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Interim Methods LTR Supplement
for
Hope Creek
Extended Power Uprate

Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

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Interim Methods LTR Supplement for Hope Creek Extended Power Uprate

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1.0 Introduction and Summary

GE's NRC-approved neutronic and thermal-hydraulic methods and code systems were utilized to support the Hope Creek Generating Station (HCGS) Extended Power Uprate (EPU) application. Table 1-2 of the Hope Creek Generating Station (HCGS) Extended Power Uprate (EPU) application [1] lists all the nuclear steam system codes used for the EPU request. Section 1.2.2 of Reference [1], "Computer Codes," indicates that the HCGS application of these codes complies with the limitations, restrictions, and conditions specified in the applicable NRC safety evaluation report (SER) that approved each code, with exceptions as noted in Table 1-2 of Reference [1].

In addition, on February 10, 2006, GE submitted, and the NRC has since been reviewing, a Licensing Topical Report (LTR) intended to address NRC questions regarding the application of GE's nuclear core and physics analytical methods to expanded operating ranges, known as the Interim Methods LTR (References [10] and [11]). The scope of the LTR includes EPU and is based upon the approach taken during the NRC approval of the Vermont Yankee license application for a constant pressure power uprate. This LTR has also been reviewed by NRC staff for TVA's Browns Ferry Nuclear Plant extended power uprate (EPU) license amendment request. GE has also submitted to the NRC responses to NRC Requests for Additional Information (RAIs) related to generic application of the Interim Methods LTR to extended power uprate via Reference [11]. GE's Interim Methods LTR and the associated responses to NRC staff RAIs are applicable also to the implementation of EPU at Hope Creek Generating Station.

GE's neutronic and thermal-hydraulic methods and code systems were submitted to the NRC with databases of performance demonstrations that represented the plants and operations of the BWR fleet. The performance demonstration databases associated with each NRC submittal were meant to establish the applicability ranges for the coupled sets of methods. The review and approval of these methods was, in whole or in part, based on the performance demonstration given in those submittals. Although most NRC SERs on the methods and code systems do not specify applicability ranges, the performance demonstration databases were utilized by GE to ensure the application of the codes and methodologies was consistent with the previously submitted performance demonstration databases. In order to maintain this bases for the approved range of applicability, GE provides periodic performance updates to the NRC to establish evidence that the methods and code systems continue to support actual performance within the limitations, restrictions, and conditions specified in the applicable NRC SERs. The current applicability of the methods and code systems to HCGS EPU conditions is also based on such a performance demonstration (i.e., the aforementioned Interim Methods LTR). In this report, four BWRs that have operated at high power density have been chosen for a performance comparison with the HCGS EPU core, and they will be referred to as the *Reference BWRs*. The use of high power density BWRs in this report will validate the applicability of NRC approved GE methods and code systems for the HCGS EPU application as described in the Interim Methods LTR.

The performance demonstration described in Section 2.0 in conjunction with the information provided in Sections 3.0, 4.0, and 5.0 will be used to address the following 4 topics:

(a) The steady state and transient neutronic and thermal-hydraulic analytical methods and code systems used to perform the safety analyses supporting the EPU conditions are being applied within the NRC-approved applicability ranges.

(b) For EPU conditions, the calculational and measurement uncertainties applied to the thermal limits analyses are valid for the predicted neutronic and thermal-hydraulic core and fuel conditions.

(c) The assessment database and the assessed uncertainty of models used in all licensing codes that interface with and/or are used to simulate the response of HCGS during steady state, transient or accident conditions remain valid and applicable for the EPU conditions.

(d) Application of GE's methods to the non-GE legacy fuel remaining in the HCGS EPU operating cycle.

Topic area (a) is addressed in Sections 2.0 and 3.0. Section 2.0 defines the key methods parameters (metrics), provides the steady state performance data base for the metrics based on the Reference BWRs, and identifies that the HCGS EPU expected values for these metrics are consistent with the range of the performance data base. Section 2.0 predominantly provides bases for the steady state applicability ranges. Transient methods and code systems applicability is discussed in Section 3.0.

Topic area (b) is addressed in Section 4.0. Section 4.0 confirms the calculational and measurement uncertainties applied to the thermal limits derived from actual operation at high power density in the Reference BWRs are within the ranges specified in the pertinent methods Licensing Topical Reports and are applicable to HCGS in the EPU conditions.

Topic area (c) has been addressed based on the information in Sections 2.0 through 5.0. The assessment database and the assessed uncertainty of models are valid and applicable to HCGS in the EPU condition. Expected HCGS EPU conditions have been shown to be consistent with the steady state and transient applicability of the methods and code systems as demonstrated in Sections 2.0 and 3.0, and the uncertainties associated with the methods and code systems have been shown to be valid in Section 4.0. The assessment in Section 5.0 concludes that the methods and code systems are applicable to the HCGS EPU core with the remaining non-GE legacy fuel based on the information in Sections 2.0, 3.0 and 4.0.

Topic area (d) is addressed in Section 5.0.

The result of the assessments presented in this Supplement supports the conclusion that HCGS can apply NRC approved GE methods and code systems to support safe operation of HCGS in the EPU condition.

2.0 Methods and Codes Range of Applications

NRC-approved or industry-accepted computer codes and calculational techniques have been applied for the power uprate analyses for HCGS: TGBLA, PANACEA, ISCOR, ODYN, TASC, SAFER, and GESTR. The application of each of these codes complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code.

Section 2.1 describes the plant specific metrics that will be used to evaluate the impact of the power uprate condition on methods and code systems applicability for HCGS.

2.1 Metrics

The overall approach showing the codes and methods performance is to show that existing BWR operating experience covers the ranges of key methods parameters that are expected to be encountered during EPU operation at HCGS. Seven specific metrics have been identified for the purpose of evaluating the impact of power uprate conditions at HCGS. These metrics are:

1. Maximum Bundle Power
2. Maximum Bundle Power/Flow Ratio
3. Exit Void Fraction of Maximum Power Bundle
4. Maximum Channel Exit Void Fraction
5. Core Average Exit Void Fraction
6. Peak Linear Heat Generation Rate
7. Peak End of Cycle Nodal Exposure

Safe BWR core operation is assured by operating within the Tech Spec limits on LHGR, MAPLHGR, MCPR, Hot Reactivity (Reactivity Anomaly), and Cold Reactivity (Shutdown Margin Demonstration). The seven specific metrics listed above are intermediate, dependent variables for which no specific limitation is implied, or necessary, except for peak linear heat generation rate. The purpose of these metrics is to define the parameters that can be used to determine the applicability of the methods and code systems to EPU, and they are not intended to be monitored during plant operation.

Values for these metrics have been obtained from the actual operation of the four Reference BWRs that have operated at high power density under power uprate conditions. The four Reference BWRs all represent high power density applications and 10x10 fuel. These four Reference BWRs are also included in the evaluations performed to demonstrate the applicability of GE methods in the EPU operating domain in the Interim Methods LTR. Key characteristics for these plants in comparison to HCGS are shown in Table 2.1.

HCGS has the largest core size compared to the Reference BWRs. Plants B, C, and D represent extremes of core size and operating strategies. Plant B is a BWR/6 uprated 5% to a power density of [[ ]] kW/ℓ. Plant C is a BWR/6 uprated 20% to a power density of [[ ]] kW/ℓ. Plant B has a large core operating on a two-year cycle with a large reload batch fraction, giving flexibility to achieve a flat power distribution. Plant C has a smaller core operating on a

two-year cycle with a large reload batch fraction, with operation ranging between 10-15% uprated power. Plant D has a lower overall power density, but is a very small core operating on an annual cycle with a smaller reload batch fraction. Efficient design for such a small core requires high radial peaking factors. Hence, its peak bundle powers are as high or higher than plants with significantly higher average power densities.

Table 2.1 – HCGS Compared to the Reference BWRs

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Expected values of these metrics for operation of HCGS at uprated conditions have been calculated. The current licensed thermal power (CLTP) case is based on the HCGS Cycle 13 which completed operation in April 2006. The 115% HCGS EPU case is representative of bundle and core designs that HCGS could actually specify and operate during Cycle 15 and therefore provides a reasonable representation of expected power shapes and core void content.

Sections 2.1.1 through 2.1.7 discuss each of the seven metrics, and show the comparison of the HCGS expected values for that metric versus the values obtained from the Reference BWRs. These comparisons are made by plotting the metric versus cycle exposure. Cycle exposure is not a key parameter with respect to each metric, but merely provides a convenient means of presenting the data in graphical form.

These comparisons show that the expected values of these metrics for HCGS are within the range of values obtained from operation of the Reference BWRs. While a more exhaustive survey of operating BWRs would probably yield higher values for these metrics, the data from the Reference BWRs is sufficient to demonstrate that HCGS at EPU conditions operates within the current operational experience range. In addition, metrics provided for the SVEA 96+ legacy

fuel operating in the HCGS EPU core demonstrate that the legacy fuel essentially operates at pre-EPU conditions. This is a result of the legacy fuel being higher exposure and low reactivity in its fourth or fifth operating cycle. Therefore, since all fuel in the HCGS EPU core is operating consistent with the current operational experience, the Interim Methods Report is applicable to the HCGS EPU core.

2.1.1 Maximum Bundle Power

The bundle power (MW) is a fundamental, direct input to the critical power ratio safety parameter calculation, the linear heat generation rate, the initial conditions for LOCA response, and the calculation of other intermediate quantities. The maximum bundle power represents a local metric of operating conditions and is relevant to the performance of the steady-state nuclear methods. Figure 2.1 shows a comparison between the expected values of maximum bundle power for HCGS and the maximum bundle power from eight cycles of the four Reference BWRs. The SVEA 96+ legacy fuel is shown to be consistent with the CLTP values and thus is effectively operating at pre-EPU conditions.

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Figure 2.1 – Maximum Bundle Power for HCGS Compared to Reference BWR Operational Experience

2.1.2 Maximum Bundle Power/Flow Ratio

High power bundles have high internal voiding and increased two-phase pressure drop. This generally results in reduced flow in the high power bundles relative to lower power bundles. Since bundle power and flow are direct inputs to the calculation of the critical power ratio safety parameter, as well as other intermediate quantities, the maximum bundle Power/Flow Ratio in the core is a meaningful metric. This is shown in Figure 2.2 for expected HCGS operation and for the Reference BWRs. The maximum bundle Power/Flow ratio usually occurs in the maximum power bundle. The SVEA 96+ legacy fuel is shown to be consistent with the CLTP values and thus is effectively operating at pre-EPU conditions.

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Figure 2.2 – Maximum Bundle Power/Flow Ratio for HCGS Compared to Reference BWR Operational Experience

2.1.3 Exit Void Fraction of Maximum Power Bundle

The void fraction results from the integration of the bundle power and flow, as well as the axial distribution of power deposition along the bundle. The SVEA 96+ legacy fuel is shown to be consistent with the CLTP values and thus is effectively operating at pre-EPU conditions.

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Figure 2.3 – Exit Void Fraction of Maximum Power Bundle for HCGS Compared to Reference BWR Operational Experience

2.1.4 Maximum Channel Exit Void Fraction

The maximum power bundle (hot channel) may not always coincide with the bundle with the highest channel exit void fraction, since this parameter is based not only on total bundle power, but also on bundle flow and axial power shape within the bundle. A comparison between the maximum channel exit void fraction for the HCGS cases and the Reference BWRs is shown in Figure 2.4 as a function of cycle exposure. The SVEA 96+ legacy fuel is shown to be consistent with the CLTP values and thus is effectively operating at pre-EPU conditions.

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Figure 2.4 – Maximum Channel Exit Void Fraction for HCGS Compared to Reference BWR Operational Experience

2.1.5 Core Average Exit Void Fraction

The core average exit void fraction is a core wide metric on the amount of heat being carried by the coolant. The combination of high core power and reduced core flow will increase the core average exit void fraction. A comparison of the core average exit void fraction between the operation of the Reference BWRs and expected values for HCGS are shown in Figure 2.5.

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Figure 2.5 – Core Average Exit Void Fraction for HCGS Compared to Reference BWR Operational Experience

2.1.6 Peak Linear Heat Generation Rate

The peak linear heat generation rate LHGR (kW/ft) is a reasonable gauge of degree of peaking in the core since it is comprised of the combination of radial, axial, and local (pin) power peaking. It is also a key design constraint and monitoring parameter. The licensing limits for LHGR for GE fuel are either 13.4 or 14.4 kW/ft, depending on the product line. For GE 10x10 fuel, the limit is 13.4 kW/ft. The SVEA 96+ legacy fuel is shown to be consistent with the CLTP values and thus is effectively operating at pre-EPU conditions.

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Figure 2.6 – Peak LHGR for HCGS Compared to Reference BWR Operational Experience

2.1.7 Peak End of Cycle Nodal Exposure

The nodal and pellet exposures are determined by integration of the energy produced from the local physical area of the fuel divided by its original specific mass. The peak pellet exposure is a license limit for GE fuel. For highly exposed fuel, the peak pellet exposure is typically ~10% higher than the peak nodal exposure. Because it is more relevant to the 3D simulator accuracy, and because the relationship between peak pellet and nodal exposure is weakly dependent on fuel type, nodal exposure has been chosen as the metric for peak local exposure. Table 2.2 shows a comparison between the peak end of cycle nodal exposure for the four Reference BWRs and two examples of projected HCGS operation, one with CLTP and another with 115% EPU.

Table 2.2 – Peak End of Cycle Nodal Exposure for HCGS Compared to Reference BWR Operational Experience

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2.2 Range of Experience and Predicted Plant Conditions

The ranges of operating experience from the metric analysis of the Reference BWRs are summarized in Table 2.3. Inspection of Tables 2.2, 2.4 and 2.5 and Figures 2.1 through 2.6, further demonstrates the fact that HCGS 115% EPU projected operation is consistent with current operational experience range.

Table 2.3 – Ranges of Operational Experience

Metric	Range of Experience	HCGS 115% EPU Prediction[1]	Within Experience (Y/N)
Max. Bundle Power (MW)	7.58	7.07	Y
Max. Power/Flow Ratio (MW/(lb/hr x1.0E-04))	0.89	0.74	Y
Exit Void Fraction of Max. Power Bundle	0.90	0.87	Y
Max. Channel Exit Void Fraction	0.90	0.87	Y
Core Avg. Exit Void Fraction	0.77	0.76	Y
Peak LHGR (kW/ft)	13.4 (10x10 fuel)	11.99	Y
Peak Nodal Exposure (GWd/ST)	58.8	55.98	Y

[1] As stated earlier, the purpose of these metrics is to define the parameters that can be used to determine the applicability of the methods and code systems to EPU, and they are not intended to be monitored during plant operation.

Table 2.4 – Metric Summary for HCGS Cycle 13 (CLTP)

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Table 2.5 – Metric Summary for HCGS 115% EPU

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Table 2.6 – Metric Summary for HCGS 115% EPU SVEA

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3.0 Applicability to Transient Methods

The transient methods discussed in this section are classified into two broad areas:

a. Transient events
b. LOCA

The transient area in this case involves the events affecting the core. These are the pressurization events, depressurization and flow decrease events, and reactivity events.

The reactivity events are analyzed with the steady state tools and the results presented in Sections 2.0 and 4.0 are directly applicable. There are some increases in power, which are significant but remain within the comparisons between the above plants for corresponding events.

The pressurization events result in higher pressures and a momentary increase in core flow, for HCGS, which reduces the void fractions and increases the power generation. The Peach Bottom tests are used as a basis for the transient model validation for the limiting pressurization events. The model bias and uncertainties have been defined for these events and are applied for the transient analysis. The core conditions are bounded by the GEXL and void-quality correlations for these events. Further still, the GEXL correlation has been qualified through full scale thermal-hydraulic testing for transient conditions by simulation of the limiting pressurization events. These response conditions keep the core within the experience base of the other plants indicated in the examples for their transient conditions.

The depressurization and flow reduction events are not the limiting events and result in a reduction of power, which is the reason they are not limiting. The reduction in power results in a decrease in void fraction. When flow decreases occur such as a pump trip, there is still a reduction in power and no challenge to the thermal limits due to the reduction in power. By the time the flow reaches levels low enough to be of concern for meeting the correlation ranges, there is no significant thermal limit impact and analysis is terminated at or before that point.

LOCA events start from the outset with a sharp drop in reactor power (reactor trip occurs almost immediately), core flow and pressure. The post-LOCA thermal hydraulic conditions are within the qualification basis of the SAFER-GESTR code models. These models have been qualified against data obtained in numerous small and full-scale experiments and tests. The GEXL and void-quality correlation are applied in the TASC code to determine dryout times. The post-LOCA thermal hydraulic conditions are within the qualification basis for these models. There is no significant dependence upon the steady-state correlations of the void-quality correlation for the limiting calculations. The GEXL and void-quality correlation applications are within the range of application.

Additional discussion of the applicability of the transient and LOCA methods for EPU conditions is provided in the Interim Methods LTR.

4.0 Core Tracking Applicability Range: Confirmation of Calculational and Measurement Uncertainties

Because the phenomenological conditions experienced for HCGS at the power uprated conditions do not exceed the operational range of experience already in hand, the validation of calculational and measurement uncertainties simply reduces to confirmation of those uncertainties for the same four Reference BWRs that form the operational experience. The most reliable set of data with which to compare are the hot eigenvalue tracking from core follow (exposure accounting of critical core through the operating cycle), cold critical analysis (from operational cold shutdown margin demonstration), and comparisons between measured and calculated instrument (TIP) responses.

4.1 Hot Critical Eigenvalues

In plant tracking calculations, the 3D simulator (using cross sections generated from the lattice physics model, thermal-hydraulic models, and all other model assumptions) is used to simulate the behavior of a plant during operation. The reactor power, total core flow, pressure, inlet conditions, and reactivity control inventory as a function of cycle exposure are all involved in the prediction of the critical state of the core. The actual operating reactor is critical; hence, the calculated k_{eff} is compared to 1.0. Accurate and technically well-founded simulators calculate a k_{eff} that is predictable and should not vary appreciably from cycle-to-cycle for any particular plant. Any trends in fuel exposure should be small and reproducible. Consistency of k_{eff} bias ensures that accurate cycle length estimates are obtained in future core designs.

Figure 4.1 demonstrates the consistency of eigenvalue behavior for the Reference BWRs. It also shows actual data for HCGS Cycle 13 (CLTP) and design target for the 115% EPU cycle. Globally, the trend is linear with the trend line indicated by "Linear (AMDA)". The deviation (one standard deviation) about this trend is approximately [[ ]], which is substantially the same as reported in Reference [5]. The local dispersion in the data reflects substantial deviations from hot, full-power steady-state conditions and do not affect the ability of the plant to correctly determine end of full power capability.

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Figure 4.1 – Reference BWR Operational Experience for Hot Critical Eigenvalue Behavior

4.2 Cold Critical Eigenvalues

BWRs are designed so that they can be shut down in the cold condition (68°F) with the single strongest control blade completely withdrawn. In order to qualify the 3D simulator to accurately predict the cold shutdown margin, cold critical startup configurations are analyzed. In all cases, enough control blades were withdrawn at a given water temperature for the reactor to be critical or on a large positive period. Accurate and technically well-founded simulators calculate a k_{eff} that is predictable and should not vary appreciably from cycle-to-cycle for any particular plant. Any trends in fuel exposure should be small and reproducible. Consistency of k_{eff} bias ensures that accurate shutdown margin estimates will be obtained in future core designs.

Figure 4.2 shows cold critical eigenvalue data for the Reference BWRs. For convenience, a

typical fleet average cold critical eigenvalue is plotted, showing the typical reduction in eigenvalue as the core depletes. This trend is very consistent with the eigenvalue trend used in the development of the nuclear design basis in the GE reload licensing process. Although the startup sequence, configurations and conditions may vary considerably between plants, the data do not show greater than expected dispersion of the critical eigenvalues. These data also contain some local critical configurations. The RMS difference between predicted nuclear design basis based on past experience and the actual measured critical eigenvalue is [[ ]], which is slightly smaller than the [[ ]] reported in Reference [5]. In addition the RMS difference for the actual measured critical eigenvalues for Hope Creek Cycles 13 and 14 is [[ ]].

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Figure 4.2 – Reference BWR Operational Experience for Cold Critical Eigenvalue Behavior

4.3 Traversing In-Core Probe (TIP) Comparison

Three-dimensional power shape information as recorded by the TIP instrument readings can be

compared to calculated instrument readings from the simulator to determine its ability to calculate power distributions. Strings of either thermal neutron sensitive detectors (often referred to as thermal TIPs) or gamma-ray sensitive detectors (referred to as gamma TIPs) may be used to assess the normalized axial power shape along almost the entire length of the bundles within a four-bundle cell. The integrated signals may be combined to evaluate the radial power distribution within the core.

The 3D simulator models the response of the instrument to the appropriate particle species at the detector location to produce a simulated signal. For TIP comparisons, this simulated detector response is compared to the relative strength of the measured signal. TIP distributions take time to accumulate and hence are obtained periodically throughout an operating cycle. The most common interval between TIP measurements is several weeks. During the time between TIP measurements, the Local Power Range Monitors are used to monitor the core power distribution. For a given TIP string, the measurement is a response to the integrated influence of the surrounding bundles. This signal strength from the fuel is primarily due to the cumulative power production of fuel rods in the four bundles surrounding the string.

The process for the TIP comparison basis is described below. The definitions of the quantities used in the calculations are:

$$P(k,j) = PCTIP(k,j) \tag{4.1}$$

$$C(k,j) = CALTIP(k,j) \tag{4.2}$$

where:

PCTIP(k,j)	= the measured six inch average TIP reading in axial segment k of TIP string j
CALTIP(k,j)	= the calculated six inch average TIP reading in axial segment k of TIP string j
J	= the total number of operational TIP strings
K	= *Kup* – *Klow* + *1*
Klow, Kup	= Node limits for axial comparison (e.g., 2 and 23)

The measured and calculated TIP strings are normalized, respectively, as follows:

$$\sum_{j=1}^{J} \sum_{k=Klow}^{Kup} P(k,j) = J \cdot K \tag{4.3}$$

$$\sum_{j=1}^{J} \sum_{k=Klow}^{Kup} C(k,j) = J \cdot K \tag{4.4}$$

The nodal RMS comparison is finally defined as:

$$R_{nod} = \sqrt{\frac{\sum_{j=1}^{J} \sum_{k=Klow}^{Kup} (P(k,j) - C(k,j))^2}{J \cdot K}} . \tag{4.5}$$

The bundle RMS is also defined as follows:

$$R_{rad} = \sqrt{\frac{\sum_{j=1}^{J} \left(\left(\frac{\sum_{k=Klow}^{Kup} P(k,j)}{K} \right) - \left(\frac{\sum_{k=Klow}^{Kup} C(k,j)}{K} \right) \right)^2}{J}} \tag{4.6}$$

Figure 4.3 demonstrates the performance of the nuclear methods in predicting instrument readings for three gamma TIP plants of the Reference BWRs: Plants A, B, and D. HCGS is a gamma TIP plant, therefore only gamma TIP plant data is discussed here. Plant C is a thermal TIP plant, and hence excluded from this comparison. Trend lines for the 2D average and 3D comparisons demonstrate the average trend for these quantities.

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Figure 4.3 – Reference BWR Operational Experience for TIP Comparisons

The TIP comparison results are well within the results previously provided to the NRC. Furthermore, the new results compare favorably with the uncertainties compiled earlier for PANAC11/TGBLA06. The quantity to consider is the RMS difference between the calculated and measured axially integrated TIP response. This RMS difference is used in part to determine the bundle power uncertainty in the Process Computer and is an input to the SLMCPR evaluation. The results are summarized below for Reference [6], which is the basis for the SLMCPR evaluations, a set of comparisons documented in Reference [7] for the PANAC11/TGBLA06 model and the results quoted for this analysis taken from the Reference BWRs. The results are shown in Table 4.1.

Table 4.1 – Summary of Bundle Average TIP Comparisons

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The data in Figure 4.3 demonstrates that not only is this number applicable to the range of operating experience for the Reference BWRs described in Section 2, but that this population of data is consistent with previous experience, as illustrated in Table 4.1.

Table 3.1 of Reference [2] shows the summary of TIP statistics for the CLTP HCGS. The average bundle RMS for Cycles 9-12 is [[ ]]. This information for CLTP combined with the information shown in Table 4.1 demonstrates confidence that the methodology is capable of predicting TIP responses in the EPU environment well.

The exposure dependence of the RMS difference has already been identified in RAI 3 of Reference [6] where use of the weighted average is justified rather than using the trend or a worst-case number.

The 3D comparisons, while not directly used in evaluations of SLMCPR or other licensing parameters, are also shown to be acceptable. The RMS difference of all non-adaptive nodal instrument predictions to measured TIP data, results in an average RMS difference of less than [[ ]] for the Reference BWRs indicating that axial power distributions are also predicted adequately.

5.0 Consideration of Legacy Fuel

The CLTP MCAR for HCGS Cycle 13[2] includes a lattice physics comparison section and a benchmark of previous operating cycles section. Section 2.0 of Reference [2] qualifies the lattice physics treatment of the legacy fuel (SVEA96+) and demonstrates its adequacy for application at the earlier CLTP condition. Such application adequately models SVEA96+ fuel for the HCGS core designs with the GE codes and methods.

Figure 5.1 illustrates the 115% HCGS EPU core analyzed in this report. This core consists of approximately 75% GE14 and 25% SVEA96+ legacy fuel. The SVEA96+ fuel in the 115% EPU core is the same fuel type benchmarked in Reference [2]. In addition, the SVEA96+ fuel types operating in the 115% EPU core experience conditions that are within the lattice physics benchmark established in Reference [2] (i.e., pre-EPU conditions) as illustrated in the metrics figures in Section 2. This demonstrates the adequacy of the lattice physics benchmark for application to the SVEA96+ fuel at the 115% EPU condition.

Section 2.0 of this report defines key methods parameters that can be compared to the existing BWR operating experience to demonstrate applicability of the GE codes and methods at the 115% EPU conditions. This section explicitly considers the legacy fuel since the 115% EPU core design shown in Figure 5.1 is used for the comparison. As demonstrated in Section 2.0, these comparisons for the HCGS EPU core show that the expected values of the key metrics are consistent with the range of values obtained from operation of the Reference BWRs. This validates that the Interim Methods LTR is applicable to the HCGS EPU core. In addition, this section illustrates that the values for the key metrics of the legacy SVEA 96+ fuel are consistent with the HCGS CLTP core values thereby demonstrating that the legacy fuel is effectively operating at pre-EPU conditions, consistent with the current operational experience, further validating the applicability of the Interim Methods LTR.

Section 4.0 of this report compares HCGS reference CLTP and 115% EPU core eigenvalues[1] to those of Reference BWRs that are representative of current operational experience. HCGS CLTP TIP statistics from previous operating cycles are also compared to TIP statistics from the Reference BWRs. This information for the CLTP and EPU cores, combined with the information shown in Table 4.1 demonstrates confidence that the methodology is capable of predicting eigenvalues and TIP responses at EPU conditions. Therefore, the calculational and measurement uncertainties applied to the thermal limits analyses are valid for the predicted neutronic and thermal-hydraulic core and fuel conditions.

Section 3.0 of this report addresses the transient methods at uprated conditions. For reactivity events, the conclusions of Section 2.0 and Section 4.0 of this report apply and, as discussed in the preceding paragraphs, both of these sections explicitly address all the fuel in the EPU core and therefore demonstrate the applicability of the reactivity event methods for the EPU conditions . For pressurization events, the core conditions are bounded by the GEXL and void-quality correlations for these events. The GEXL80 correlation [8] application range covers the

[1] Projected hot eigenvalues are reported for the 115% EPU cycle. The actual cold eigenvalues for the CLTP Cycle 13 and Cycle 14 at the beginning of the cycle are also reported.

complete range of expected operation of the SVEA96+ fuel during steady state and transient conditions for the HCGS EPU-core. For LOCA, the post-LOCA thermal hydraulic conditions are within the qualification basis of the SAFER-GESTR models and are not sensitive to the fuel type[9]. The GEXL and void-quality correlations are applied in the TASC code to determine dryout times for each fuel type. As discussed above, the GEXL80 correlation[8] application range covers the complete range of expected operation of the SVEA96+ fuel in the HCGS EPU core.

Lastly, [[ ]] analyses have also been performed and documented in the Fuel Transition Report for HCGS.[3,4] These analyses summarize the results of the [[ ]] review of the HCGS core in both CLTP and EPU conditions, and they conclude that HCGS can safely operate with a core consisting of GE14 and legacy SVEA96+ fuel at the EPU condition.

Together, the information presented in this report in conjunction with the MCAR, the [[ ]] analyses described above, and the Interim Methods LTR support the conclusion that HCGS can apply NRC approved GE methods and code systems to support safe operation of HCGS in the EPU condition.

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Figure 5.1 – HCGS 115% EPU Core Configuration

6.0 References

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