. The response further provides GT calibration and LPRM calibration and power shape adaption process details. For such calibrations the reactor should be in a steady-state condition. These conditions are equivalent to those for TIP power shape measurement and LPRM calibration and power shape adaption for the operating fleet.

The staff reviewed these provisions and finds that they are sufficient to ensure appropriate calibration because these provisions eliminate GT error due to transient effects. The final revision of the LTR includes these provisions.

The primary difference between the gamma TIP and GT calibration relates to control blade motion. The staff observed some biases in the Laguna Verde 2 test data when control blades were moved and the power was monitored using the GT system. The response is acceptable insofar as provisions described in the response preclude the introduction of any local biases as a result of blade motion.

References

7.2-59.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-59.2 MFN-07-613, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter Nos. 76, 100, and 105 Related to ESBWR Design Certification Application—RAI Numbers 7.1-53, 7.2-59 Supplement 1, 7.3-11, and 7.9-16 Supplement 1," November 21, 2007 (ADAMS Accession Nos. ML073330143, ML073330148).

7.2-59.3 MFN-07-613, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 137 Related to ESBWR Design Certification Application—RAI Number 7.2-59 Supplement 2," July 3, 2008 (ADAMS Accession Nos. ML081920699, ML081920700).

7.2-59.4 U.S. Nuclear Regulatory Commission, Audit Results Summary Report, "Gamma Thermometers for the ESBWR," August 2008 (ADAMS Accession No. ML082810409).

7.2-59.5 U.S. Nuclear Regulatory Commission, Audit Summary, "Final Audit Summary Including Phase 4 for the ESBWR Gamma Thermometer July 2008," November 5, 2008 (ADAMS Accession Nos. ML082940529, ML082940542).

B.47 RAI 7.2-61

In RAI 7.2-61, the staff requested that the applicant determine the GT lifetime and replacement schedule. The analyses provided by the applicant to assess the irradiation damage and [[ ]] indicated acceptable performance for up to 8 effective full-power years, which corresponds roughly to a fluence of $2x10^{22}$ nvt [neutron density times speed times time]. Eight effective full-power years is consistent with the LPRM lifetime. The applicant compared this lifetime to operating experience with GT at the Arkansas Nuclear One plant and found this operating life to be consistent with industry experience. The staff therefore determined that the GT lifetime predictions are reasonable. Furthermore, because the GT will be calibrated before use for calibration or adaption purposes, any additional drift in sensitivity over the GT lifetime will be corrected. Therefore, the staff agrees that concurrent replacement of the LPRMs and GT is an acceptable practice.

References

7.2-61.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.48 RAI 7.2-62

The staff requested that the applicant provide detector correlations; the GEH response to RAI 4.2-12S2-22 includes them. Therefore, the staff does not need further information to close the open item associated with RAI 7.2-62.

References

7.2-62.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-62.2 MFN-08-293, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 106—Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 Supplement 2 and 4.3-2 Supplement 2," July 3, 2008 (ADAMS Accession Nos. ML081930310, ML082940542).

B.49 RAI 7.2-63

The staff requested additional information regarding the sensitivity decrease model. In the response to RAI 7.2-63, the applicant stated that the sensitivity decrease model is not required because the GT sensors are calibrated before their use for LPRM calibration and power shape adaption. The staff determined that this is acceptable.

References

7.2-63.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.50 RAI 7.2-64

The response to RAIs 7.2-64 and 7.2-64S1 are obsolete based on a revision to the adaption technique described in the response to RAI 7.2-18S2. Therefore, the staff does not need the response to this RAI to complete its review of the subject LTRs.

References

7.2-64.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

7.2-64.2 MFN-07-162, Supplement 1, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 105 Related to ESBWR Design Certification Application—RAI Numbers 7.2-14 S01, 7.2-55 Supplement 1, 7.2-56 Supplement 1, 7.2-58 Supplement 1, 7.2-60 Supplement 1, 7.2-64 Supplement 1," April 4, 2008 (ADAMS Accession Nos. ML080990404, ML080990405).

7.2-64.3 MFN-07-544, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Number 7.2-18 Supplement 2," August 18, 2008 (ADAMS Accession Nos. ML082350337, ML082350338).

B.51 RAI 7.2-65

In RAI 7.2-65, the staff requested that the applicant address the potential to damage a GT during a reactor transient. The applicant evaluated the expected heat deposition in the GT core region during anticipated transients and determined the specific energy deposition to be approximately half of the saturation specific energy deposition. Therefore, the staff determined that the GT instruments will function properly following an anticipated transient condition.

References

7.2-65.1 MFN-07-162, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 78 Related to ESBWR Design Certification Application—Gamma Thermometers—RAI Numbers 7.2-5 through 7.2-18 and 7.2-52 through 7.2-65," May 14, 2007 (ADAMS Accession Nos. ML071490211, ML071490214).

B.52 RAI 7.2-66

The staff requested that GEH determine the minimum instrumentation configuration for the AFIP system that is used to complete the LPRM calibration surveillance requirement in Technical Specification Section 3.3.1.4.4. The response states that the minimum acceptable instrument configuration is as follows:

- [[

-

- [[

]]

The GEH uncertainty analysis for the ESBWR, in terms of the [[ ]] and [[ ]] provided in the response to RAI 7.2-18S2, is consistent with these conditions. Thus, the staff determined that the configuration and the uncertainty analysis are consistent and therefore acceptable.

References

7.2-66.1 MFN-08-621, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 169 Related to ESBWR Design Certification Application—RAI Number 7.2-66," August 18, 2008 (ADAMS Accession Nos. ML082330100, ML082330101).

7.2-66.2 MFN-07-544, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 127 Related to ESBWR Design Certification Application—RAI Number 7.2-18 Supplement 2," August 18, 2008 (ADAMS Accession Nos. ML082350337, ML082350338).

B.53 RAI 7.2-71

In RAI 7.2-71, the staff requested that GEH incorporate the staff-identified conditions, limitations, and restrictions (CLRs) during its review of the subject LTRs into the body of the "-A" version of the LTRs. The staff reviewed the CLR language (Ref. 7.2-71.1) to ensure consistency with the staff's intended CLRs.

B.53.1 CLR1: Peaking Factor Uncertainty and Fuel Exposure Condition

B.53.1.1 CLR1: Staff Wording

The calculated peak pellet exposure cannot exceed the validation range of the thermal mechanical methodology qualification database. The peaking factor uncertainties used in the MLHGR limit must represent the full range of fuel exposure.

B.53.1.2 CLR1: GEH Implementation

The LHGR infinite lattice pin power uncertainty must represent the full range of fuel lattice exposure values. The calculated peak pellet exposure must be confirmed to comply with the corresponding licensing limit approved by the NRC. The design analysis described in NEDC-33242P establishes the licensing limit for GE14E.

B.53.1.3 CLR1: Review

The CLR is incorporated in a revision to NEDE-33197P. The staff determined that the wording proposed by GEH to implement CLR1 reflects the staff's intended condition. Therefore, the staff determined that the response is acceptable.

B.53.2 CLR2: TGBLA06 8-Weight-Percent Gadolinia Restriction

B.53.2.1 CLR2: Staff Wording

TGBLA06 is not approved to analyze fuel lattices with gadolinia burnable poison loadings in excess of 8 w/o because the NRC has not quantified and reviewed the gadolinia bias.

B.53.2.2 CLR2: GEH Implementation

TGBLA06 is not approved to analyze fuel lattices with gadolinia burnable poison loadings in excess of 8 w/o gadolinia until the NRC staff quantifies and reviews the gadolinia bias.

B.53.2.3 CLR2: Review

The CLR is incorporated in a revision to NEDC-33239P. The staff determined that the wording proposed by GEH to implement CLR2 reflects the staff's intended condition. Therefore, the staff determined that the response is acceptable.

B.53.3 CLR3: Bypass Flow Lookup Table Condition

B.53.3.1. CLR3: Staff Wording

Licensing evaluations performed using either PANAC11 or TRACG04 for ESBWR operating state points other than the nominal operating state point (SP0) require that the bypass flow fraction lookup tables be evaluated for acceptability. If found to be unacceptable, these tables must be regenerated by TRACG and input in the core simulator in order to accurately determine the bypass flow.

B.53.3.2 CLR3: GEH Implementation

Licensing evaluations performed with PANAC11 must use bypass flow fractions consistent with all core operating states, as determined by TRACG04, and input in the core simulator to accurately determine the bypass flow. Bypass flow tables or explicit modeling of data from TRACG04 can be used for PANAC11 input values.

B.53.3.3 CLR3: Review

The CLR is incorporated in a revision to NEDC-33239P. The staff determined that the wording proposed by GEH to implement CLR3 reflects the staff's intended condition. The GEH implementation allows for explicit modeling of the alternative core operating state points using TRACG04 directly as opposed to initially evaluating the bypass flow lookup tables. Since the flow lookup tables are generated by TRACG04, the process of explicitly utilizing the TRACG04 results for the core simulator input is equally acceptable. Therefore, the staff determined that the response is acceptable.

B.53.4 CLR4: Steady-State 5-Percent Bypass Voiding Limitation

B.53.4.1 CLR4: Staff Wording

Bypass voiding under conditions of steady-state operation within the allowable operating domain must be analyzed and shown not to exceed 5 percent at any LPRM location.

B.53.4.2 CLR4: GEH Implementation

The bypass voiding will be evaluated on a cycle-specific basis to confirm that the void fraction remains below 5 percent at all LPRM levels when operating at steady-state conditions at the upper boundary of the allowable operating domain.

B.53.4.3 CLR4: Review

The CLR is incorporated in a revision to NEDC-33239P. The staff determined that the wording proposed by GEH to implement CLR4 reflects the staff's intended condition. The GEH implementation language clarifies that the analysis is cycle specific. The staff agrees with this clarification. The GEH language also specifies that the evaluation will be performed at the upper boundary of the allowable operating domain. In this context, the upper boundary refers to the highest thermal power according to the operating map. The staff determined that the bypass void fraction will be greatest at the highest power levels. Therefore, the staff determined that the analysis conditions are appropriate to bound the anticipated bypass void fraction for the entire operating domain. Therefore, the implementation wording specifies the analysis conditions (upper boundary), but the analysis will be bounding for the entire operating domain. Thus, the staff determined that the response is acceptable.

B.53.5 CLR5: R-Factor Condition

B.53.5.1 CLR5: Staff Wording

The bundle R-factor must be calculated using representative lattice pin power distributions and axial void and power profiles.

B.53.5.2 CLR5: GEH Implementation

The bundle R-factor must be calculated using representative lattice pin power distributions and axial void and power profiles.

B.53.5.3 CLR5: Review

The CLR is incorporated in a revision to NEDC-33239P. The staff determined that the wording proposed by GEH to implement CLR5 is identical to the staff's wording. Therefore, the staff determined that the response is acceptable.

B.53.6 CLR6: Scram Reactivity Calculation Condition

B.53.6.1 CLR6: Staff Wording

The scram reactivity calculated using the PANAC11 neutronic solver must be calculated with Doppler reactivity feedback modeling activated to accurately determine the reactivity effect of the blades without including the Doppler reactivity in the scram reactivity.

B.53.6.2 CLR6: GEH Implementation

The scram reactivity calculated using the PANAC11 neutronic solver must be calculated with Doppler reactivity feedback modeling activated to accurately determine the reactivity effect of the blades without including the Doppler reactivity in the scram reactivity.

B.53.6.3 CLR6: Review

The CLR is incorporated in a revision to NEDC-33239P. The staff determined that the wording proposed by GEH to implement CLR6 is identical to the staff's wording. Therefore, the staff determined that the response is acceptable.

B.53.7 CLR7: Boron Branch Limitation

B.53.7.1 CLR7: Staff Wording

The TGBLA06 borated libraries must be generated with lattice boron inventories between 600 parts per million (ppm) and 1,000 ppm natural boron equivalent.

B.53.7.2 CLR7: GEH Implementation

For the standby liquid control system shutdown analysis, the TGBLA06 borated libraries must be generated with lattice boron inventories between 600 ppm and 1,000 ppm natural boron equivalent.

B.53.7.3 CLR7: Review

The CLR is incorporated in a revision to NEDC-33239P. The staff determined that the wording proposed by GEH to implement CLR7 reflects the staff's intended condition. The GEH wording clarifies that the boron branch limitation is applied to the standby liquid control system shutdown analysis. PANAC11 only utilizes the boron libraries to perform this calculation. Therefore, the staff determined that the clarification is appropriate. Thus, the staff determined that the response is acceptable.

B.53.8 CLR8: Lattice Peaking Factor Uncertainty for OLMCPR Condition

B.53.8.1 CLR8: Staff Wording

The R-factor uncertainty used in determining the OLMCPR must be consistent with the LHGR uncertainty determined consistent with Condition 13 (below) or a conservatively high value.

B.53.8.2 CLR8: GEH Implementation

NEDC-32601P-A describes the method for calculating the R-factor uncertainty. When determining the R-factor uncertainty for ESBWR analyses, the infinite lattice peaking model uncertainty value will be assumed as equal to/or more conservative than, the LHGR infinite lattice peaking factor uncertainty value for a particular ESBWR core loading.

Any change of the uncertainty value of CLR8 must be submitted to the NRC before the change is incorporated into any safety analysis basis.

B.53.8.3 CLR8: Review

The CLR is incorporated in a revision to NEDE-33197P. The staff determined that the wording proposed by GEH to implement CLR8 reflects the staff's intended condition. The GEH wording clarifies that the uncertainty referenced by the staff is the infinite lattice peaking factor uncertainty. This uncertainty is analogous to the infinite lattice peaking model uncertainty in NEDC-32601P-A. Therefore, the GEH implementation language reflects the staff's intent of ensuring that the uncertainties used in generating the OLMCPR are appropriate. Therefore, the staff determined that the response is acceptable.

53.9 CLR9: [[ ]] Condition

B.53.9.1 CLR9: Staff Wording

The bundle power distribution uncertainty used in determining the OLMCPR must be calculated according to a [[ ]] prescribed in the most recently reviewed and approved version or supplement of NEDC-33173P-A or a conservative value.

B.53.9.2 CLR9: GEH Implementation

The [[ ]] is a component of the LHGR and OLMCPR calculation uncertainties. Its value is determined using a [[ ]] on gamma scan data. NEDC-33173P-A reports the value determined using this approach as [[ ]].

The applicability of… [CLR9] is dictated by the [[ ]] approved in NEDC-33173P. Should the NRC approve an alternative approach for establishing the aforementioned uncertainties in subsequent supplements to or revisions of the NEDC-33173P LTR, the approved, alternative approach may be adopted in NEDE-33197P-A in lieu of [this condition] without separate NRC review and approval.

Any change of the uncertainty value of CLR9 must be submitted to the NRC before the change is incorporated into any safety analysis basis.

B.53.9.3 CLR9: Review

The CLR is incorporated in a revision to NEDE-33197P. The staff determined that the wording proposed by GEH to implement CLR9 reflects the staff's intended condition. The GEH wording provides more detail regarding the basis for the [[ ]]. The GEH wording also clarifies that the [[ ]] may be revised in accordance with subsequent approved supplements to NEDC-33173P without NRC review and approval. Furthermore, the GEH condition provides a commitment to inform the NRC of any of these changes. Therefore, the staff determined that the response is acceptable.

B.53.10 CLR10: Flux Harmonic Calculation

B.53.10.1 CLR10: Staff Wording

The regional mode stability analysis must be performed using a radial nodalization in TRACG04 based on the PANAC11-generated first harmonic mode. The harmonic calculation performed by PANAC11 must use a full-core representation.

B.53.10.2 CLR10: GEH Implementation

The regional mode stability analysis must be performed using a radial nodalization in TRACG04 based on the PANAC11-generated first harmonic mode. The harmonic calculation performed by PANAC11 must use a full-core representation.

B.53.10.3 CLR10: Review

The CLR is incorporated in a revision to NEDC-33239P. The staff determined that the wording proposed by GEH to implement CLR10 is identical to the staff's wording. Therefore, the staff determined that the response is acceptable.

B.53.11 CLR11: Void Exposure History Bias Condition

B.53.11.1 CLR11: Staff Wording

Use of PANAC11-generated nuclear data for ESBWR reload transient analyses (AOO, stability, or ATWS) requires that TRACG utilize the void reactivity coefficient correction model described in response to RAI 21.6-111. The fuel lattices input to the model must represent the cycle-specific fuel loading.

B.53.11.2 CLR11: GEH Implementation

Use of PANAC11-generated nuclear data for ESBWR reload transient analyses (AOO, stability, or ATWS) requires that TRACG utilize the void reactivity coefficient correction model described in NEDE-32906P-A, Supplement 3. The fuel lattices input to the model must represent the cycle-specific fuel loading.

B.53.11.3 CLR11: Review

The CLR is incorporated in a revision to NEDC-33239P. The staff determined that the wording proposed by GEH to implement CLR11 reflects the staff's intended condition.

The response to RAI 21.6-111 is identical to the response to RAI 30 from the staff's review of NEDE-32906P-A, Supplement 3 (Refs. 7.2-71.2 and 7.2-71.3). Therefore, GEH has provided an alternative reference to an equivalent methodology. The balance of the condition language is identical. Therefore, the staff determined that the response is acceptable.

B.53.12 CLR12: Local Geometry Refinement Condition

B.53.12.1 CLR12: Staff Wording

As required on a cycle-specific basis, the methodology used to generate the response to RAI 7.2-20S01-B must be used to quantify any GT-specific spacer geometry biases. These biases must be input and utilized in the CMS.

B.53.12.2 CLR12: GEH Implementation

The parameters used to compensate for biases introduced in the GT sensor signal by the proximity to spacers or fuel type changes or both will be determined only when a new bundle

design (i.e., new axial lattice composition) or a new spacer design (i.e., material) is applied to a particular ESBWR core loading, as described in Section 8.6 of NEDE-33197P-A. The parameters will be incorporated into the GT-based monitoring system on a cycle-specific basis, as required.

B.53.12.3 CLR12: Review

The CLR is incorporated in a revision to NEDE-33197P. The staff determined that the wording proposed by GEH to implement CLR12 reflects the staff's intended condition. The response to RAI 7.2-72 incorporates the methodology of the response to RAI 7.2-20S01-B to determine the local geometry effects of spacers and hybrid nodes. The response to RAI 7.2-72 incorporates the comprehensive local geometry treatment methodology in Section 8.6 of NEDE-33197P. Therefore, the condition incorporates the methodology evaluated by the staff in the response to RAI 7.2-20S01-B.

The response clarifies that the geometry effects are fuel product line dependent. The staff agrees with this distinction, but noted that the methodology may not be valid unless the parameter values are input into the CMS to reflect the core loading and the location of specific bundle types within the core relative to the GT instrumented locations. The GEH implementation language reflects the staff's condition in that the CMS parameters are input on a cycle-specific basis. Therefore, the staff determined that the response is acceptable.

B.53.13 CLR13: Lattice Peaking Factor Uncertainty for MLHGR Condition

B.53.13.1 CLR13: Staff Wording

Nuclear design methodology uncertainties applied in the MLHGR are expected to be fuel product dependent. Infinite lattice peaking factor uncertainties applied in the MLHGR must be (1) consistent with the [[ ]], as reported in the response to RAI 4.3-2S02-A, for the specific GE14E fuel design when GE14E is loaded, (2) generated using the statistical approach approved in the most recently approved revision of or supplement to NEDC-33173 and based on the specific fuel product, or (3) conservative relative to (1) and (2).

The NRC staff would not consider changes to the infinite lattice peaking factor uncertainty that conform to the above requirements to constitute a departure from a method of evaluation in the safety analysis, and they may be used for licensing calculations without prior NRC review and approval. Should these values change, they must be documented in the cycle-specific core operating limit report (COLR) or supplemental reload licensing report (SRLR).

B.53.13.2 CLR13: GEH Implementation

The LHGR infinite lattice peaking factor uncertainty value is determined as the [[ ]] using the statistical analysis of the population of peak power as a function of exposure. The GE14E-specific LHGR infinite lattice peaking factor uncertainty determined using this approach is [[ ]]. This uncertainty will be determined whenever a new fuel product is applied to a particular ESBWR core loading.

The applicability of CLR13 is dictated by the [[ ]] approved in NEDC-33173P. Should the NRC approve an alternative approach for establishing the aforementioned uncertainties in subsequent supplements to or revisions of the NEDC-33173P

LTR, the approved, alternative approach may be adopted in NEDE-33197P-A in lieu of this condition without separate NRC review and approval.

Any change of the uncertainty value of CLR13 must be submitted to the NRC before the change is incorporated into any safety analysis basis.

B.53.13.3 CLR13: Review

The CLR is incorporated in a revision to NEDE-33197P. The staff determined that the wording proposed by GEH to implement CLR13 reflects the staff's intended condition. The GEH wording provides the GE14E-specific LHGR infinite lattice peaking factor uncertainty. The GEH wording also clarifies that the uncertainty may be revised to be consistent with subsequent approved supplements to NEDC-33173P without NRC review and approval. In addition, the GEH condition provides a commitment to inform the NRC of any of these changes. Therefore, the staff determined that the response is acceptable.

B.53.14 CLR14: GT Operability Condition

B.53.14.1 CLR14: Staff Wording

Failure of a GT heater requires that the GT string be declared as inoperable.

B.53.14.2 CLR14: GEH Implementation

The failure of a GT heater is considered a loss of calibration capability of the full GT string (all sensors). Therefore, in case of failure of a GT heater, the GT CMS will declare the GT string as inoperable.

B.53.14.3 CLR14: Review

The CLR is incorporated in a revision to NEDE-33197P. The staff determined that the wording proposed by GEH to implement CLR14 reflects the staff's intended condition. The GEH wording clarifies the reason for declaring the GT inoperable.

The only difference between the staff condition and the GEH implementation is the additional clarification. Therefore, the staff determined that the response is acceptable.

B.53.15 CLR15: Code Usage Condition

B.53.15.1 CLR15: Staff Wording

The limitations on TGBLA06 and PANAC11 code usage, as described in the user manuals, are a condition of the acceptance of these methodologies for the ESBWR. Changes to the manuals that are made in accordance with the quality assurance procedures audited by the staff, as documented in the applicable reference, do not require NRC review and approval. However, if used in the safety analysis, the cycle-specific COLR or SRLR must document these changes.

B.53.15.2 CLR15: GEH Implementation

The limitations on TGBLA06 and PANAC11 code usage, as described in the user manuals, are a condition of the acceptance of these methodologies for the ESBWR. Changes to the manuals that are made in accordance with the quality assurance procedures audited by the staff, as documented in the applicable reference, do not require NRC review and approval. However, if used in the safety analysis, the cycle-specific SRLR must document these changes.

B.53.15.3 CLR15: Review

The CLR is incorporated in a revision to NEDC-33239P. The staff determined that the wording proposed by GEH to implement CLR15 reflects the staff's intended condition. The response specifies that the SRLR will provide the specific documentation—the original staff language allowed this documentation to be provided in either the SRLR or COLR. The staff, therefore, determined that the response is acceptable.

B.53.16 CLR16: Code Change Limitation 1

In regard to CLR16 through CLR21, Title 10 of the *Code of Federal Regulations* (10 CFR) 52.98, "Finality of Combined Licenses; Information Requests," outlines the regulatory change processes that may apply to address the potential for future code updates. Requirements for prior NRC review of future code updates are consistent with the definition of a methodology change in 10 CFR 50.59(a)(1) and the criteria of 10 CFR 50.59(c)(2)(viii) to ensure that the methodology is not adversely impacted for reload licensing or core monitoring purposes.

B.53.16.1 CLR16: Staff Wording

The NRC staff considers modifications to the models described in NEDC-33239P or MFN-098-96 (Ref. 7.2-71.4) to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval.

B.53.16.2 CLR16: GEH Implementation

The NRC staff considers modifications to the models described in NEDC-33239P-A or MFN-098-96 to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval.

B.53.16.3 CLR16: Review

The CLR is incorporated in a revision to NEDC-33239P. The staff determined that the wording proposed by GEH to implement CLR16 is identical to the staff's wording. Therefore, the staff determined that the response is acceptable.

B.53.17 CLR17: Adaption Method Condition

B.53.17.1 CLR17: Staff Wording

The NRC staff considers modifications to the adaption technique in the PANAC11-based GT CMS, described in NEDE-33197P, to constitute a departure from a method of evaluation in the

safety analysis, and they may not be used for licensing calculations without prior NRC review and approval.

B.53.17.2 CLR17: GEH Implementation

The NRC staff considers modifications to the adaption technique in the PANAC11-based, GT CMS, described in NEDE-33197P-A, to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval.

B.53.17.3 CLR17: Review

The CLR is incorporated in a revision to NEDE-33197P. The staff determined that the wording proposed by GEH to implement CLR17 reflects the staff's intended condition. The language is essentially identical. Therefore, the staff determined that the response is acceptable.

B.53.18 CLR18: Code Change Limitation 2

B.53.18.1 CLR18: Staff Wording

The NRC staff considers modifications to the TGBLA06/PANAC11 codes or the GT CMS software that result in inconsistency with the NEDC-33239P-A and NEDE-33197P-A LTRs to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval of the necessary revisions to the LTRs.

B.53.18.2 CLR18: GEH Implementation

B.53.18.2.1 NEDC-33239P

The NRC staff considers modifications to the TGBLA06/PANAC11 codes that result in inconsistency with the NEDC-33239P-A LTR to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval of the necessary revisions to the LTR.

B.53.18.2.2 NEDE-33197P

The NRC staff considers modifications to the TGBLA06/PANAC11 codes or the GT CMS software that result in inconsistency with the NEDE-33197P-A LTR to constitute a departure from a method of evaluation in the safety analysis, and they may not be used for licensing calculations without prior NRC review and approval of the necessary revisions to the LTR.

B.53.18.3 CLR18: Review

The CLR is incorporated in revisions to both NEDC-33239P and NEDE-33197P. The wording proposed by GEH to implement the CLR reflects the staff's intended condition. The primary difference in the implementation is that the condition is divided into two portions applicable to the specific LTR in which it is incorporated. The staff determined that this an acceptable means of documenting the CLR. Therefore, the staff determined that the response is acceptable.

B.53.19 CLR19: Code Change Limitation 3

B.53.19.1 CLR19: Staff Wording

The NRC staff does not consider updates to the PANAC11 nuclear methods to ensure compatibility with other NRC-approved methods (e.g., TGBLA06) to constitute a departure from a method of evaluation in the safety analysis (i.e., they may be used for licensing calculations without prior NRC review and approval) so long as the predicted ESBWR equilibrium cycle MLHGR or the downstream ΔCPR/ICPR for the potentially limiting transients (calculated by TRACG04) show less than a 1-standard-deviation difference.

B.53.19.2 CLR19: GEH Implementation

The NRC staff does not consider updates to the PANAC11 nuclear methods to ensure compatibility with other NRC approved methods to constitute a departure from a method of evaluation in the safety analysis. These updates may be used for licensing calculations without prior NRC review and approval so long as the predicted ESBWR equilibrium cycle MLHGR or the downstream ΔCPR/ICPR for the potentially limiting transients (calculated by TRACG04) show less than a 1-standard-deviation difference.

B.53.19.3 CLR19: Review

The CLR is incorporated in a revision to NEDC-33239P. The wording proposed by GEH to implement CLR19 reflects the staff's intended condition. The language is essentially identical. Therefore, the staff determined that the response is acceptable.

B.53.20 CLR20: Code Change Limitation 4

B.53.20.1 CLR20: Staff Wording

The NRC staff does not consider increases in the spatial or energy resolution in the TGBLA06 lattice physics method to constitute a departure from a method of evaluation in the safety analysis (i.e., they may be used for licensing calculations without prior NRC review and approval) so long as the uncertainties in the lattice parameters do not increase as a result. In all cases, the cycle-specific COLR or SRLR, if utilized in the safety analysis, must document modifications or updates done without prior NRC review and approval.

B.53.20.2 CLR20: GEH Implementation

The NRC staff does not consider increases in the spatial or energy resolution in the TGBLA06 lattice physics method to constitute a departure from a method of evaluation in the safety analysis. These updates may be used for licensing calculations without prior NRC review and approval so long as the uncertainties in the lattice parameters do not increase as a result. In all cases, the cycle-specific SRLR, if utilized in the safety analysis, must document the modifications or updates done without prior NRC review and approval.

B.53.20.3 CLR20: Review

The CLR is incorporated in a revision to NEDC-33239P. The wording proposed by GEH to implement CLR20 reflects the staff's intended condition. The GEH implementation language

specifies that the documentation will be contained in the SRLR. The staff wording allowed the documentation to be in either the COLR or the SRLR. Therefore, the staff determined that the response is acceptable.

B.53.21 CLR21: Code Change Limitation 5

B.53.21.1 CLR21: Staff Wording

The NRC staff does not consider changes in the numerical methods to improve code convergence to constitute a departure from a method of evaluation in the safety analysis (i.e., they may be used in licensing calculations without prior NRC review and approval).

B.53.21.2 CLR21: GEH Implementation

The NRC staff does not consider changes in the numerical methods to improve code convergence to constitute a departure from a method of evaluation in the safety analysis, and they may be used in the licensing calculations without prior NRC review and approval.

B.53.21.3 CLR21: Review

The CLR is incorporated in a revision to NEDC-33239P. The wording proposed by GEH to implement CLR21 reflects the staff's intended condition. The language is essentially identical. Therefore, the staff determined that the response is acceptable.

References

7.2-71.1 MFN-09-073, Kingston, R., General Electric Hitachi Nuclear Energy America, LLC, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 267 Related to ESBWR Design Certification Application—Instrumentation and Control Systems—RAI Number 7.2-71," February 2, 2009 (ADAMS Accession Nos. ML090350398, ML090350404).

7.2-71.2 MFN-08-483, Letter from General Electric Hitachi Nuclear Energy America, LLC, to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) 30 RE: GE Topical Report NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, (TAC No. MD2569)," May 30, 2008 (ADAMS Accession Nos. ML081550192, ML081550193).

7.2-71.3 MFN-08-504, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 147—Related to ESBWR Design Certification Application—RAI Number 21.6-111," June 24, 2008 (ADAMS Accession No. ML081780577).

7.2-71.4 MFN-098-96, Reda, R., General Electric, Letter to Jones, R.C., U.S. Nuclear Regulatory Commission, "Implementation of Improved GE Steady-State Nuclear Methods," July 2, 1996 (ADAMS Accession Nos. ML070400507, ML083660032).

B.54 RAI 7.2-72

The staff noted that PANACEA tracks the precise bundle geometry and lattice geometry before performing the hybridization and nodal diffusion calculations, thereby retaining sufficient information in the code to correct for the hybridization effect on the nodal GT J-factors.

The use of the GT CMS for dominant and plenum lattices that are hybridized should include a correction to the GT instrument response calculation to account for the specific axial geometry to ensure that biases are not introduced in the adaption and calibration process as a result of nodal hybridization.

PANAC11 retains the lattice-specific J-factors before hybridization, the GT sensor location, and the stack size of the lattices within the hybrid node. Therefore, refinement of the J-factor methodology is possible within PANAC11. PANAC11 may calculate the J-factors using a preprocessing step in the calculation that considers the smaller region of interest about the GT sensor. [[ ]] In either case, the accuracy of the GT prediction may be easily maintained within hybridized nodes by implementing a preprocessing step utilizing the information already within the PANAC11 representation of the core in the GT CMS. Therefore, such a refinement should be used within the GT CMS.

In RAI 7.2-72, the staff requested that GEH describe the details of the local axial geometry correction methodology in the GT CMS. The response provides the results of detailed nuclear simulations of the GT response based on spacer and lattice change local geometry effects. In particular, the results provided in Figure 3 of the response (Ref. 7.2-72.1) demonstrate that in certain instances (such as spacers near the dominant-to-plenum-zone transition) the volume weighting of the J-factors would lead to significant error in the measured nodal power [[ ]].

On the basis of the spacer effects study, GEH has determined that the effects of the local geometry on the GT signal [[ ]]. Figure 1 of the response depicts the spacer influence (Ref. 7.2-72.1). The influence of the spacer alone may contribute to [[ ]] in the GT measured nodal power based on gamma shielding. The staff audited these studies, which were based on detailed, sophisticated nuclear simulations (Ref. 7.2-72.2). The RAI response describes the methodology that relies on MCNP gamma transport calculations. MCNP is a highly accurate transport methodology, therefore, the staff finds that these calculations are performed using an acceptable approach. Therefore, the staff determined that the results are acceptable for deriving a form function for the local geometry correction terms in the GT CMS.

The response likewise addresses the potential for partially controlled nodes to introduce biases in the nodal J-factors. The staff noted that the J-factor for hybrid controlled/uncontrolled nodes combines the [[ ]]. The staff noted that under controlled conditions the lattice power distribution is heavily tilted towards the instrument corner. This radial power shift could result in large J-factors. When the control blade is inserted partially into a node with a GT, a linear averaging method could introduce an error by over- or under-estimating the contribution of the gamma source of the controlled portion of the node to the GT. Such an error may introduce a bias in the relation between the GT signal and the nodal power. Therefore, a similar weighting technique is

employed to combine these hybrid J-factors. The staff determined that the weighting technique is based largely on the gamma transport characteristics within the bypass and fundamentally captures the "range of vision" of the GT sensor. Employing a similar technique for all hybrid nodes is therefore appropriate. On this basis, the staff determined that the methodology appropriately accounts for the potential of the fine motion control rod drive to position a control blade partially within a GT instrumented node.

On the basis of the detailed nuclear simulations, the staff agrees that the functional form for the correction should be [[ ]]. The spacer effects study provides a reasonable basis for establishing the empirical constants in the [[ ]]. The response further states that the correction methodology will utilize fuel-specific parameters determined through the detailed nuclear simulation and will serve as input to the GT CMS on a cycle-specific basis (Ref. 7.2-72.1). The staff determined that this approach is acceptable and will ensure that the GT CMS includes appropriate model input on a cycle-specific basis to account for the cycle-specific fuel loading.

References

7.2-72.1 MFN-09-026, Kingston, R., General Electric Hitachi Nuclear Energy America, LLC, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 267 Related to ESBWR Design Certification Application—Instrumentation and Control Systems—RAI Number 7.2-72," January 29, 2009 (ADAMS Accession Nos. ML090330155, ML090330156).

7.2-72.2 U.S. Nuclear Regulatory Commission, Audit Results Summary Report, "Gamma Thermometers for the ESBWR," August 2008 (ADAMS Accession No. ML082810409).

B.55 RAI 21.6-54

The staff requested additional detailed information regarding the ESBWR core design to perform independent calculations.

B.55.1 RAI 21.6-54-A

The staff requested exposure data, fuel composition, and void history data at beginning of cycle, middle of cycle, and end of cycle, as calculated by PANACEA. The response provides these data. Therefore, the staff determined the response acceptable.

B.55.2 RAI 21.6-54-B

The staff requested the fuel, clad, and coolant temperature at hot full-power conditions. The response provides these data. Therefore, the staff determined the response acceptable.

B.55.3 RAI 21.6-54-C

The staff requested the size of the temperature and void perturbations used to determine the reactivity coefficients. The response provides these data. Therefore, the staff determined the response acceptable.

B.55.4 RAI 21.6-54-D

The staff requested the fuel density. The response provides these data. Therefore, the staff determined the response acceptable.

B.55.5 RAI 21.6-54-E

The staff requested the bundle materials and densities. The response provides these data. Therefore, the staff determined the response acceptable.

B.55.6 RAI 21.6-54-F

The staff requested information regarding the control rod design. The response provides these data. Therefore, the staff determined the response acceptable.

B.55.7 RAI 21.6-54-G

The staff requested dimensional information regarding Figure 1-1 in NEDC-33239P. The response provides these data. Therefore, the staff determined the response acceptable.

B.55.8 RAI 21.6-54-H

The staff requested clarification of the LTR language, including the term “shutdown margin.” The response provides the requested clarification and is consistent with the staff’s definition of the terms. Therefore, the staff determined the response acceptable.

B.55.9 RAI 21.6-54-I

The staff requested additional information regarding the fission rate distribution and power distribution for lattices 81802 and 81902 for higher exposures. The response provides these data. Therefore, the staff determined the response acceptable.

B.55.10 RAI 21.6-54-J

The staff requested that GEH clarify the pin power peaking factors. The staff specifically requested that GEH confirm whether the peaking factors took into account the effect of gamma smearing on the power distribution. The response states that the analysis did consider gamma smearing therefore, the staff determined this response acceptable.

The response includes in tabular form, the parameter values used to account for void history effects. The historical void parameter was slightly different from the information that the staff needed. Therefore, the staff requested additional information.

B.55.11 RAI 21.6-54S1

The staff requested information regarding the void fraction for each node during cycle exposure. The RAI states that the information may be provided as the relative water density for each node for a series of points during exposure. The staff additionally requested information regarding the complete void history for any particular bundle during its full residency in the core. The RAI stated that this information may be supplied by providing a shuffle sequence for the equilibrium

cycle that characterizes, for each bundle location within the core, the new bundle location for the beginning of the next cycle in equilibrium, the discharged bundles, and the bundle locations into which new fuel is loaded. The staff additionally requested the time duration between each depletion point.

The requested information was provided to the staff in response to RAI 21.6-54S1.

The information provided in response to RAI 21.6-54 was used to develop confirmatory calculations. The confirmatory calculations referenced by the staff in this review are discussed in Appendix A of this SE.

References

21.6-54.1 MFN-06-295, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 49 Related to ESBWR Design Certification Application—TRACG Application for ESBWR Stability Evaluation and NEDC-33239P, 'GE14 for ESBWR Nuclear Design Report'—RAI Numbers 4.4-14 and 21.6-54," August 22, 2006 (ADAMS Accession Nos. ML062480427, ML062480429).

21.6-54.2 MFN-06-295, Supplement 1, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 49 Related to ESBWR Design Certification Application—Nuclear Design—RAI Number 21.6-54," November 15, 2006 (ADAMS Accession Nos. ML063380204, ML063380208).

B.56 RAI 21.6-85

As part of the review of the subject nuclear design methods, the staff reviewed the interface of the nuclear design methods with the transient methodology. This code interface dictates the efficacy of the downstream transient methodology to demonstrate compliance with General Design Criteria (GDC) 20, "Protection System Functions," and 12, "Suppression of Reactor Power Oscillations." Therefore, the staff requested that GEH provide additional information regarding the PANACEA wrapup file in RAI 21.6-85. The PANACEA wrapup file is the body of information transferred from PANACEA to TRACG for subsequent transient calculations.

In response to RAI 21.6-85, the applicant provided a table of contents to a PANACEA wrapup file (Ref. 21.6-85.1). The staff reviewed the contents to determine whether the PANACEA wrapup file contained sufficiently detailed parameters to allow for the initialization of the TRACG power distribution while maintaining a sufficiently detailed characterization of the nuclear parameters to allow the TRACG kinetics solver to model the neutronic feedback. The wrapup file contains both the functional cross-sections and power distribution. Therefore, in the initialization procedure, the functional cross-sections are preserved, allowing for accurate feedback modeling. The staff determined that sufficiently detailed nuclear information is conveyed from the PANACEA wrapup file to TRACG to both initialize the model and provide for acceptable kinetic feedback modeling.

References

21.6-85.1 MFN-07-347, Kinsey, J., General Electric Hitachi Nuclear Energy America, LLC, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application—RAI Numbers 21.6-65 and 21.6-85," June 21, 2007 (ADAMS Accession No. ML071930530).

B.57 RAIs 21.6-86 and 21.6-94

The staff requested additional information regarding the bundle isotopic tracking method described in NEDC-33239, Revision 0. In its responses to these RAIs, the applicant specified that it is removing the model from the LTR and is not seeking approval of the model. The NRC approval of NEDC-33239P does not constitute NRC approval of the bundle isotopic tracking model.

References

21.6-86.1 MFN-06-467, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application—Chapter 4 and GNF Topicals—RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89," November 29, 2006 (ADAMS Accession Nos. ML063450255, ML063450251).

21.6-86.2 MFN-07-079, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 87—NEDC-33239P—RAI Number 21.6-94," March 29, 2007 (ADAMS Accession No. ML070930617).

21.6-86.3 MFN-07-079, Supplement 1, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 87—NEDC-33239P—RAI Number 21.6-94 Supplement 1," November 1, 2007 (ADAMS Accession No. ML073090053).

B.58 RAI 21.6-87

The staff requested additional information regarding the methodology for evaluating the CPR and thermal margin in the PANAC11 core simulator. In response to RAI 21.6-87, the applicant described the iteration method for the critical power determination and explained that iteration is performed from bundle powers above the critical power maintaining the same power shape. PANAC11 calculates the critical power by iterating the channel power until the PANAC11 predicted equilibrium quality intersects the critical quality in a single node calculated by using the appropriate GEXL correlation (Ref. 21.6-87.1). The staff reviewed the iteration technique and determined that it is sufficiently capable and therefore acceptable insofar as it is used to identify the critical power.

References

21.6-87.1 MFN-06-467, Hinds, D., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application—Chapter 4 and GNF Topicals—RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89," November 29, 2006 (ADAMS Accession Nos. ML063450255, ML063450251).

B.59 RAI 21.6-88

The staff requested additional information in RAI 21.6-88 regarding the calculation of the channel flow distribution calculation. The prediction of the individual bundle flows rates affects the efficacy of the nuclear design methodology to accurately predict the radial power distribution.

The applicant's response to RAI 21.6-88 did not include sufficient information for the staff to evaluate the application of this method to the ESBWR (Ref. 21.6-88.2). The staff requested supplemental information in RAI 21.6-88S1. Specifically, the staff requested that the applicant provide the following:

- the process used to select characteristics channels
- a comparison of the characteristic channels to those in the ESBWR core
- the process for calculating the flow in the characteristic channels
- the mathematical procedure for adjusting flow based on channel differences
- the correlated response surface for each channel parameter

In a supplement to the original RAI response (Ref. 21.6-88.3), the applicant provided sufficient details regarding the core flow distribution calculation for the staff to complete the review for application to the ESBWR. The revised topical report includes this information. The response indicates that PANACEA calculates the number of characteristic bundles at each exposure point during a depletion calculation. The total number of characteristic channels is the product of five factors. The first factor is the number of different bundle geometries. For the ESBWR equilibrium cycle core, there is only one bundle geometry. The second factor is equal to the number of different orifice types, which are two for the ESBWR.

The third factor is the crud factor. If crud buildup is considered, then the crud factor is two because the characteristic channels consider both a clean bundle and a bundle with crud.

The remaining factors relate to the power distribution. The radial factor is typically 2, as the characteristic channels will include a high radial power bundle and a low radial power bundle. The last factor is the axial factor. The axial factor is also typically two because the characteristic channels include both a top-peaked and a bottom-peaked power shape. For the ESBWR calculation, the PANACEA-calculated number of characteristic channels is therefore eight.

Reference 21.6-88.1 provides a more detailed description of the power-void outer loop iteration performed to converge on the final power and flow distribution. In the outer loop iteration, the relative heat deposition in the bypass and channel coolant flow is compared to the total core power. If the summation of the energy deposition rate in the coolant and the reactor core power

level do not agree, the fraction of the power removed by in-channel convection is adjusted and the flow distribution is recalculated. This is performed to establish the relative flow in the bypass and core channels. According to the information in Reference 21.6-88.3, the means for determining the bypass flow is using interpolation based on power flow tables. This is not fully consistent with the channel heat flux adjustment in the outer loop iteration described in Reference 21.6-88.1 and reiterated in Section 1.5.5 of NEDC-33239P.

This difference in the application for the ESBWR is related to the iterative use of TRACG calculations to determine the flow boundary conditions instead of the internal PANACEA automated plant heat balance module. The staff requested additional supplemental information in regard to the supplemental information already provided in RAI 21.6-88S2. The staff specifically requested that the applicant provide any differences in the process in the determination of the relative bypass to channel coolant flow using the methods in Reference 21.6-88.1 relative to the techniques for the ESBWR where the flow boundary conditions are determined by iteratively performing TRACG calculations. Additionally, in the supplemental request the staff asked that the applicant clarify the use of the axial number of nodes in the determination of the axial peaking factor for the channels.

In the response to the staff's request for supplemental information, the applicant described the process for developing the lookup table for the ESBWR calculation (Ref. 21.6-88.4). In this case, the bypass flow is established by determining a set of curves of bypass flow fraction as a function of the total core flow for a constant power level. TRACG is used to perform these analyses and the information is fed into the PANACEA calculation through a lookup table. The steady-state values are based on the SP0 operating state point.

The staff requested clarification information in RAI 21.6-88S3 to understand the statements made in the response to RAI 21.6-88S2. The response to RAI 21.6-88S3 was adequate to clarify the previous response.

The staff determined that the differences between the previously approved core flow distribution calculation and the ESBWR calculation are subtly different. However, as PANAC11 does not include a natural circulation model, the calculational process uses TRACG to predict the core flow rate. The staff determined that the thermal hydraulic modeling capabilities of TRACG are sufficiently sophisticated and accurate for this purpose, and therefore, its use is acceptable. However, the bypass flow fraction lookup table should be evaluated to determine if it is acceptable for use in the CMS to monitor power and flow distributions at off-rated conditions.

Secondly, while the core flow distribution calculation is simplified, the staff compared the range of parameter variation against the core design parameters and concluded that the analysis considered adequate parameter ranges.

The staff also determined that historically accurate radial power distribution calculations provide assurance that for BWR operating conditions the model is sufficiently robust to predict the radial channel flow distribution.

In terms of the [[ ]] assumption, the staff requested additional information in RAI 4.4-39. In particular, the staff noted that the presence of the chimney partitions above the core may impede thermal hydraulic communication and the radial core outlet pressure distribution may not be uniform. In RAI 4.4-39 the staff requested that an independent methodology be used to establish the validity of this assumption. The analyses provided in response to RAI 4.4-39S2 provide additional assurance that the predicted PANAC11

bundle flow rates are consistent with those flow rates predicted by the more sophisticated TRACG thermal hydraulic model (Ref. 21.6-88.6).

References

21.6-88.1 NEDC-30130P-A, “Steady State Nuclear Methods,” General Electric, April 1985 (ADAMS Accession No. ML070400570).

21.6-88.2 MFN-06-467, Hinds, D., General Electric, Letter to U.S. Nuclear Regulatory Commission, “Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application—Chapter 4 and GNF Topicals—RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89,” November 29, 2006 (ADAMS Accession Nos. ML063450255, ML063450251).

21.6-88.3 MFN-06-467, Supplement 1, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, “Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application—DCD Chapter 4 and GNF Topical Reports—RAI Numbers 21.6-86 S01 and 21.6-88 S01,” March 6, 2007 (ADAMS Accession Nos. ML070720706).

21.6-88.4 MFN-06-467, Supplement 2, Kinsey, J., General Electric, Letter to U.S. Nuclear Regulatory Commission, “Response to Portion of NRC Request for Additional Information Letter No. 66—RAI Number 21.6-88 S02,” June 13, 2007 (ADAMS Accession Nos. ML071930138, ML071930139).

21.6-88.5 MFN-06-467, Supplement 3, Kinsey, J., General Electric Hitachi Nuclear Energy America, LLC, Letter to U.S. Nuclear Regulatory Commission, “Response to Portion of NRC Request for Additional Information Letter No. 66—RAI Number 21.6-88 Supplement 3,” April 15, 2008 (ADAMS Accession Nos. ML081090180, ML081090181).

21.6-88.6 MFN-08-949, Kingston, R., General Electric Hitachi Nuclear Energy America, LLC, Letter to U.S. Nuclear Regulatory Commission, “Response to Portion of NRC Request for Additional Information Letter No. 106—Related to ESBWR Design Certification Application—RAI Number 4.4-39 Supplement 2,” December 15, 2008 (ADAMS Accession Nos. ML083520263, ML083520264).

B.60 RAI 21.6-89

In RAI 21.6-89, the staff requested additional information regarding the detector response kernel model. The response provides details of the CALTIP calculation in PANAC11. The CALTIP is the calculated TIP response. The response provides details of the nodal detector response correlation. The response addresses neutron and gamma TIP response and clarifies the CALTIP and PCTIP comparisons provided in the NEDC-33239P LTR and Reference 21.6-89.1.

However, the staff requested additional information regarding the J-factors themselves and their applicability to the GT instrument. The response to RAI 4.2-12S2-22 provides the details of the GT detector response models and the selection of the J-factor correlation parameters.

Therefore, the staff determined that the response to RAI 4.2-12S2-22 supersedes the response to RAI 21.6-89 (Ref. 21.6-89.2).

References

21.6-89.1 MFN-06-467, Hinds, D., General Electric, Letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application—Chapter 4 and GNF Topicals—RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89," November 29, 2006 (ADAMS Accession Nos. ML063450255, ML063450251).

21.6-89.2 MFN 08-293, Supplement 1, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 106 Related to ESBWR Design Certification Application—RAI Numbers 4.2-12 Supplement 2 and 4.3-2 Supplement 2," July 3, 2008 (ADAMS Accession Nos. ML081930310, ML081930311).

B.61 RAI 21.6-111

In RAI 21.6-111, the staff requested that GEH revise the void reactivity coefficient correction model to account for void history effects in the determination of the void reactivity coefficient biases. GEH has developed the revised model and implemented the model in TRACG04. Reference 21.6-111.1 provides details of the model.

The response provides descriptive details of the implementation of the void history correction model. This model is implemented to account for biases and uncertainties in the TRACG04 void reactivity feedback as calculated by the PANAC11 kinetics engine. The staff evaluated the historical void reactivity coefficient correction in the response to RAI 7 from the staff's review of NEDC-32906P, Supplement 3 and found the correction unacceptable for application to EPU and EPU/maximum extended load line limit analysis plus (MELLLA+) applications because the previous model was based on lattice exposure calculations performed at a single void fraction (40 percent) as discussed in the staff's review of NEDE-32906P, Supplement 3 (Ref. 21.6-111.3).

The revised model is based on comparisons between TGBLA06 and MCNP for various exposure histories and branches to more accurately characterize any biases in the prediction of reactivity feedback for transient calculations. The applicant also updated the database forming the basis for the void reactivity correction to include modern fuel lattices of 10X10 rod arrays.

The staff previously issued RAIs in similar reviews regarding the applicability of the database used to calculate the eigenvalue response surfaces to advanced fuel designs. The response to RAI 30 from the staff's review of NEDC-32906P, Supplement 3 indicates that the TRACG04 revised void reactivity coefficient correction model allows for the flexibility of updating the lattice database via input. Therefore, any license application referencing NEDC-32906P, Supplement 3, that the licensee should confirm that the lattice database is applicable to the specific cases considered or revise the database input to ensure that the database is consistent with the fuel being analyzed.

The basis for the correction model is to perform lattice calculations using TGBLA06. The predicted infinite eigenvalue is compared to eigenvalues predicted using a sophisticated MCNP

code. Based on the calculated eigenvalues, the eigenvalue can be fitted as a function of the void fraction, exposure, and void fraction history.

The staff reviewed the basis for the comparison, noting that a code-to-code comparison is used. The response states, and the staff agrees, that the MCNP qualification is extensive and indicates very small biases and uncertainties, such that there is a high degree of confidence that any uncertainty in the MCNP prediction is sufficiently small that the code-to-code comparison will serve as an acceptable indication of any bias or uncertainty in TGBLA06.

Furthermore, the staff noted that the comparisons were performed for uncontrolled lattices. In its evaluation of the response to RAI 7, the staff concluded that the use of the uncontrolled lattices will bound any uncertainty for similar analyses performed for controlled lattices.

The void reactivity correction model response surface has also been increased to encompass 90 percent void fraction cases. The staff determined that the inclusion of high void cases serves as an improvement in the overall process to more accurately characterize any trends in the biases or uncertainty at these higher void fraction conditions that are more prevalent in EPU and EPU/MELLLA+ cores. The staff reviewed the means for determining the 90 percent void fraction eigenvalues. The eigenvalues are calculated according to extrapolation of the TGBLA06 analytical results at the standard production void fractions. The staff determined that this approach is acceptable and appropriate because it is characteristic of the means by which the TGBLA06 calculations are used in the PANAC11 code. That is, extrapolation errors associated with extrapolation of TGBLA06 parameters in PANAC11 are included in the uncertainties and biases by comparing the extrapolated values against MCNP instead of direct TGBLA06 calculations. The intention of the correction model, the staff noted, is not to characterize the efficacy of the TGBLA06 code, but rather, to normalize the PANAC11 neutronic response to match the more accurate void coefficient predicted by MCNP.

The applicant evaluated the results of the comparisons for modern fuel designs statistically. The staff reviewed the results of these comparisons and determined that the results indicate normality of the uncertainties.

Equation 17 in Reference 21.6-111.5 provides the means by which TRACG implements the correction model. The change in relative water density calculated by the thermal hydraulic solver is normalized according to the void reactivity coefficient ratio produced by the correction model, and the PANAC11-based kinetics solver uses the revised change in nodal relative water density to evaluate the nuclear parameters during the transient. This does not impact the thermal hydraulic calculation, but effectively normalizes the PANAC11-predicted eigenvalue response to changing void conditions to an equivalent change that would have been predicted using a sophisticated transport code.

The void reactivity coefficient ratio is fitted based on the eigenvalue response surfaces that explicitly account for the void history covering a range from 0 percent to 90 percent. The response states that the applicant did not use the 0 percent void fraction cases to develop the fitted function because one lattice code, but not the other, could predict, at low void fractions, a positive void reactivity coefficient. The staff determined that the extrapolation from higher void conditions is acceptable to characterize the general behavior of the void coefficient. The staff concluded that this is acceptable on the basis that, as void fraction increases, the void reactivity coefficient tends to increase in magnitude and become more negative.

Therefore, the correction model at low void conditions is correcting a nodal response that is somewhat insensitive and non-limiting (low void fractions correspond to low power). Generally these nodes do not play a significant role in the transient progression in terms of overall core response.

The staff reviewed the fitting and interpolation schemes for the discrete points in the database to ensure that no errors were introduced due to extrapolation. The staff concluded that these techniques were accurate and therefore acceptable. On the basis of the fitting and interpolation techniques and the range of void fractions covered by the database, the staff determined that the void reactivity coefficient correction model is acceptable to characterize the biases and uncertainties in the void reactivity coefficient in TRACG over a range of instantaneous and exposure-weighted void fractions between 0 percent and 100 percent.

GEH provided a sample calculation demonstrating the effect of the void reactivity correction model. The applicant performed two representative pressurization transient analyses using TRACG04; in one case the void reactivity coefficient correction model was deactivated. The calculations indicate that the change in critical power ratio divided by the change in initial critical power ration (ΔCPR/ICPR) is sensitive to the void reactivity coefficient correction, and the predictions varied by approximately [[ ]] in the maximum ΔCPR/ICPR. The staff determined that [[ ]] is a significant change and agrees with GEH that the new model continue to be applied for anticipated operational occurrence analyses. Transient analyses for licensing applications should be performed with the revised void reactivity coefficient correction model activated.

References

21.6-111.1 MFN-08-504, Kinsey, J., General Electric, letter to U.S. Nuclear Regulatory Commission, “Response to Portion of NRC Request for Additional Information Letter No. 147—Related to ESBWR Design Certification Application—RAI-Number 21.6-111,” June 24, 2008 (ADAMS Accession No. ML081780577).

21.6-111.2 NEDE-32906P, Supplement 3, “Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,” GENE, May 2006 (ADAMS Accession Nos. ML061500186, ML061500184, ML061500185).

21.6-111.3 Safety Evaluation Report by the Office of Nuclear Reactor Regulation for Licensing Topical Report NEDE-32906P, Supplement 3, “Migration to TRACG04/PANAC11 From TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,” December 2008 (ADAMS Accession No. ML082910795)

21.6-111.4 MFN-07-445, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, “Partial Response to Request for Additional Information RE: GE Topical Report NEDE-32906P, Supplement 3, ‘Migration to TRACGO4/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients’ (TAC No. MD2569),” August 15, 2007 (ADAMS Accession Nos. ML072330518, ML072330520).

21.6-111.5 MFN-08-483, Kingston, R., General Electric, letter to U.S. Nuclear Regulatory Commission, “Response to Request for Addition Information (RAI) 30, RE: NEDE-32906P, Supplement 3, ‘Migration to TRACG04/PANAC11 from

TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,' (TAC No. MD2569)," May 30, 2008 (ADAMS Accession Nos. ML081550192, ML081550193)

APPENDIX C

MONTE CARLO N PARTICLE TRANSPORT CODE VALIDATION OF THE TRACG BORON REACTIVITY MODEL FOR THE ECONOMIC SIMPLIFIED BOILING-WATER REACTOR

(JCN Q-4044, Task Order 3)

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EXECUTIVE SUMMARY

This report validates the averaged boron-10 cross-section used in the analysis of transient events that are not terminated in the traditional manner against an independent method. The transient of interest is an anticipated transient without scram (ATWS), particularly the case in which the reactor undergoes a transient event, but the scram system fails. Boron is injected into the downcomer, flows into the inlet plenum, and then enters the core at the bottom, flowing up through the various fuel assemblies. The addition of boron to the coolant in the core essentially scrams the reactor at this stage, and only the decay heat needs to be removed.

GE- Hitachi Nuclear Energy Americas (GEH) used the TRACG code to analyze the above sequence of events and to determine fuel, clad, and coolant temperatures; void fraction; and the variation of boron concentration during the transient. These results indicate that boron does not enter the core until approximately 300 seconds after initiation of the transient. At this stage, the boron concentration increases steadily with time at various heights. In addition, the void fraction varies from operating conditions. As the feedwater is run back, the pressure head driving the core flow is reduced. Power is further reduced due to increasing boron concentration. Subsequently, the core flow rate is reduced, the in-core void fraction increases and power is reduced. The microscopic boron-10 cross-section varies inversely with the neutron velocity. The model used in TRACG is based on this relationship, with suitable modifications to account for deviations from the theoretical model.

As an independent check of this model, the staff carried out a series of Monte Carlo N Particle Transport Code calculations to determine the effective microscopic boron-10 cross-section and the average velocity. The staff carried out these calculations for the appropriate assembly type (depending on axial height), for the void fraction and boron concentrations, and for four burnup levels. This resulted in a total of 60 combinations of assembly type, burnup, void fraction, and boron concentration. Based on the relationships between cross-section and neutron velocity, it is clear that the microscopic cross-section should decrease with increasing average velocity. This decrease should vary inversely with velocity, but it could be modified by non-1/v effects. This dependence might be closer to linear, because any variation of the boron cross-section will be small as compared to the variation in boron number density for this particular transient.

The average cross-section and velocity were determined over the thermal range (less than 0.625 electron volts (eV)) and the total range (0–20.0 million electron volts). For all cases at each height, the thermal range microscopic cross-sections are essentially linearly proportional to the average neutron velocity, regardless of burnup level, boron concentration, or height (which implies assembly type). However, this is not the case for the cross-sections averaged over the entire energy range. In this case, there are four distinct “straight” lines for each burnup level at each height. The correlation is still largely linear, but burnup effects separate the lines. In addition, height (neutron spectral effect) appears to affect the cross-section magnitude. These conclusions are only valid for the conditions encountered in this transient; thus, any possible self-shielding effects resulting from much higher boron concentrations were not explored and might not be encountered in mitigating an ATWS event. It should be pointed out that the fast range (greater than 0.625 eV) microscopic cross-section has essentially no correlation with average neutron velocity. The cross-section values sort themselves into distinct groups (as a function of burnup) with no easily identifiable correlation.

Furthermore, the average macroscopic thermal range cross-section can be determined by multiplying the microscopic cross-section by the boron number density at the time of interest. The variation of the macroscopic cross-section with average velocity for the thermal range shows an increasing cross-section with increasing boron concentration (and time into the transient), regardless of height or burnup. The increase appears essentially linear, with a slightly different slope, depending on burnup.

Finally, the validity of the TRACG boron model based on the equations shown in the first section is largely confirmed, since the microscopic boron cross-section varies as $1/v_{ave}$, regardless of boron concentration, void fraction, and assembly type.

1. INTRODUCTION

The economic simplified boiling-water reactor (ESBWR) design certification documentation submitted by General Electric Hitachi Nuclear Energy America, LLC (GEH) included an analysis of an anticipated transient without scram (ATWS). GEH analyzed this event using the TRACG code, which simulates the coupled thermal hydraulic and three-dimensional neutron kinetic behavior of the reactor core. This simulation involved determination of the boron-10 capture cross-section as a function of void fraction, burnup, and boron concentration. The model is unique to this application.

The objective of this task is to validate the accuracy of the GEH proposed model using completely independent means. The method used in this validation is based on Monte Carlo methods, which differ from the model used in TRACG.

This section describes the transient related to boron injection and outlines the TRACG method and output specific to a particular transient. This section also discusses the need to carry out an independent check of the method of determining the boron cross-section.

1.1 Transient and Related TRACG Output

The transient of interest is an ATWS, particularly the case in which the reactor undergoes a transient event, but the scram system fails. Generally, the main steam isolation valve will close shortly after initiation of the transient, resulting in a sudden increase in the primary system pressure, which collapses the vapor bubbles in the core and subsequently adds a significant amount of reactivity to the core. The sudden increase in reactivity causes a power pulse. Only Doppler feedback can influence the immediate pulse, but in the longer term, feedwater runback can be started to decrease the core inlet subcooling, reduce power, and control any power oscillations. However, boron must eventually be injected into the core to guarantee that it is shut down. Boron is injected into the downcomer, flows into the inlet plenum, and then enters the core at the bottom, flowing up through the various fuel assemblies. The addition of boron to the coolant in the core essentially scrams the reactor at this stage, and only the decay heat needs to be removed.

GEH has analyzed the above sequence of events using TRACG to determine fuel, clad, and coolant temperatures; void fraction; and the variation of boron concentration during the transient. The values of these parameters are determined as a function of height within the core. The results indicate that boron does not enter the core until approximately 300 seconds after initiation of the transient. At this stage, the boron concentration increases steadily with time at various heights. In addition, the void fraction varies from operating conditions. As the feedwater is run back, the pressure increase compresses the void within the core, and the power is reduced significantly at this time. Tables 1 and 2 present the values of boron concentration and void fraction as a function of time and height above the core inlet.

Table 1 Boron Concentration as a Function of Height and Time (kg/m^3)

Assembly type	81902	81902	81905
Height (m)	0.133	0.688	2.51
[[			
			]]

As Table 2 indicates, the void fraction remains fairly constant at any given height above the core inlet.

Table 2 Void Fraction as a Function of Height and Time (kg/m^3)

Assembly type	81902	81902	81905
Height (m)	0.133	0.688	2.51
[[			
			]]

1.2 Boron-10 Cross-Section Model Used in TRACG

The negative reactivity introduced into the core as a result of the boron injection is controlled by the variation of the boron-10 absorption with concentration, void fraction, and, to a lesser extent coolant temperature. The following expression summarizes the model used in TRACG to determine the microscopic cross-section:

[[ ]]

Where:

[[

]]

The macroscopic cross-section is given by:

[[ ]]

As an independent check of this model, it was proposed that a series of Monte Carlo N Particle Transport Code (MCNP) calculations be carried out to determine the effective microscopic boron-10 cross-section and the average velocity. These calculations were carried out for the appropriate assembly type (depending on axial height), for the void fraction and boron concentrations shown above in Tables 1 and 2, and for four burnup levels. A total of 60 combinations of assembly type, burnup, void fraction, and boron concentration resulted. Based on the above relationships, it is clear that the microscopic cross-section should decrease with increasing average velocity. This decrease should vary inversely with velocity, but it could be modified by the importance of the last term in the first equation. The macroscopic cross-section is strongly influenced by the boron number density and should increase as the boron concentration increases. This dependence should be close to linear, since any variation of the boron cross-section will be small as compared to the variation in boron number density.

2. Monte Carlo N Particle Transport Code MODEL

The MCNP assembly models used for this study are based on those created for the study of void fraction feedback. The assembly types of interest are determined by their height above the core inlet. Assembly type 81902 corresponds to the first two heights (0.113 meter and 0.688 meter), and assembly type 81905 corresponds to 2.51 meters. The calculations recognized the following variations:

- fuel burnup and time after transient initiation
- coolant temperature and void
- fuel temperature
- water hole and inter-assembly water temperature
- boron concentration.

Using MCNP, the microscopic cross-section was determined by calculating the boron-10 capture reaction rate and the flux in the cells of interest - thus, dividing the reaction rate by the flux results in the average microscopic cross-section. The following relationship illustrates this procedure:

$$\sigma = \frac{\int \sigma(E).\varphi(E).dE}{\int \varphi(E).dE}$$

Where:

σ = Average cross-section
σ (E) = Energy-dependent cross-section (from ENDF/B file)
φ(E) = Energy-dependent flux

The integrals are carried out over the cell volume of interest and, in this case, the volume corresponds to the following:

- the coolant water surrounding the fuel pins
- coolant in the gap between the fuel pins and the inside of the assembly can
- water in the inter-assembly water gap
- water in the two water holes within the assembly

In addition, the energy integrals are carried out over two ranges: the first over the thermal range (0–0.625 electron volts (eV)) and the second over the entire energy range considered by the code (0–20 million electron volts (MeV)).

To estimate the average velocity, consistent with the average boron cross-section determined above, the following method was used: (1) an artificial cross-section was defined that varies as the velocity varies and was thus proportional to $(E)^{1/2}$, (2) this value was processed through NJOY[34] so that it could be used in MCNP, and (3) the flux averaged reaction rate of this artificial cross-section was determined. Thus—

$$\sigma = \frac{\int \sigma(E).\varphi(E).dE}{\int \varphi(E).dE}$$

Where:

σ = Average cross-section—velocity
σ (E) = Energy-dependent cross-section varies as $(E)^{1/2}$
φ(E) = Energy-dependent flux

The integrals are carried out over both the thermal range and the entire energy range, because it was not clear what range is used in TRACG. The boron cross-section is determined using the same formulation, except the energy-dependent cross-section is obtained from the ENDF/B library file (the same way in which the average cross-sections have been determined up to now).

The assumptions regarding the addition of boron to water in an ESBWR assembly will be outlined. The boron density shown in Table 2 is seen to vary from 0 to a maximum of approximately 0.35 kilogram per cubic meter (kg/m^3), and Table 1 presents the corresponding void fraction. The void fraction information is necessary, since the boron is dissolved in the water and a higher void fraction would imply a lower boron concentration. The following assumptions will be made regarding number densities:

1. The number densities will be estimated using the following equations:

N-Boron = ((density-B) x 0.6022)/(10.811)

[34] NJOY is an industry standard code for performing cross section broadening calculations at various temperatures.

N-Water = (((density-H_2O) x 0.6022)/(18.015)) x (1.0 – (density-B)/(density-H_2O))
Where:
density-B = Boron density (Table 2)
density-H_2O = Water density (consistent with void fraction given in Table 1)

2. Boron is assumed to be present in all the water volumes (i.e., coolant, water holes, and between assemblies).

3. Boron is assumed to be natural (i.e., B^{10} = 19.8 percent, B^{11} = 80.2 percent).

The number densities for the remaining nuclides will be determined in the traditional manner for beginning-of-life conditions. The number densities for the fueled regions corresponding to various burnup levels are determined by MONTEBURNS, which includes the appropriate depletion and build up of transuranic nuclides and fission products. The burnup calculations are carried out assuming no soluble boron and normal operating conditions, which corresponds to conditions before an ATWS event occurs.

For example, Figure 1 illustrates the cross-sectional view of assembly type 81902. This assembly type comprises 14 rods containing gadolinia, 78 rods containing fuel with various enrichments, and 2 large water holes. The assembly box structure and intra-assembly water gap are explicitly represented. The 78 rods containing fuel are divided into 14 burnup regions, and the 14 gadolinia rods are divided into an additional 4 burnup regions, each of which is subdivided into seven “onion skin” radial subregions to recognize the spatial depletion of the gadolinium.

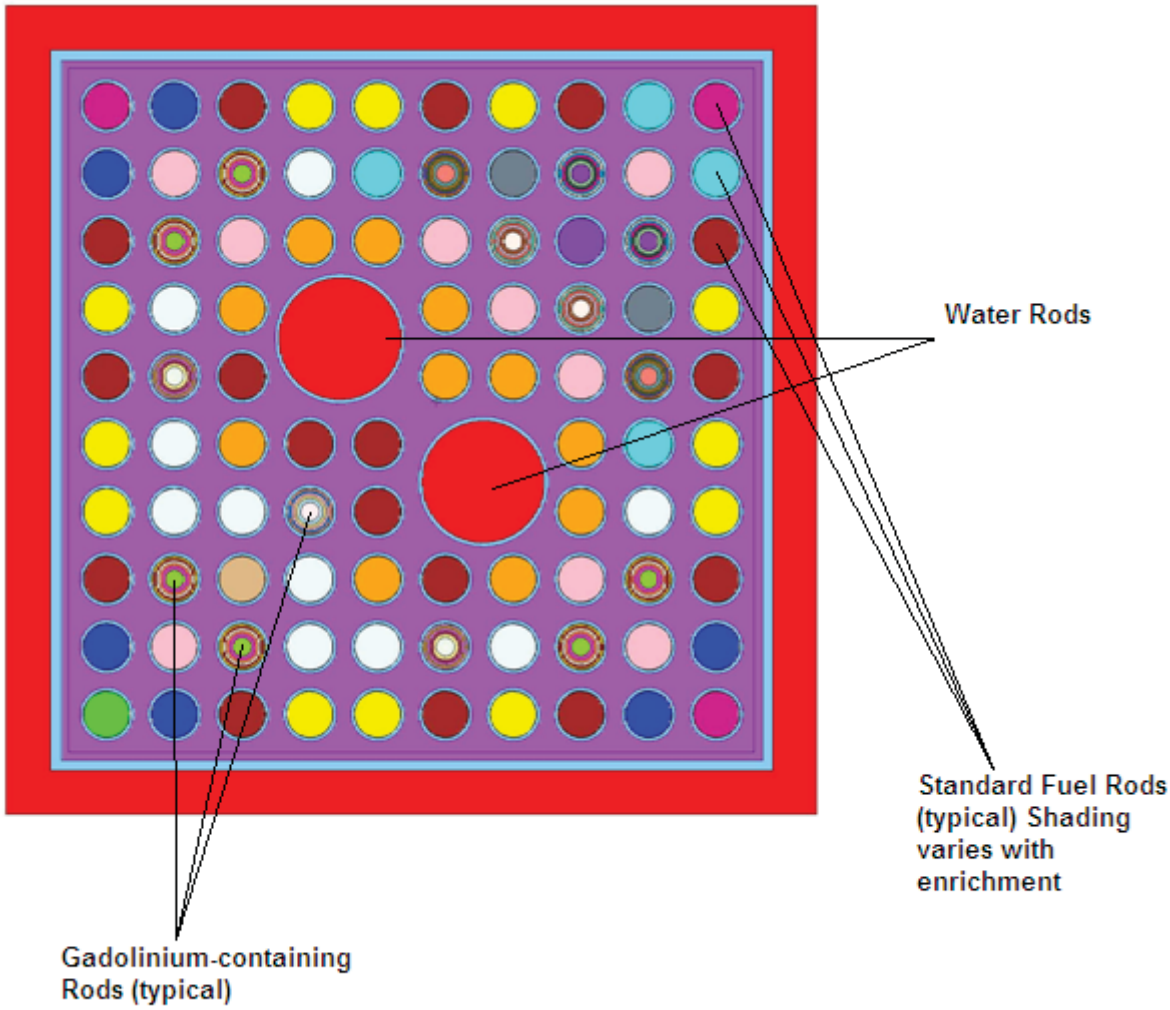

Figure 1 MCNP model for assembly type 81902

The matrix of calculations carried out using MCNP consisted of (1) one for each of three different height values, (2) one for each of the time steps considered (i.e., 400, 500, 600, 700, and 720 seconds), and (3) one for each of four burnup levels (i.e., 0, 20, 40, and 60 gigawatt days per metric ton). This matrix of calculations yields a total of 60 calculations, and in each calculation, the boron cross-section and corresponding average velocity was determined for each of four different water volumes. The volume of primary interest is the coolant volume surrounding the fuel rods. However, the water holes, inner assembly water volume around the edge of the rodded volume, and the intra-assembly water gap were also included separately. These data indicate the potential change in cross-section across the assembly and thus indicate the neutron energy spectral shift within the assembly.

3. RESULTS

This section presents the results of the model calculations in both tabular and graphical form. The first series of results consists of tables of boron-10 cross-section, averaged over the entire

energy range, since all neutrons in a reactor core contribute to the reaction rate. These tables are presented for all of the water volumes described above. These results are followed by tables of cross-sections for the coolant volumes only; in this case, the boron cross-sections are averaged over the thermal range only (up to 0.625 eV). A series of graphical presentations follow, the first of which shows the variation of the microscopic cross-section averaged over the entire energy range, as a function of average velocity averaged over the entire energy range. The second series shows the thermal range microscopic cross-section plotted against the thermal range average velocity. The final graph shows the macroscopic thermal range average cross-section plotted against the thermal range velocity.

The results were determined for five time steps in the ATWS transient, as calculated by TRACG, and for four different fuel burnup rates. This results in 80 MCNP calculations per axial position. Tables 3 through 5 present the boron cross-sections in the coolant volume and the three axial positions. As these tables indicate, the cross-section decreases monotonically with increasing time at constant burnup. In addition, the burnup increases as the cross-section increases for a given time. This indicates that the neutron energy spectrum gets softer with burnup. This phenomenon primarily results from the burning out of the gadolinium in the gadolinia rods and the relative inefficiency of the fission products versus the gadolinium itself in absorbing low energy neutrons. Furthermore, the variation with height at constant burnup and time indicates that the cross-section decreases with increasing axial height and then increases again at the top position. However, this behavior does not apply to the zero burnup case, which shows a decreasing trend. The variation of boron-10 cross-section with time, position, and burnup is clearly a complicated function.

Table 3 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 0.113 m (type 81902): Coolant)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	359.54	352.59	342.20	335.33	333.32
20	401.97	393.31	382.78	373.13	370.33
40	460.59	449.08	435.97	424.59	421.23
60	494.38	481.52	466.71	453.92	450.13

Table 4 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 0.161 m (type 81902): Coolant)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	339.13	330.32	319.60	313.30	311.23
20	380.19	369.39	357.12	349.76	347.06
40	436.12	422.63	407.84	398.66	395.19
60	468.73	453.59	436.87	426.86	423.16

Table 5 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 2.15 m (type 81905): Coolant)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	330.97	327.65	321.49	315.82	313.19
20	401.25	395.98	387.26	379.84	376.31
40	478.81	470.32	458.72	448.63	444.17
60	524.45	513.83	499.99	489.07	483.76

The results in Tables 6 through 14 show the boron-10 cross-section for the remaining water locations mentioned above. It is interesting to note that the cross-section increases in the coolant gap as compared to the coolant cross-section. This indicates a softening of the neutron energy spectrum along the outside edge of the fuel assembly, presumably because of the increased amount of water resulting from the inter-assembly water gap. The cross-section corresponding to the inter-assembly water gap is higher still, indicating a further softening of the neutron energy spectrum. The cross-section for the two water holes is intermediate between that of the coolant and the coolant gap, indicating some softening.

The boron-10 (n,α) cross-section varies as 1/v and is thus a good measure of the neutron spectral hardness or softness. These results indicate that there is a significant neutron energy spectral shift in the assembly of an ESBWR, which must be accounted for when determining a single representative cross-section for any region or volume.

Table 6 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 0.113 m (type 81902): Coolant gap)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	607.41	593.35	576.95	562.61	558.62
20	636.30	621.23	603.34	588.27	583.10
40	687.39	669.24	648.58	631.82	626.71
60	712.01	692.47	671.14	652.94	647.52

Table 7 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 0.113 m (type 81902): Inter-assembly water)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	708.06	692.01	673.67	658.19	653.24
20	733.61	716.14	696.11	680.23	674.42
40	779.20	758.95	736.86	718.47	713.06
60	801.27	779.51	756.84	737.01	731.28

Table 8 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 0.113 m (type 81902): Water hole water (average of two water holes))

GWD/T	400 s	500 s	600 s	700 s	720 s
0	453.39	447.30	436.69	427.70	425.53
20	473.44	464.41	453.87	442.12	439.98
40	515.10	504.16	491.62	478.85	475.90
60	545.58	533.92	518.21	505.22	501.02

Table 9 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 0.161 m (type 81902): Coolant gap)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	577.84	561.39	542.64	530.74	527.03
20	606.75	588.17	567.87	555.94	551.06
40	654.63	634.62	611.79	597.35	592.18
60	679.67	657.37	633.28	618.33	613.02

Table 10 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 0.161 m (type 81902): Inter-assembly water)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	680.62	661.59	641.48	628.02	624.09
20	705.78	685.04	663.07	649.90	644.68
40	748.70	726.34	701.85	686.36	680.80
60	771.01	746.64	720.92	704.95	698.91

Table 11 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 0.161 m (type 81902): Water hole water (average of two water holes))

GWD/T	400 s	500 s	600 s	700 s	720 s
0	425.35	415.58	403.16	396.94	393.36
20	444.78	434.71	420.47	412.78	409.92
40	485.58	471.00	457.79	447.80	444.56
60	515.43	500.62	483.85	473.51	470.02

Table 12 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 2.15 m (type 81905): Coolant gap)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	514.32	508.10	497.48	488.08	483.92
20	569.62	561.46	548.50	537.64	532.68
40	639.88	628.08	612.07	598.95	593.28
60	675.60	661.41	644.79	630.05	623.50

Table 13 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 2.15 m (type 81905): Inter-assembly water)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	617.72	609.28	596.73	586.30	581.36
20	668.44	658.03	643.36	630.82	625.88
40	731.94	718.08	700.56	686.02	680.05
60	764.11	747.93	729.57	713.65	706.79

Table 14 Variation of Boron-10 Microscopic Cross-Section in Barns with Time and Burnup (Axial position = 2.15 m (type 81905): Water hole water (average of two water holes))

GWD/T	400 s	500 s	600 s	700 s	720 s
0	425.10	421.31	414.42	408.36	404.88
20	472.61	466.96	458.11	450.30	445.97
40	535.65	526.65	515.01	504.46	499.95
60	578.86	568.38	553.75	543.35	537.88

Tables 15 through 17 present the thermal range cross-sections for the coolant volumes. As can be seen, the thermal range cross-sections are significantly higher than those averaged over the entire energy range, indicating a significant contribution to the reaction rate with boron in this particular core from neutrons above the thermal range.

Table 15 Variation of Boron-10 Microscopic Thermal Cross-Section in Barns with Time and Burnup (Axial position = 0.113 m (type 81902): Coolant)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	2070.0	2060.0	2050.0	2050.0	2040.0
20	2120.0	2110.0	2110.0	2100.0	2100.0
40	2180.0	2170.0	2160.0	2160.0	2150.0
60	2200.0	2190.0	2190.0	2180.0	2180.0

Table 16 Variation of Boron-10 Microscopic Thermal Cross-Section in Barns with Time and Burnup (Axial position = 0.161 m (type 81902): Coolant)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	2050.0	2040.0	2030.0	2030.0	2020.0
20	2110.0	2100.0	2090.0	2080.0	2080.0
40	2170.0	2160.0	2150.0	2140.0	2140.0
60	2190.0	2180.0	2170.0	2170.0	2160.0

Table 17 Variation of Boron-10 Microscopic Thermal Cross-Section in Barns with Time and Burnup (Axial position = 2.15 m (type 81905): Coolant)

GWD/T	400 s	500 s	600 s	700 s	720 s
0	2040.0	2040.0	2030.0	2030.0	2020.0
20	2120.0	2110.0	2100.0	2100.0	2100.0
40	2180.0	2180.0	2170.0	2160.0	2160.0
60	2210.0	2200.0	2200.0	2190.0	2190.0

The average thermal range capture cross-section varies in the same manner as the cross-section averaged over the entire energy range. Briefly, this variation results from the softening of the neutron spectrum with burnup (i.e., as the gadolinium burns out) and the hardening of the neutron spectrum with height above the core inlet due to the increase in void fraction.

Figures 2 through 4 illustrate the variation of the thermal range microscopic cross-section, total range microscopic cross-section, and thermal range macroscopic cross-section with the respective average neutron velocity for the three heights. For all cases, at each height, the thermal range microscopic cross-sections are essentially linearly proportional to the average neutron velocity, regardless of burnup level, boron concentration, or height (which implies assembly type). This is not the case for the cross-sections averaged over the entire energy range. In this case, four distinct “straight” lines emerge for each burnup level at each height. The correlation is still largely linear, but there are burnup effects that separate the lines. In addition, there is a height effect (neutron spectral effect) in the cross-section magnitude. It should be pointed out that the fast range (above 0.625 eV) microscopic cross-section has essentially no correlation with average neutron velocity. The cross-section values sort themselves into distinct groups (as a function of burnup) with no easily identifiable correlation.

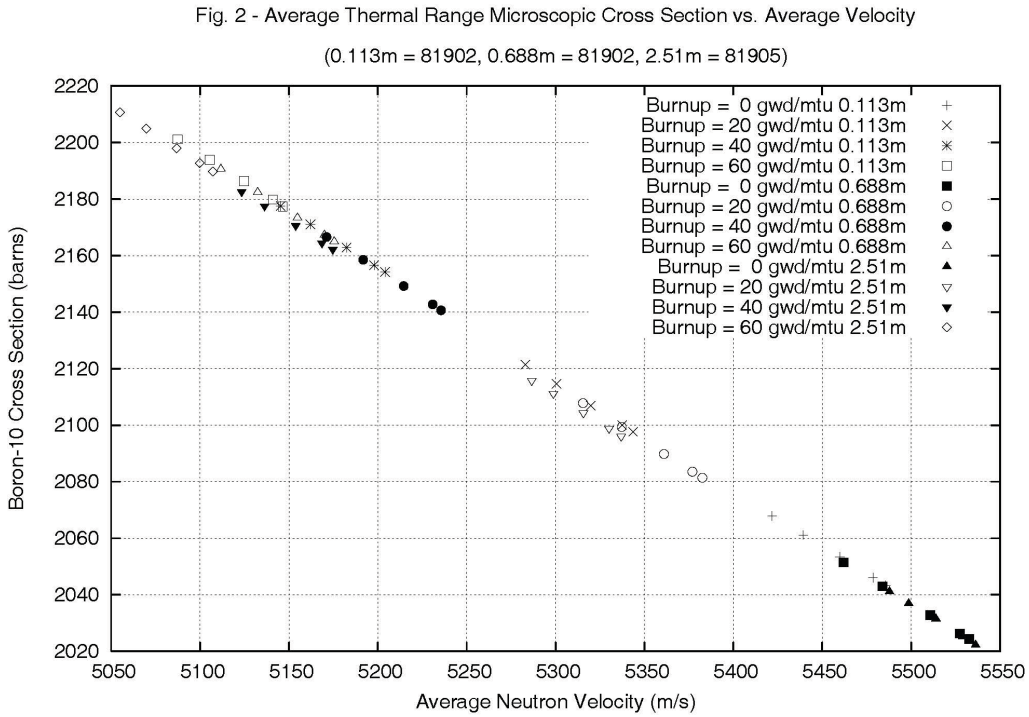

Figure 2 Average thermal range microscopic cross-section versus average velocity

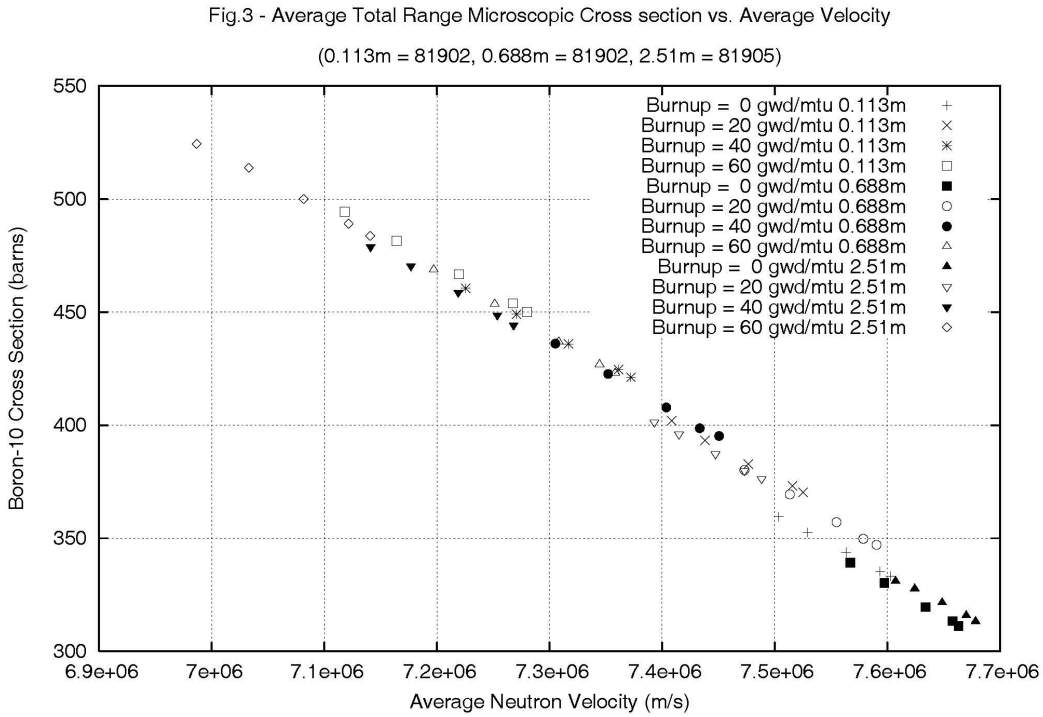

Figure 3 Average total range microscopic cross-section versus average velocity

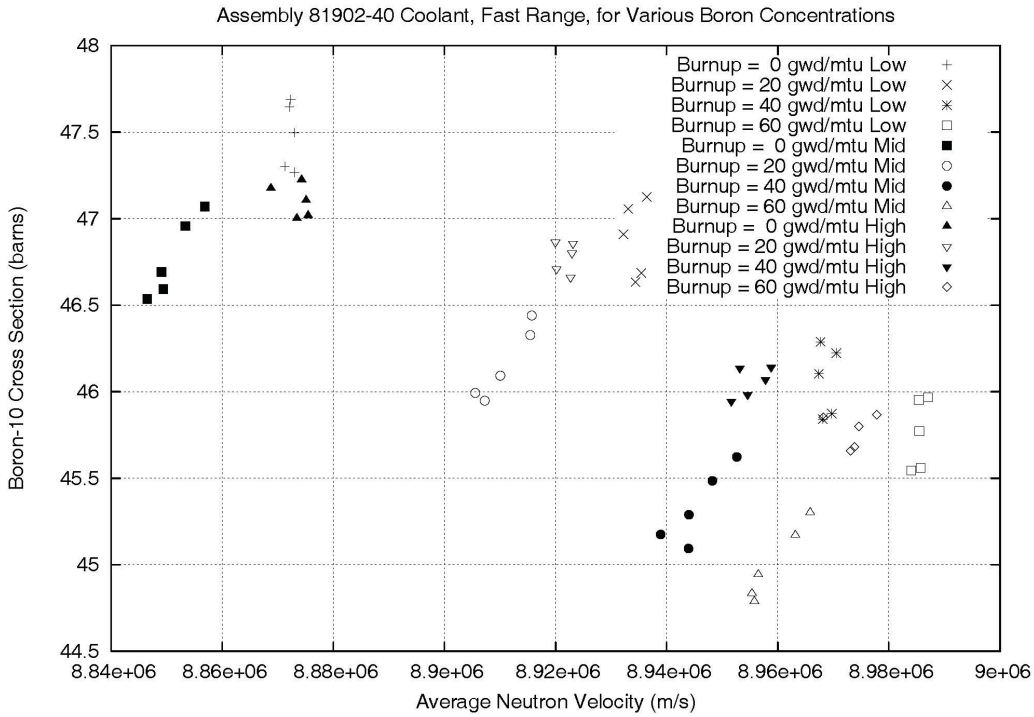

Figure 4 Assembly type 81902-40 coolant, fast range at various boron concentrations

Finally, the average macroscopic thermal range cross-section can be determined by multiplying the microscopic cross-section by the boron number density at the time of interest. The variation of the macroscopic cross-section with average velocity for the thermal range suggests an increasing cross-section with increasing boron concentration (and time into the transient), regardless of height or burnup. The increase appears to be essentially linear, with a slightly different slope depending on burnup.

4. CONCLUSIONS

The validity of the TRACG boron model based on the equations shown in Section 1.2 is largely confirmed, since the microscopic boron cross-section varies as $1/v_{ave}$ regardless of boron concentration, void fraction, and assembly type. This conclusion is only valid for the conditions encountered in this transient; thus, any possible self-shielding effects resulting from much higher boron concentrations were not explored and might not be encountered in mitigating an ATWS event. There is a slight dependence on burnup for the cross-sections averaged over the entire energy range of interest (i.e., 0–20 MeV).

APPENDIX D
APPENDIX B ABBREVIATIONS AND ACRONYMS

Abbreviation	Definition
3D MONICORE	core monitoring software
[[ ]]	[[ ]]
[[ ]]	[[ ]]
ABWR	advanced boiling-water reactor
ADAMS	Agencywide Document Access and Management System
AFIP	automated fixed in-core probe
AOO	anticipated operational occurrence
APS	axial power shape
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
ΔCPR/ICPR	change in critical power ratio divided by initial critical power ratio
CFR	*Code of Federal Regulations*
CLR	condition, limitation, or restriction
CMS	core monitoring system
COLR	core operating limits report
CPR	critical power ratio
CPRRAT	critical power ratio ratio
CRDA	control rod drop accident
DCD	design control document
DOM	dominant
EPU	extended power uprate
eV	electron volt
ESBWR	economic simplified boiling-water reactor
FMCRD	fine motion control rod drive
GDC	general design criterion or criteria
GE	General Electric
GEH	General Electric Hitachi Nuclear Energy America, LLC
GENE	General Electric Nuclear Energy
GNF	Global Nuclear Fuel
GT	gamma thermometer
GT LTR	Gamma Thermometer Licensing Topical Report (NEDE-33197P)
GWD/T or GWD/mT	gigawatt-day per metric tonne
GWD/ST	gigawatt-day per short ton
IC LTR	Initial Core Licensing Topical Report (NEDC-33326P)

Abbreviation	Definition
IE	infrequent event
IMLTR	Interim Methods Licensing Topical Report (NEDC-33173P)
K5	Kashiwazaki-Kariwa Unit 5
kW/ft	kilowatt per foot
kW/l	kilowatt per liter
kW/m	kilowatt per meter
LHGR	linear heat generation rate
LPRM	local power range monitor
LTR	licensing topical report
M+LTR	MELLLA+ Licensing Topical Report (NEDC-33006P)
MCNP	Monte Carlo N Particle Transport Code
MELLLA+	maximum extended load line limit analysis plus
MFLPD	maximum fraction of limiting power density
Migration LTR	Migration Licensing Topical Report (NEDE-32906P, Supplement 3)
MLHGR	maximum linear heat generation rate
MWe	megawatt electric
MWt	megawatt thermal
NRC	U.S. Nuclear Regulatory Commission
OLMCPR	operating limit minimum critical power ratio
PANAC10	earlier version of PANAC11
PANAC11	Global Nuclear Fuel's three-dimension core simulator code
PCGEN	TIP/GT Response Model
ppm	parts per million
RAI	request for additional information
RMS	root-mean-square
SAFDL	specified acceptable fuel design limit
SDM	shutdown margin
SE	safety evaluation
SER	safety evaluation report
SLMCPR	safety limit minimum critical power ratio
SP0	nominal operating statepoint in the feedwater temperature/power operating domain
SRP	Standard Review Plan
T-M	Thermal Mechanical
TGBLA04	earlier version of TGBLA06
TGBLA06	Toshiba-General Electric boiling lattice analysis code
TIP	traversing in-core probe
TRACG	Transient Reactor Analysis Code developed by General Electric Hitachi

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Mark J. Colby, state as follows:

(1) I am the Manager, New Plants Engineering, GE Hitachi Nuclear Energy ("GEH"), have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 of GEH letter MFN 10-301, Mr. Richard E. Kingston to U.S. Nuclear Regulatory Commission, entitled "*Transmittal of GEH Licensing Topical Reports NEDE-33197P-A, Revision 3, "Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring," and NEDO-33197-A, Revision 3, "Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring," dated October 5, 2010.* The GEH proprietary information in Enclosure 1, is considered GEH Proprietary Information and is delineated by a [[dotted underline inside double square brackets.{3}]]. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret," within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH competitors without license from GEH constitutes a competitive economic advantage over other companies;
b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information, which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a., and (4)b, above.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements, which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it identifies detailed GEH ESBWR design information. GEH utilized prior design information and experience from its fleet with significant resource allocation in developing the system over several years at a substantial cost.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 6th day of October, 2010.

Mark J. Colby

Mark J. Colby
GE-Hitachi Nuclear Energy Americas LLC