Attachment 17

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HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

REQUEST FOR LICENSE AMENDMENT
EXTENDED POWER UPRATE

MARGIN IN GE ANALYTICAL METHODS SUPPORTING HOPE CREEK EPU SUBMITTAL

Enclosure 2 to GE-HCGS-EPU-650, Rev. 2

Enclosure 2

GE-HCGS-EPU-650, Rev. 2

Margin in GE Analytical Methods Supporting Hope Creek EPU Submittal

Non-proprietary version

This is a non-proprietary version of Enclosure 1 to GE-HCGS-EPU-650, Rev. 2, which has the proprietary information removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here [[ ]].

Margin in GE Analytical Methods Supporting Hope Creek EPU Submittal

1. Summary:

To be included as part of PSEG's submittal of a license amendment request for the Hope Creek extended power uprate (EPU), PSEG has requested additional information to address NRC concerns related to GE's standard methodologies and the lack of recent gamma scan data. The information is being provided to facilitate NRC acceptance of the license amendment request. The information addresses the margins in the pertinent safety parameters. The information provides a basis for additional conservatism being proposed to address the NRC concerns and a basis for the adequacy of other existing conservatisms. Based upon the information, an operational restriction on the bundle critical power ratio is being proposed by PSEG as an additional conservatism. The operational restriction would be implemented via an adder of 0.02 to the operating limit minimum critical power ratio (OLMCPR). It is intended that the operational restriction would only be implemented as a condition of the EPU License Amendment if the aforementioned NRC concerns are not satisfactorily resolved prior to NRC approval of the license change request.

2. Introduction:

In its review of the Vermont Yankee EPU submittal, the NRC has asked questions related to the adequacy, given the absence of recent gamma-scan test data, of the standard uncertainties and biases utilized in GE's bundle lattice and core simulation methodologies for current fuel designs and operating strategies and has asked questions related to the potential effect on safety parameters influenced by such uncertainties and biases. As noted in Appendix A of this report, GE has benchmarked its methods using industry standard techniques and utilized gamma scan data to retrospectively confirm the adequacy of certain elements of its methods and benchmarking. GE has provided, and continues to provide, information to the NRC supporting the adequacy of GE's methodologies for application to BWRs and BWR EPU applications.

The following discussion addresses the NRC concerns regarding both gamma-scan and isotopic data as applicable to the HCGS EPU. The discussion identifies the six pertinent safety parameters that require disposition relative to the NRC concerns, provides a basis for the disposition of each, and includes the identification of an operational restriction that will be proposed on the bundle critical power ratio as an additional conservatism to address the aforementioned NRC concerns. The operational restriction would be implemented via an adder of 0.02 to the OLMCPR. For each of the six pertinent safety parameters:

- (a) the fuel parameters which affect it are identified,
- (b) the treatment of fuel parameter uncertainties in the safety parameter limit development is considered, and
- (c) the adequacy of the existing treatment in conjunction with any proposed additional conservatisms is supported.

3. Safety Parameters Influenced by Noted Uncertainties and Biases
GE has reviewed its methodologies to determine the uncertainties and biases which were confirmed by earlier gamma scan test data or measurements of irradiated fuel isotopics and to confirm that the existing types of uncertainties already included in GE's NRC-approved treatment of uncertainties and biases address the NRC staff concerns regarding the absence of recent confirmatory test data.

The associated fuel parameters related to such test data and measurements that are not otherwise measurable directly or indirectly by existing operating plant instrumentation, e.g., local power range monitors (LPRMs) and traversing in-core probes (TIPs), are:

a. Local fuel pin power and exposure (depletion) vs. axial position,
b. Relative local fuel pin power and exposure (local in-bundle peaking),
c. Void reactivity coefficient, and
d. [[ ]]

The fuel parameter uncertainties of interest are thus related to relative local and pin power peaking, void reactivity coefficient, and [[ ]]. Other nodal fuel and bundle parameters, e.g., lattice reactivity, bundle power, and bundle axial power shape, can be and are satisfactorily and adequately confirmed by comparisons to operating plant data or tests, e.g., TIP data and shutdown margin demonstrations.

The safety parameters potentially influenced by the local and relative pin power uncertainties and the [[ ]] uncertainty are:

1. Critical power (protected by the safety limit minimum critical power ratio (SLMCPR) and OLMCPR),
2. Shutdown margin (controlled with a technical specification limit of 0.38% $\Delta k/k$ for in-sequence demonstrations),
3. Fuel rod thermal-mechanical performance (protected by limits on linear heat generation rate, LHGR),
4. LOCA-related nodal power limits (controlled via the maximum average planar linear heat generation rate, MAPLHGR),
5. Stability (protected by the SLMCPR, OLMCPR, and stability solutions), and
6. Licensed pellet exposure (e.g., 70 GWd/MT for GE14 fuel)

Each of the uncertainties in question is currently included and addressed in the treatment of uncertainties and biases in GE's NRC-approved methodologies to determine these safety parameters. It is appropriate to continue to utilize the NRC-approved GE treatment of uncertainties and biases. If consideration of larger uncertainties is deemed appropriate, such uncertainties can be utilized in the existing treatments of propagation and combination of uncertainties. Direct application of biases into best estimate codes in an attempt to address potential uncertainty concerns is not appropriate because such introduction of unqualified biases would lead to potential nonconservatisms in resulting predictions. Therefore, the fidelity of GE's codes and methods is best maintained by not artificially adding biases. Conservative limits on safety parameters, developed with

consideration for such uncertainties, provide reasonable assurance of plant and public safety.

A discussion of the adequacy of the margin existing in, and, as applicable, augmented margin for, each of these limits is provided below. For each limit, the potential influence of, and applicable associated conservatisms related to, the pin power/exposure, void reactivity coefficient, and [[ ]] uncertainties are discussed.

3.1A SLMCPR Margin

GE's NRC-approved process of determining the SLMCPR incorporates the applicable uncertainties in the lattice and core physics parameters, and the method of determining SLMCPR assures that fuel is protected from boiling transition when such uncertainties are incorporated. The SLMCPR is directly affected by the fuel parameters confirmed by gamma scan data, i.e., local pin power, relative pin power, and [[ ]]. SLMCPR is not affected by void reactivity coefficient uncertainties. Uncertainties in local pin power peaking and [[ ]] (and bundle power) are explicitly included in the SLMCPR determination and considered separately, then cumulatively, below.

The potential effect of larger pin power uncertainty on the SLMCPR has been considered. First, in lieu of an arbitrary increase in the uncertainty, a review of [[ ]] In the determination of SLMCPR, the use of additional pin power uncertainty so derived, i.e., [[ ]] (to be confirmed for the final Hope Creek EPU cycle core), providing real additional critical power margin relative to GE's standard methodology, and this additional margin addresses local peaking uncertainty concerns.

[[ ]] is a component of the total bundle power uncertainty. The total bundle power uncertainty for application within GE's NRC-approved SLMCPR determination process is comprised of the component uncertainties in the following table (From Table 4.2, page 4-2 of Reference 5.1).

Quantity	Uncertainty	Source
[[		
		]]

GE has continued to provide the NRC with BWR fleet information on the consistency of integral TIP comparisons on a periodic basis, e.g., in fuel technology updates. In 2005, GE formally provided a large amount of these data for uprated plants loaded primarily with 10x10 fuel in methods related RAI responses under the MELLLA+ docket (Reference 5.2). Examination of this data confirms the applicability of the original [[ ]] uncertainty documented in GE's NRC-approved topical report describing the SLMCPR methodology power distribution uncertainties (References 5.3 and 5.1).

[[

]]

BWRs have always operated at void fractions higher than 70% with some of the earlier gamma scan data from fuel exceeding 80% void fractions so that the effect of void fraction is included in the confirmation of local and bundle power peaking uncertainty and, thus, is not a significant concern. Instead, the largest differences in bundle power are the result of depletion and are not the result of differing product lines, composition, or core power. This key aspect is already addressed in the current NRC approved value [[

]] Therefore, the procedure of using the current gamma scan data to determine a conservative bound on the uncertainty is reasonable and valid.

[[

]] (to be confirmed for the final Hope Creek EPU cycle core). This additional critical power margin provides a real additional assurance of safety and is developed consistent with current NRC-approved bundle power uncertainty methodology.

The effect of [[ ]] in the above table on the bundle power uncertainty for SLMCPR determination [[ ]]

[[ ]] 0.02 CPR adder on SLMCPR based on these conservatively increased local peaking and [[ ]] uncertainties. [[ ]] is further conservative.

In summary, use of alternative, even more conservative values for uncertainties in the local peaking factor and [[ ]] results in an estimated increase in the SLMCPR for Hope Creek of 0.02 relative to that calculated with current GE standard methodology and provides reasonable assurance of safety for Hope Creek EPU with respect to SLMCPR. This result will be confirmed for the final Hope Creek EPU cycle core as indicated above.

Application of the above 0.02 adder to the Hope Creek OLMCPR (set by the limiting anticipated operational occurrence) rather than to the SLMCPR achieves the same arithmetic effect and protection of plant and public safety as does application of the 0.02 as an adder to the SLMCPR.

3.1B OLMCPR Margin

The analysis of anticipated operational occurrences (AOOs) examines the change in critical power ratio relative to the original starting point conditions and determines the

most limiting transient event. The fuel parameters identified previously, i.e., the local (pin) power peaking, void reactivity coefficient, and [[ ]], are factors in the evaluation of limiting AOOs. [[

]] Using this process assures that the analysis is both realistic and conservative.

Accommodation for uncertainties in local pin power peaking and [[ ]] (and bundle power), i.e., consideration of bundle and nodal powers higher (or lower) than expectations, is directly incorporated in the licensing methodology by the conservative initial conditions described above and the conservative treatment of SLMCPR described in section 3.1A. Thus, there is no direct effect on the ΔCPR methodology due to the NRC staff concerns regarding the local pin power peaking and [[ ]] uncertainties.

Both the ODYN and TRACG transient methodologies have established application ranges for void coefficient uncertainty. The prior NRC approval of and GE confidence in the basis for these methodologies is based upon comparison of calculations for a wide variety of plant transients in which the nominal void coefficient is used. The acceptable performance of these codes relative to the data justifies that no large errors in void coefficient exist.

The ODYN model uncertainty is based on comparisons to the benchmark Peach Bottom turbine trip tests (Reference 5.4). [[

]] Because inputs to the OLMCPR analysis are conservative, and the pressurization transients are conservatively analyzed by ODYN and typically establish the limiting ΔCPRs, conservatisms existing in the process of determining OLMCPRs address NRC staff concerns related to gamma scans and fuel isotopics as they relate to OLMCPR.

The rod withdrawal error (RWE), loss of feedwater heating (LOFWH), and fuel loading error events not analyzed using the ODYN methodology are not significantly, if at all, affected by the postulated larger uncertainties in the fuel parameters previously identified. [[

]] Finally, these events are typically not limiting relative to the pressurization events, i.e., they typically do not establish the OLMCPR. If one of these nonpressurization events were to be the limiting AOO, the adder discussed in the SLMCPR section would be added to the corresponding OLMCPR.

In summary, the standard GE methodologies utilized to establish the OLMCPR conservatively address the uncertainty issues and provide reasonable assurance of plant and public safety for Hope Creek EPU with respect to OLMCPR.

3.2 Shutdown Margin (SDM)

The analysis of SDM considers whether core reactivity can be safely controlled. The fuel parameters identified previously, i.e., the local (pin) power peaking and [[

]], are indirect factors in the evaluation of SDM because uncertainties in those parameters may ultimately influence prediction of fuel depletion and, thus, fuel reactivity. The void reactivity coefficient is not a contributor because essentially zero voiding is present at hot or cold shutdown conditions. As described in Appendix A to this report, the GE bundle lattice and core simulation methodologies are best estimate predictions so that validation of operating benchmark data, core follow, and core licensing can proceed continuously using consistent methodology. Comparisons to actual cold critical states are an important part of this verification because any error in bundle or nodal power (or exposure) would tend to degrade the ability of the core simulator to establish a stable bias (in eigenvalue), a measure of the ability of the model to reliably predict core hot and cold critical conditions. While the technical specification for SDM is 0.38% Δk/k reactivity (for an in-sequence check only), the normal GE design procedure is to provide design cold shutdown margins of 1% depending on customer request and GE procedure. The uncertainty in cold critical predictive capability is directly considered and included in this choice of design SDM requirement.

However, it is very important to note that actual SDM is a demonstrated quantity (plant verification) during plant startups or by use of local criticality confirmations. In addition, plants perform trending of hot eigenvalue (i.e., reactivity anomalies), which is required by plant Technical Specifications and is another direct confirmation of the adequacy of GE's methods with respect to fuel depletion and reactivity predictions. Because such plant verification data from power uprated plants and plants with modern fuel designs, including GE14, have continued to confirm that adequate SDM exists and that eigenvalue biases in GE's methods are stable and well-understood, there is sufficient justification for the adequacy of GE's bundle lattice and core simulation methodologies and the uncertainties in the nodal and bundle power and exposure even without recent confirmatory gamma scan or fuel isotopic data.

The analysis of the SDM provided by standby liquid control system (SLCS) utilizes a conservative SDM criterion which accounts for all of the biases and uncertainties inherent in the components of the methodology. The SLCS SDM also is driven more by core-wide reactivity effects and is much less sensitive to nodal uncertainties in exposure and isotopic content than the cold SDM.

In summary, the current design process and design goal, in combination with the existing process of plant verification of SDM and trending of hot eigenvalues, provide reasonable assurance of adequate SDM.

3.3 LHGR Margin

For each GNF fuel design, including GE14, thermal-mechanical based linear heat generation rate (LHGR) limits are specified for each fuel rod type (for both UO_2 and gadolinia-bearing rods) such that, if each rod type is operated within its LHGR limits, all thermal-mechanical design and licensing criteria, including those which address response to anticipated operational transients (AOOs), are explicitly satisfied and fuel rod integrity is maintained. The fuel parameters identified previously, i.e., the local (pin) power peaking, void reactivity coefficient, [[ ]], are factors, to differing extents, in the development of LHGR limits. The fuel parameters ultimately determine the local power, which is explicitly addressed by the LHGR limit.

Fuel rod thermal-mechanical licensing criteria explicitly considered in the specification of LHGR limits include fuel centerline temperature, cladding plastic strain, and fuel rod internal pressure. Each of these criteria is limiting over a portion of the fuel rod lifetime. For development of the final limit curve, the peak power node is conservatively assumed [[ ]] In addition, model and operating power uncertainties are explicitly addressed in the development of limits, including an additional fuel rod nodal power uncertainty of [[ ]] applicable to and suitable for addressing local pin power peaking uncertainty, as well as a [[ ]] conservative power bias in the fuel rod internal pressure calculation. The uncertainty and bias also apply to exposure because, in the determination of LHGR limits, the exposure is the integrated power.

Moreover, the model uncertainties in GE's NRC-approved thermal-mechanical analysis methodology (GESTR-Mechanical, Reference 5.5) are based on temperature benchmark data and are also validated via fission gas benchmark data for which the nominal power history is produced in the steady-state core simulator. Because the large uncertainties included by this process encompass the uncertainties in local and rod power reflected in the NRC staff questions and because separate experimental benchmarking information confirms that the model uncertainties remain valid, an adjustment to provide additional LHGR margin is unnecessary.

In summary, the standard GE methodology for determining LHGR limits includes conservative consideration for, and provides reasonable assurance of adequate margin to address, the power and void reactivity uncertainties in question.

3.4 MAPLHGR Margin

The purpose of the maximum average planar linear heat generation rate (MAPLHGR) limits is to assure adequate protection of the fuel during a postulated loss-of-coolant accident (LOCA) with the defined operation of emergency core cooling system (ECCS). The fuel parameters identified previously, i.e., the local (pin) power peaking and [[ ]], are factors, to differing extents, in the development of MAPLHGR limits. The fuel parameters ultimately determine the local power, which is the subject of the MAPLHGR, a local limit. The void reactivity coefficient is not a factor in the ECCS-LOCA analysis.

The ECCS-LOCA analysis applicable to the Hope Creek EPU follows the NRC-approved SAFER/GESTR application methodology documented in Reference 5.6. The analytical models used to perform ECCS-LOCA analyses are also documented in Reference 5.6 together with Reference 5.7 and Reference 5.8.

When SAFER/GESTR methodology is applied, the hot bundle is initialized with a [[ ]] In addition, a [[ ]] In order to ensure that the SAFER analysis is bounding for all exposures, the hot rod of the hot bundle is placed at the exposure corresponding to the [[ ]]

Total bundle power is also important to the severity of the ECCS-LOCA analysis. [[ ]] Furthermore, the ECCS-LOCA basis target MCPR is set lower than the OLMCPR so that the OLMCPR is not set by the ECCS-LOCA analysis (thus set by the AOO analysis).

Pin power peaking for the hot rod is set to a [[ ]] to further insure that the ECCS-LOCA results are bounding.

Lastly, the axial power profile [[ ]]

The above considerations indicate that significant conservatisms related to initial local (pin) and bundle powers exist in the GE SAFER/GESTR ECCS-LOCA methodology.

In addition to the above conservatisms, the Hope Creek Licensing Basis peak clad temperature (PCT) determined by the methodology described above is 1380°F for GE14 fuel and 1540°F for SVEA-96+ fuel. This result includes application of Appendix K modeling assumptions. For the limiting break, the nominal PCT is about [[ ]] When the nominal PCT is adjusted to account for model uncertainties (at 95% probability), the PCT (also known as the Upper Bound PCT in the SAFER/GESTR methodology) is lower than the Licensing Basis PCT. The 95% probability PCT includes an uncertainty of [[ ]] on the LHGR (or approximately [[ ]]).

The SAFER/GESTR methodology assumes a bounding post-LOCA core power trajectory and, thus, core kinetics are not modeled. The average and hot bundle void profile is determined by SAFER at the limiting initial conditions described above as well as at the post-LOCA conditions. Uncertainties in predictions of void reactivity have no impact in the SAFER/GESTR methodology. The overall SAFER/GESTR methodology is designed to maximize the PCT.

In summary, the conservatism of the present ECCS-LOCA methodology used to determine MAPLHGR limits adequately considers the effects of the uncertainties in local and bundle power and provides reasonable assurance that those limits provide adequate margin to protect the fuel.

3.5 Stability

BWR thermal-hydraulic stability analyses are performed to assure that SLMCPR is protected in the event of a thermal-hydraulic instability event. The fuel parameters identified previously, i.e., the local (pin) power peaking, void reactivity coefficient, [[ ]], are factors, to differing extents, that could affect the stability evaluation.

Hope Creek has implemented the Option III stability solution. The examination of core and fuel stability behavior begins with fuel assumed to be at the OLMCPR and terminates

once power oscillations cause fuel critical power to reach the SLMCPR. As discussed in Section 3.1A, an effect of 0.02 on the SLMCPR based on conservatively increased local peaking and [[ ]] uncertainties has been described. Hope Creek is proposing to implement this additional conservatism as a 0.02 adder to the OLMCPR to address NRC concerns relative to these parameters.

A key goal in the determination of an Oscillation Power Range Monitor (OPRM) setpoint is that stability is not expected to establish the rated power OLMCPR. In other words, the change in CPR during an instability event must be less than or equal to the change in CPR from the limiting AOO. The OPRM setpoint will be reduced until this relationship is true, and will include an appropriate consideration of the 0.02 adder. Consequently, the uncertainties in power distribution calculation and void reactivity coefficient as they affect the SLMCPR will not impact the safety margin in the stability analysis for Hope Creek. If an OPRM setpoint was selected that resulted in stability setting the rated power OLMCPR, the same 0.02 adder would be applied to this stability result.

As discussed before in relation to nodal and core reactivity, uncertainties or biases in depletion isotopics at high exposure and void conditions from prediction which might have a postulated effect on void reactivity coefficient would manifest themselves in separately observable differences in local and core power and reactivity. The variation of void reactivity coefficient across the GE BWR fleet encompasses significant variations in bundle and core exposures and void fraction and is well behaved. The effect of void reactivity coefficient on instability events is well understood via existing code qualification parametric studies. Large unknown uncertainties in the void reactivity coefficient would be noticeable as an inability to reasonably model instability events. The existing GE thermal-hydraulic stability models reasonably and adequately model the magnitude and period of industry thermal-hydraulic instability events. Both GE stability codes (frequency domain code ODYSY [Reference 5.9] and time-domain code TRACG) model past events, including the recent thermal-hydraulic instability events at Nine Mile Point 2 and Perry, relatively well and this lends credence to the accuracy of the void model in the GE methodology.

Key inputs to the OPRM setpoint determination analysis are the DIVOM (delta CPR/initial CPR versus oscillation magnitude) slope and Hot Channel Oscillation Magnitude (HCOM). These inputs would not be affected by an increase in the OLMCPR or the SLMCPR. Since the transient stability analysis results (delta CPR/initial CPR) are not affected, the stability envelope is not affected.

Key HCOM inputs are LPRM to OPRM assignments, total scram delay time, RPS trip logic, and averaging/conditioning filter cutoff frequencies. A new HCOM is required only if one of these key (but unrelated to OLMCPR or SLMCPR) parameters changes.

In summary, the uncertainties in power distribution calculation and void reactivity coefficient are adequately addressed and will not impact the safety margin in the stability analysis for Hope Creek.

3.6 Margin to Licensed Fuel Exposure

GE fuel designs are licensed to a peak pellet exposure (i.e., 70 GWd/MTU for GE14). This is equivalent to a GE14 rod average exposure of ~61.4 GWd/MTU, but there is not an explicit rod average exposure limit for GE14 or other GE fuel designs. This limit is used to assure that fuel is not operated beyond its analysis basis. The fuel parameters identified previously, i.e., the local (pin) power peaking, void reactivity coefficient, [[ ]], are factors, to differing extents, in the development of LHGR limits, and, thus, the fuel exposure limit. The fuel parameters ultimately determine the local power, which is explicitly addressed by the LGHR limit.

Fuel rod internal pressure is the limiting licensing criterion at end-of-life for GE fuel designs. The fuel cladding creep rate is a function of cladding temperature and in turn of LHGR. As discussed previously, the LHGR limits for GE14 are deliberately conservative with respect to local power, assume a conservative pellet swelling rate uncertainty, and are also specified in such a way that the margin to the criterion for limiting pellet-cladding gap increase due to rod internal pressurization is actually smaller several GWd/MTU before end-of-life than at the peak pellet exposure (end-of-life) limit. Thus, existing uncertainties and margins in GE's NRC-approved fuel thermal-mechanical methodology adequately address the NRC concerns regarding local peaking uncertainty with respect to the licensed fuel exposure limit.

In summary, the GE standard fuel thermal-mechanical analysis basis considers and provides adequate margin for uncertainties in local and bundle power.

4. Additional Margin Summary

The evaluation above provides the basis for a determination that an adder to the OLMCPR of 0.02 (reflecting the underlying effect on the SLMCPR) provides additional and reasonable assurance of plant and public safety for Hope Creek at EPU conditions and addresses NRC concerns related to the lack of recent gamma scan data. Significant conservatisms already exist in the processes for determination of the other safety parameters, i.e., OLMCPR margin, SDM, LHGR, MAPLHGR, thermal-hydraulic stability protections, and fuel (peak pellet) exposure, to address the NRC staff concerns, and adjustments or operational restrictions for these are, thus, not required.

5. References

5.10 NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations", August 1999.
5.11 GE Letter to NRC, MFN 05-029, TAC No. MC5780, April 8, 2005.
5.12 NEDC-32601P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluation", August 1999.
5.13 NEDO-24154P-A, Volume III, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", October 1978.
5.14 NEDE–24011–P–A, GESTAR II, Amendment 7.
5.15 NEDE-23785-1-PA Rev. 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-Of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," October 1984.
5.16 NEDE-30996P-A, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-jet Pump Plants, Volume I, SAFER – Long Term Inventory Model for BWR Loss-of-Coolant Analysis," October 1987.
5.17 NEDC-32950P, "Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," January 2000.
5.18 NEDC-32992P-A, "ODYSY Application for Stability Licensing Calculation", July 2001.

Appendix A
GE Bundle Lattice and Core Simulation Methodology
&
Utilization of Gamma Scan and Fuel Isotopic Data

Summary:
GE's bundle lattice and core simulation codes, TGBLA and PANACEA, are best-estimate methods with uncertainties and biases in inputs and outputs of those codes addressed by the conservative treatment, previously approved by the NRC, of uncertainties and biases propagation in GE's calculations of conservative limits for various fuel safety parameters. The bundle lattice methods have been benchmarked, using industry standard practice, against Monte Carlo calculations for all GE fuel types. These benchmarks have been further confirmed for certain GE fuel types, retrospectively, with gamma-scan data available to GE. The core simulator methodology has been benchmarked, again using industry standard practice, against the operating plant instrumentation, e.g., traversing in-core probes (TIPs). [[

]] Operating plant data are continuously utilized to evaluate the accuracy of predictions of the bundle lattice and core simulator methodologies on both a plant-specific and BWR fleet-wide basis, and such trending is periodically (approximately annually) reviewed with the NRC staff in fuel technology update meetings.

In accordance with its understanding of previous NRC-approved licensing topical reports and NRC-issued safety evaluations for GE's methods, GE has evaluated and reflected the accuracy of its methodologies as it has introduced new fuel designs and operating strategies. Consequently, the GE bundle lattice and core simulator methodologies, including the associated uncertainties and biases utilized by GE, in combination with its NRC-approved treatment of uncertainties and biases, are adequately predicting the performance, and assuring the safety, of BWRs at up to and including 120% EPU conditions.

Qualification Process:
GE utilized rod gamma scans, i.e., measurements of gamma emissions from certain fission product isotopes in individual irradiated BWR fuel rods, to further confirm the ability of its benchmarked methods to adequately predict local (fuel pin) power and exposure (i.e., burnup or depletion). GE utilized bundle gamma scans, i.e., scans of entire BWR fuel bundles, to confirm an appropriate value for uncertainty related to the [[ ]] GE utilized irradiated fuel rod isotopic measurements, i.e., radiochemistry determination of inventory of certain fission and activation products, which are necessarily limited in number due to the difficulty in obtaining such measurements, in lattice physics code development but not as part of code benchmarking.

GE evaluates methods on multiple geometrical bases. The process of monitoring operational core parameters provides an up-to-date (hourly) evaluation of steady-state

core reactivity control and provides a way to evaluate the core simulator eigenvalue bias. Comparison of calculated to measured TIP signals provides confirmation of the three-dimensional field of flux/power on a very timely basis (monthly) but with a resolution scale that only reflects the coarse mesh resolution of the three-dimensional simulator. Natural noise in the TIP instrumentation conservatively results in a fundamental contribution of 1% to the evaluated comparison (Reference 5.1, page A-7). Bundle-wise or pin-wise gamma scans allow for a better resolution in space but result in a poor temporal comparison because the present concentration of the typically measured fission product (^{140}Ba) requires an integration of the power history for the prior sixty days. Moreover, because of limitations, gamma scans may only be achievable once per cycle for operating power reactors. Bundle gamma scans usually entrain an experimental uncertainty of 1% (1σ) in the measured values while rod gamma scans entrain an uncertainty of 2% (1σ).

Because the injection of experimental error of non-routine benchmarking may confound physical phenomena of interest and for purposes of more timely and comprehensive evaluation, it is meaningful to compare production lattice physics methods (TGBLA) to Monte Carlo methods whose efficacy has been established through comparison to critical benchmarks. Assuming adequate trials have been considered, the local accuracy provides significant insight for examination of relative local pin peaking accuracy. If the local power is being produced correctly, the subsequent depletion of the fuel is occurring at the correct rate and location. Furthermore, assuming the nominal production lattice physics code produces stable core eigenvalue behavior (evaluated in the operational core follow examination), use of the depleted isotopic compositions from the deterministic code for comparisons to Monte Carlo later in the life of the fuel is both meaningful and produces further insight into modeling accuracy. The conclusion is that it is meaningful and proper to consider comparisons between TGBLA and Monte Carlo methods in evaluation of methods accuracy.

In summary, the GE standard fuel lattice and core simulator methodology qualification process utilizes a large volume of contemporaneous operating plant data supported by available confirmatory, retrospective gamma scan data to assure high-quality best-estimate predictions of local, nodal, bundle, and core power. As discussed in the body of this report, GE's NRC-approved treatment of the uncertainties in the power predictions assure conservative limits for the safety parameters influenced by the local, nodal, bundle, and core power.