ENCLOSURE 2

MFN 16-024

Comment Summary Table and Draft SE Markup

Non-Proprietary Information – Class I (Public)

IMPORTANT NOTICE

This is a non-proprietary version of Enclosure 1, which has the proprietary information removed. Portions of the document that have been removed are indicated by white space with an open and closed bracket as shown here [[ ]].

Comment Summary for Draft Safety Evaluation for NEDC-33376P Revision 1, "Lattice Physics Model Description" and NEDC-33377P Revision 1, "Lattice Physics Model Qualification"

Note: Page numbers shown in this table reflect the page numbers in this enclosure. Due to suggested changes in the Safety Evaluation (SE) and the addition of a change summary table, these page numbers differ from the page numbers in the draft SE sent to GNF for review.

Location	Comment
Section 3.1 LANCR02 Calculation Overview	Page 9: This section mistakenly indicates that LANCR02 condenses the ENDF cross sections to the fine group cross sections. This condensation process is performed outside of LANCR02 and provided as input to the code. This is in contradiction to Section 3.3.2.1 which correctly indicates that this condensation is a preprocessing step. GNF suggests the following changes: (Lines 25-27) the evaluated nuclear data were condensed from ENDF to fine group cross sections and made available as input to the LANCR02 code ~~LANCR02 condenses the evaluated nuclear data from ENDF to fine group cross sections~~ (Line 28) This external calculation *Suggested changes shown in the markup.*
Section 3.1 LANCR02 Calculation Overview	Page 9: The f-tables require both the background cross section and fuel temperature to obtain the corresponding fine group cross section. GNF suggests the following change (Lines 40-42): that is used in conjunction with the fuel temperature to interpolate the f-tables to extract the correct fine-group cross sections~~that is used to interpolate the fine group cross sections according to the f tables~~ *Suggested changes shown in the markup.*
Section 3.1 LANCR02 Calculation Overview	Page 10: GNF suggests the following change (Line 17): calculations are ~~also~~ performed during the fifth step. *Suggested changes shown in the markup.*
Section 3.3.1.1 Neutron Cross Sections	Page 11: GNF suggests the following change (Line 34): ~~(~~ENDF~~)~~/B-VII.0 *Suggested changes shown in the markup.*

Location	Comment
Section 3.4 Model Description	Page 14: The section heading font is inconsistent with similar headings in this document which appear in bold and are underlined. GNF suggests the following change (Line 16): **Model Description** *Suggested changes shown in the markup.*
Section 3.4.1.1 Background Cross Section	Page 14: GNF suggests the following change (Line 45): ~~macroscopic~~ potential scattering *Suggested changes shown in the markup.*
Section 3.4.1.1 Background Cross Section	Page 15: The typical Bell factor value of 1.16 is not only limited to pressurized water reactors and it would be more appropriate to use the phrase light water reactor. GNF suggests the following change (Line 10): ~~pressurized water~~light water *Suggested changes shown in the markup.*
Section 3.4.5 Depletion	Page 27: The exposure time steps are typically 1 GWd/ST but may increase after all the Gd-155 and Gd-157 have been sufficiently depleted (see page 5-3 of the Model Description LTR). GNF suggests the following change (Line 29): the flux solution typically at each 1 GWd/ST to establish the change in flux distribution *Suggested changes shown in the markup.*
Section 3.5.2.4 International Benchmarks	Page 32: Section 4.4.1 on page 4-22 of the Model Qualification LTR indicates that 127 critical benchmarks from the ICSBEP were used. GNF suggests the following change (Line 27): GNF identified a set of ~~270~~ 127 critical evaluations *Suggested changes shown in the markup.*
Section 3.5.3.3 Edge Part Length Rods	Page 36: The first sentence of the 3rd paragraph cites the incorrect LTR. Figure 5.3-3 is located in the model qualification LTR. GNF suggests the following change (Line 11): Figure 5.3-3 of the model ~~description~~ qualification LTR illustrates… *Suggested changes shown in the markup.*

Location	Comment
Section 3.5.3.5 Reactivity Worth of Depleted Lattice	Page 36: The 2nd sentence contains a typographical error. GNF suggests the following change (Line 46): The suite considered both ~~urania~~ uranium oxide (UOX) and MOX fuel. *Suggested changes shown in the markup.*
Section 3.5.3.5.1 Uranium Oxide Results	Page 37: The first indicated trend is actually between the uncontrolled cases at 0 and 40 GWd/ST as described on page 5-104 of the Model Qualification LTR. GNF suggests the following change (Line 36): between the hot uncontrolled case at BOL and 40 GWd/ST *Suggested changes shown in the markup.*
Section 3.5.3.7 Gadolinium Concentration	Page 40: Table 5.7-4 in the Model Qualification LTR gives the hot doppler eigenvalue for all cases as an average of [[ ]] and a standard deviation of [[ ]]. GNF suggests the following change (Line 31): than [[ ]] *Suggested changes shown in the markup.*
Section 3.5.3.7 Gadolinium Concentration	Page 40: Table 5.7-4 in the Model Qualification LTR gives the hot controlled eigenvalue for all cases as an average of [[ ]] and a standard deviation of [[ ]]. GNF suggests the following change (Line 46): than [[ ]] *Suggested changes shown in the markup.*
Section 3.5.3.9 Enrichment	Page 42: The enrichment test suite corresponds to test suite 9 in Section 5.9 of the Model Qualification LTR. GNF suggests the following change (Line 3): ~~8~~ 9 *Suggested changes shown in the markup.*
Section 3.5.3.9 Enrichment	Page 42: The fission rate RMS is provided in Figure 5.9-3 for the enrichment test suite in Section 5.9.4 of the Model Qualification LTR. GNF suggests the following change (Line 41): from Figure 5.9-~~2~~ 3 from *Suggested changes shown in the markup.*

Location	Comment
Section 3.5.3.11 Alternative Control Blade Designs	Page 44: Table 5.1-8 in the Model Qualification LTR gives average and standard eigenvalues of [[ ]] for the lattices in the baseline test suite. GNF suggests the following change (Line 6): [[ ]] *Suggested changes shown in the markup.*
Section 3.5.3.19 Summary of Trends and Conclusions	Page 48: The analysis described in Section 3.5.3.9 demonstrates the LANCR02 ability to simulate enrichments up to [[ ]] without exceeding the trend criterion. GNF suggests the following changes (Lines 43 to 46): [[ ]] *Suggested changes shown in the markup.*
Table 3.5.3.19.1 Summary of Observed Trends	Page 50: In Table 3.5.3.19.1 the eigenvalue differences for the UOX fuel in the first two rows of the table have been transposed GNF suggests the following changes: (first row of the table): [[ ]] (second row of table) [[ ]] *Suggested changes shown in the markup.*
Section 4.1 Range of Applicability	Page 53: Line 32 indicates a maximum permissible enrichment of 5.0 weight percent. A distinction should be made between the physically allowed enrichments and the general ability of LANCR02 to simulate different enrichments. The analysis described in Section 3.5.3.9 demonstrates the LANCR02 ability to simulate enrichments up to [[ ]] without any trends. GNF suggests the following changes (Line 32): with enrichment at or below ~~the maximum permissible enrichment of~~ [[ ]] weight percent *Suggested changes shown in the markup.*

Location	Comment
Section 4.1 Range of Applicability	Page 53: Lines 35 to 37 suggest that the range of applicability for the exposures in LANCR02 be based on bundle average or nodal parameters rather than lattice physics parameters. The bundle average exposure and peak pellet exposure limits are fuel bundle design limits which are a reflection of a variety of fuel thermal-mechanical design and other core-wide considerations. The combination of analyses described in SE Sections 3.5.3.5 (Model Qualification LTR Section 5.5) and 3.5.3.18 (Model Qualification LTR Section 6.0) demonstrate the LANCR02 capability for average lattice exposures of up to [[ ]]. Lines 1 to 5 on page 48 of the SE further indicates that the exposures considered in the depletion test suite are sufficiently long and that trends beyond this exposure do not provide additional insight into the depletion capability of the codes involved. While this range of exposure is not representative of the currently approved exposures in physical fuel products, it is within the ability of LANCR02 to model. Extended burnup Lead Test Assemblies as well as future fuel products will push the fuel bundle exposure limit to improve fuel economics and reduce quantities of stored bundles. GNF suggests the following changes (Lines 35-37): LANCR02 may only be considered applicable for analyses up to an [[ ]] average lattice exposure. *Suggested changes shown in the markup.*
Section 4.1 Range of Applicability	Page 53: It is understood that evolutionary changes are allowed provided that they do not alter the LANCR02 methodology. GNF suggests the following changes (Line 47): [[ ]] *Suggested changes shown in the markup.*
Section 4.2 Methodology and Data Source Updates	Page 54: The limitation described in the last sentence in Section 3.6.1 should also be included in Section 4.2. GNF suggests the following changes (Line 11): associated justifications are found to be acceptable. Any increased capabilities resulting from code updates should not be used for alternative safety analyses beyond those reviewed by the staff without prior NRC review and approval. *Suggested changes shown in the markup.*

Location	Comment
Section 4.2 Methodology and Data Source Updates	Page 55: The exception for updates to correct coding errors listed on page 9-2 of the Model Description LTR has not been included in the list of exceptions that start on Line 34 of page 54 of the SE. GNF suggests the following changes (Line 4): [[ ]] *Suggested changes shown in the markup.*
Section 4.2 Methodology and Data Source Updates	Page 55: The limitation on numerical method improvements to improve code convergence described on Lines 22 to 30 on page 51 is not included in Section 4 of the SE GNF suggests the following changes (Line 8): [[ ]] *Suggested changes shown in the markup.*
Section 4.2 Methodology and Data Source Updates	Page 55: Lines 12 to 16 should be understood to refer to changes in the LANCR02 methodology rather than the code changes which are recorded and maintained in the GNF quality records. GNF suggests the following changes: (Line 12) [[ ]] *Suggested changes shown in the markup.*

Markup of Draft Safety Evaluation for Global Nuclear Fuel - Americas Licensing Topical Reports Draft Safety Evaluation for NEDC-33376P Revision 1, "Lattice Physics Model Description" and NEDC-33377P Revision 1, "Lattice Physics Model Qualification"

The following markup illustrates the GNF proprietary content and suggestions per the comment summary table.

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO TOPICAL REPORTS: NEDC-33376P, REVISION 1, "LANCR02 LATTICE PHYISCS MODEL DESCRIPTION," AND NEDC-33377P, REVISION 1, "LANCR02 LATTICE PHYSICS MODEL QUALIFICATION"

GLOBAL NUCLEAR FUEL – AMERICAS, LLC

1.0 INTRODUCTION

Global Nuclear Fuel – Americas (GNF) is currently developing an advanced nuclear design analysis methodology based on the LANCR and AETNA codes. Specifically, by letter dated June 30, 2009 (Reference 1), GNF has submitted the LANCR02 lattice physics code model description Licensing Topical Report (LTR) (Reference 2) and model qualification LTR (Reference 3) for U.S. Nuclear Regulatory Commission (NRC) review and approval.

LANCR02 is used in the LANCR/AETNA code system to: (1) provide nuclear data input to AETNA for the purpose of performing nuclear design safety analysis (such as shutdown margin calculations), (2) provide data for other applications such as Safety Limit Minimum Critical Power Ratio calculations, and (3) input to transient analysis codes such as TRACG.

As specified in the letter dated June 30, 2009, GNF has requested that the NRC conduct a review of the LANCR02 method for calculating these inputs to downstream analysis codes.

2.0 REGULATORY EVALUATION

The regulations in Title 10 of the *Code of Federal Regulations*, Section 50.34, (10 CFR 50.34) require licensees to perform safety analyses of their facilities. GNF has submitted the LANCR02 lattice physics methodology for generic review by the NRC so that licensees may reference these methods in safety analyses performed to support licensing without further NRC review.

LANCR02 is a lattice physics code that processes nuclear data for use in downstream analysis methods.

As such, the staff reviewed this limited application of the LANCR02 code in accordance with the applicable review guidance of Standard Review Plan (SRP), Section 4.3 (SRP 4.3). The review procedures of SRP 4.3 (Reference 4) are selected and applied as appropriate for the particular review. In the subject matter, the NRC staff has identified those review procedures and acceptance criteria relevant to the review of analytical lattice physics methods. These are provided in SRP 4.3, III.7. Specifically, the staff review is based on the following considerations:

- The computer codes used in the nuclear design are described in sufficient detail to enable the reviewer to establish that the theoretical bases, assumptions, and numerical approximations for a given code reflect the current state of the art.

- The source of the neutron cross-sections used in fast and thermal spectrum calculations is described in sufficient detail so that the reviewer can confirm that the cross-sections are comparable to those in the current ENDF/B data files (i.e., ENDF/B-VII) and other sources of nuclear data, such as JENDL and JEFF3, etc. If modifications and normalization of the cross-section data have been made, the bases used must be determined to be acceptable.

- The procedures used to generate problem-dependent cross-section sets are given in sufficient detail so that the reviewer can establish that they reflect the state of the art. The reviewer confirms that the methods used for the following calculations are of acceptable accuracy: the fast neutron spectrum calculation; the computation of the uranium-238 resonance integral and correlation with experimental data; the computation of resonance integrals for other isotopes as appropriate (for example, plutonium-240); calculation of the Dancoff correction factor for a given fuel lattice; the thermal neutron spectrum calculation; the lattice cell calculations, including fuel rods, control assemblies, lumped burnable poison rods, fuel assemblies, and groups of fuel assemblies, and calculations of fuel and burnable poison depletion and buildup of fission products and transuranic isotopes.

3.0 TECHNICAL EVALUATION

3.1 LANCR02 Calculation Overview

The basic LANCR02 methodology includes several discrete steps to generate the downstream nuclear data. In the first step, the evaluated nuclear data were condensed from ENDF to fine group cross sections and made available as input to the LANCR02 code ~~LANCR02 condenses the evaluated nuclear data from ENDF to fine-group cross-sections~~. In the current method description, 118 groups are used to characterize the fine group nuclear data set. This external calculation is performed by condensing the ENDF data according to an assumed 1/E[1] spectrum assuming infinite dilution conditions. The infinite dilution fine-group cross-sections are then corrected using f-tables. These tables correct the infinite dilution cross-sections at a reference temperature according to the problem-specific dilution and material temperature.

In the second step, background cross-sections are calculated. The background cross-sections are calculated for use in the f-table correction factor. The background cross-sections are calculated according to the potential scattering and escape cross-sections. The potential scattering cross-section calculation is very straightforward. The escape cross-section is calculated according to a Wigner rational expression with a Bell factor correction for an isolated geometry. Dancoff factors are then calculated to adjust the escape cross-section. The final result is the background cross-section that is used in conjunction with the fuel temperature to interpolate the f-tables to extract the correct fine-group cross sections ~~that is used to interpolate the fine-group cross-sections according to the f-tables~~.

At the end of the second step in the calculation, LANCR02 has generated a problem-dependent nuclear data set of fine-group cross-sections. Prior to performing detailed transport calculations, however, the nuclear data set is further condensed to coarse group cross-sections. In the current methodology description, the coarse group library is based on 23 groups.

[1] 1/E is a common notation to denote that the magnitude of the flux spectrum varies inversely with the neutron energy. This notation is used throughout this safety evaluation.

The coarse group energy condensation is the third step. The energy condensation is performed by approximating the fine-group-wise neutron flux spectrum. The fine-group-wise flux spectrum is approximated using a two-step calculational methodology. In the first step of this methodology, a pin-cell calculation is performed based on a collision probability method. This method is based on a modified Wigner-Seitz geometric approximation. The next step is to perform a two-dimensional calculation that adjusts the pin-cell spectra to account for the spectral interaction effect of nearby cells, such as other fuel pins or water rods. The two dimensional calculation is performed by homogenizing the unit-cells and solving a simplified transport equation based on transmission kernels using the response matrix method. The unit cell spectra calculated using the homogenized two-dimensional solution are used to update the unit cell region specific spectra based on simple volumetric weighting.

The fourth step of the calculation solves the Boltzmann equation in the coarse group structure for the detailed problem geometry using a Method of Characteristics (MoC) approach. The MoC approach solves for the detailed region-wise fluxes for a fine spatial mesh using the problem two-dimensional geometry. Power distributions, kinetic parameters, and lattice eigenvalue calculations are ~~also~~ performed during the fifth step.

In the fifth step, the flux distribution generated by the MoC is used to determine the isotopic destruction and production rates in the regions of the problem. These calculations are combined with nuclear parameters such as decay constants in the depletion model to generate predictions of the material composition over incremental time steps. The LANCR02 calculations are then performed again based on updated material compositions at each exposure interval.

3.2 Scope of Review and Code Applicability

Section 7 of the model description LTR describes the types of fuel geometries that can be modeled with LANCR02. These descriptions address the various fuel lattice geometries, channel geometries, and control blade designs. The fuel types addressed include all fuel bundle geometries that are in current use in the domestic industry, including fuel assembly designs provided by other fuel vendors. Generally speaking, the MoC transport solution is a robust method that may be applied to various geometric configurations and the staff agrees that variations to those configurations provided in Section 7 should not present a challenge to the efficacy of the method.

Non-fuel cell geometries are also considered; these geometries include: vanished cells (coolant only), plena, small circular water rods, medium sized circular water rods, large (4 lattice locations) circular water rods, water boxes, and water diamonds. For the specific case of the water diamond geometry, LANCR02 approximates the water diamond as a water box. The staff review considered aspects unique to water cross geometries and the results presented in Section 5.6 of the model qualification LTR. The staff has found that these specific geometries afford LANCR02 the capability to explicitly model the key features of modern fuel designs currently employed in the operating fleet, as documented in Section 3.5.3.6.

The LANCR02 method also treats a variety of control blade designs. These may be either pin or plate type control blades with various absorber materials including enriched boron carbide or hafnium. Those designs currently employed in the operating fleet are explicitly treated by LANCR02.

Section 9 of the model description LTR describes potential updates to the LANCR02 method to implement advanced features that may be required to accommodate new fuel designs that originate from GNF or other vendors. The staff has separately considered the process for updating the LANCR02 capabilities in this regard and documented its findings in Section 3.6 of this SE. It suffices for the current application that the staff has considered the capabilities of LANCR02 relative to those fuel and control blade designs currently employed by the operating fleet. The staff found that the LANCR02 method employs sufficiently detailed modeling capability to address the unique features of the various fuel and control blade designs currently used.

The NRC staff requested additional information regarding any inherent limitations of the LANCR02 methodology (such as being unable to model non-symmetric fuel designs) and its range of applicability with respect to pertinent parameters (such as beginning of life (BOL) uranium enrichment and gadolinia concentration). GNF's response to Request for Additional Information (RAI) 1 (Reference 5) indicated that there are no inherent limitations in the parameters of interest within their physically realizable ranges or within the LANCR02 methodology with the exception of hexagonal fuel assembly geometry and problems in which transport corrected P0 cross-sections do not apply. As hexagonal fuel assemblies are not a design found within the light water reactor (LWR) fleet and use of P0 cross-sections is generally accepted as industry standard practice, the NRC staff finds the response acceptable.

GNF's response to RAI 1 also provided a list of restrictions for LANCR02, as specified in the user manual. The response included a description of the restrictions that stated they are for guidance to the user with respect to the range of LANCR02's application and do not imply modeling limitations. NRC staff observed that the listed restrictions do act to constrain the range of applicability to LWR designs. As a result, NRC staff finds the response acceptable.

3.3 Nuclear Data

3.3.1 Basic Nuclear Data

3.3.1.1 Neutron Cross Sections

The basic neutron cross-section data is the evaluated nuclear data file ~~(~~ENDF~~)~~/B-VII.0. The exception is for certain naturally occurring element cross-section tables – the source of these data is the ENDF/B-VI.8 libraries. These libraries are the most recent cross-section evaluations and are consistent with the state-of-the-art. Therefore, the staff finds that the basic nuclear data for the neutron cross-sections are acceptable.

NRC staff requested additional information regarding the source of the gamma cross-sections. GNF's response to RAI 3 included information that discussed the source of gamma cross-sections with respect to both the basic nuclear data and the processed nuclear data. The relevant information is discussed in the appropriate sections below.

3.3.1.2 Gamma Cross Sections and Spectra

As mentioned above, GNF's response to RAI 3 included information on the source of the gamma cross-sections. Specifically, GNF indicated that the continuous energy gamma cross-sections are expressed by formulae based on combined experimental and theoretical

data. The sources of the gamma cross-sections are the various expressions given in NEDE-23695, a reference provided by GNF in Appendix B of the model description LTR. NRC staff reviewed the portions of NEDE-23695 provided in the RAI response and found the formulae which produce the continuous energy gamma cross-sections to be acceptable on the basis that they: (1) are derived from generally acceptable first principles, (2) account for each of the most important gamma interactions of interest; pair creation, photoelectric effect, and Compton scattering, and (3) make the required corrections to formulae to account for energy dependent effects of the interactions. On a minor note, during review of the pair production cross-section discussion in NEDE-23695, Nuclear Reactor Regulation staff observed a typo in the
[[
]]. GNF acknowledged the typo in a conference call with NRC staff and confirmed the typo was not present in the actual code.

GNF's response to RAI 3 also stated that the fission and capture gamma spectra are taken from ENDF/B-V for the main nuclides such as uranium 235 and 238 and plutonium 239 and 240. Data sources for other nuclides are mixed amongst older releases of ENDF or JENDL, depending on the availability of the data for each. The staff finds the use of ENDF and JENDL fission and capture gamma spectra data acceptable on the basis that the libraries reflect the current state-of-the-art and/or remain applicable for use in the code. Additionally, the validity of the basic nuclear data sources is demonstrated through the general agreement of the results presented in the model qualification LTR test suites, which have been reviewed in Section3.5.

The Model Description LTR provides for updates to the basic nuclear data to be consistent with future releases of ENDF. This is consistent with maintaining the pedigree of the LANCR02 method as state-of-the-art. The model update section of the LTR states that such an update, however, will not be made if the code performance relative to the qualification basis is adversely impacted. The staff finds that this type of update: (1) does not constitute a change in the LANCR02 methodology, (2) ensures that performance of the code is not adversely affected by any updates to the nuclear data, and (3) ensures that the code remains consistent with the current state of the art knowledge regarding basic nuclear data. Updates to the basic nuclear data and the LANCR02 methodology are further discussed in Section 3.6.

On these bases the staff finds that the basic nuclear data selected for the code are currently acceptable and that the Model Description LTR provides a valid and acceptable approach for ensuring consistency of the basic nuclear data with the state-of-the-art.

3.3.2 Processed Nuclear Data

The methods used in LANCR02 require cross-sections for discrete energy intervals (groups) rather than continuous energy distributions for each nuclide and interaction. Additionally, cross-sections dependent upon varying physical properties of materials are required to analyze different or changing reactor conditions. For these reasons, it is required to preprocess the continuous energy cross-section data such that the appropriate groups are used and properties accounted for.

3.3.2.1 Neutron Cross Sections

For basic nuclear data, preprocessing calculations are used to condense the continuous energy cross-sections in ENDF to fine group (118 group) libraries. The method essentially condenses

the cross-sections using a "generic" 1/E neutron spectrum under infinite dilution conditions for a specific reference temperature (293 K). Adjustments to these fine group cross-sections that account for the effects of background cross-section and temperature are made using f-tables constructed during preprocessing.

The LTR did not directly address how H_2O and D_2O cross-sections were generated. Since these compounds play an important role in LWR analysis and their scattering cross-sections differ from their constituents, NRC staff inquired about them in RAI 2. GNF's response stated that, with the exception of the scattering cross-section, all the cross-sections are created assuming H_2O and D_2O are isotopic mixtures of O, H, and D. However, scattering cross-sections are determined for H_2O and D_2O compounds by considering the chemical binding effects on the constituent scattering cross-sections. Specifically, the LEAPR module of NJOY99 accounts for the change in coherent and incoherent elastic scattering data. Because the chemical binding effects are taken into consideration when determining the scattering cross-sections of the moderators, NRC staff finds the response acceptable.

The exact definition of the f-tables used by LANCR02 was not stated in the LTR. GNF's response to RAI 4 provides the definition as the ratio of processed self-shielded cross-sections at different temperatures and backgrounds to the base infinitely diluted cross-sections at 293K. The temperatures and background cross-sections range from 293K to 2,573K and 1E+2 barns to 1E+10 barns, respectively. NRC staff finds the f-table definition and range of temperature and background cross-section values acceptable.

The accumulation of these generically condensed cross-sections and the accompanying f-tables essentially provides the cross-section "library" from which LANCR02 can extract values for the specific temperatures and background cross-sections required in the analysis. The staff finds this approach for generating interpretable cross-section data for LANCR02 acceptable on the basis that the technique is generally considered industry standard.

3.3.2.2 Gamma Cross Sections and Spectra

As mentioned in Section 3.3.1.1, GNF's response to RAI 3 included information on the source of the gamma cross-sections with respect to the processed nuclear data. GNF's response stated that the final 8-group energy structure used in gamma calculations is too coarse to be generated directly from continuous energy cross-section data. Therefore, a two-step process is used to collapse the continuous data. First, 80 energy groups are determined with bounds chosen such that 10 of these "fine" groups correspond to each of the "coarse" groups that make up the final desired 8-group structure. The continuous energy data is collapsed to this 80-group fine structure using a simple group-wise average. The staff finds this method sufficient for collapsing the continuous energy data to coarse group structure since gamma cross-sections tend to be relatively smooth and the nature of the fine group structure ensures that unreasonable variations in cross-sections should not be present within a single fine group. Additionally, spectra-specific affects are not expected to be prevalent in such a fine group structure and are accounted for in the next step of the process.

After the first collapse, the 80 energy groups are further collapsed to the final 8-group structure using a gamma spectrum weighting. The gamma spectrum used is generated from a 1-D pin cell transport calculation. Since gamma spectra are mainly affected by heavy elements and are generally insensitive to coolant properties (i.e., void fraction), a single typical gamma spectra is

used to cover a wide range of reactor conditions. Due to the mentioned insensitivity, the staff finds this approach acceptable for determining the gamma spectra to be used for weighting. Further, the staff finds the use of gamma spectrum weighting to collapse from 80 to 8 groups appropriate since this technique is similar to what is used for collapsing neutron cross-section data, with the gamma and neutron spectra being analogous between the methodologies.

During this process, energy deposition cross-sections are also calculated for later use when calculating the energy deposited in the coolant, fuel, and structures from gammas. The energy deposition cross-sections are generated using a weighted sum of the effects described in the GNF referenced document, NEDE-23695, including pair creation, the photoelectric effect, and Compton scattering. The staff has reviewed the formulation of the energy deposition cross-section and has found it acceptable. Additionally, gamma energy deposition results are presented in Test Suite 17 of the Qualification LTR and have been reviewed by the staff in Section 3.5.3.17.

3.4 Model Description

Section 3.1 provides an overview of the calculations that are performed by LANCR02. The staff has conducted a detailed review of the specific models, equations, and approximations used to perform the overall calculation and documented its findings in this section of the Safety Evaluation (SE). The staff has reviewed the LANCR02 method in accordance with the review procedures and criteria specified in SRP 4.3.

3.4.1 Fine Group Cross Section Generation

The first step in the LANCR02 methodology is to calculate the fine group, infinitely dilute, microscopic cross-sections at a reference temperature. The initial calculation of the fine group cross-sections were reviewed by the staff as documented in Section 3.3.2 of this SE. Following this step, however, the cross-sections must be corrected to account for the actual background cross-section and temperature.

The temperature is an input to the LANCR02 code and its effect is captured in the f-tables implicitly, allowing LANCR02 to perform calculations over a wide range of temperatures. This was reaffirmed when NRC staff inquired about the effects of moderator temperature on determining nuclear parameters in RAI 22. GNF's response stated that moderator temperature is used to interpolate the temperature dependent moderator cross-sections at the given state point. This is an industry standard approach, and NRC staff finds it acceptable. Unlike correcting for temperature effects, the background cross-sections however, must be calculated based on problem-specific factors, such as geometry and composition, as discussed below in Section 3.4.1.1.

3.4.1.1 Background Cross Section

The calculation of the background cross-section considers the potential scattering and escape cross-sections. The treatment of the calculation of the ~~macroscopic~~ potential scattering cross-section using the nuclear data (i.e., hard sphere scattering radius) is acceptable. The calculation of the escape cross-section considers the region specific geometry and is based on the Wigner rational expression. The Wigner rational expression is used to determine the escape cross-section based on the inverse mean chord length for a simple convex body. In this

case, the geometry is assumed to be cylindrical. The staff finds this approach acceptable when appropriate corrections are made. In particular, the escape cross-section is corrected according to a Bell factor.

The Bell factor serves as an error correction to the Wigner rational expression for use in LWR applications. This approach is accepted as industry standard when using the Wigner rational expression. However, determination of an acceptable Bell factor for LANCR02 was not discussed in the LTR. GNF's response to RAI 7 indicated that in LANCR02, the bell factor is a parametric function weakly dependent on fuel temperature and pellet radii that is centered on typical ~~pressurized water~~light water reactor values (i.e., 1.16). Figure 2-1 in the response to RAI 7 demonstrates the validity of the bell factor function used in LANCR02 through comparison with the standard Wigner approximation and a fixed bell factor of 1.16 using the Carlvik two-term rational approximation as a benchmark. Additionally, the bell factor selected was validated by comparison to Monte Carlo N Particle (MCNP) results. Upon reviewing the comparison presented in the model description LTR and acknowledging GNF's conclusion regarding comparison to MCNP results, the staff finds the use and selection of the bell factor to adjust the Wigner rational expression for determination of escape probabilities acceptable.

In the calculation of the escape probability, an initial calculation is performed for a cylindrical geometry assuming an isolated fuel pellet. However, in reactor applications, the fuel design includes several tightly packed fuel elements in a regular lattice geometry. Therefore, the first flight collision probability approach that is the basis for the Wigner rational expression does not account for the very likely probability that a first flight neutron emanating from the fuel pellet will suffer its first collision in a neighboring fuel element. To correct for this effect, a Dancoff factor is calculated to adjust the escape cross-section. This correction accounts for the rod-to-rod shadowing effect of the lattice.

The Dancoff factor is calculated in LANCR02 by performing two transport calculations based on a fixed source distribution. In the first calculation, an isolated fuel pin is considered. In the second calculation, the entire lattice is considered. The purpose of the calculation is to determine the ratio of the difference in the surface current intensity for an isolated pin and an array of pins. For cases where the fuel pins can be considered black at resonance energies, this ratio collapses to the ratio of the flux differences.

The exact nature of how the isolated fuel pin model is constructed for the first transport calculation was not clear in the LTR. In their response to RAI 8, GNF stated that a square 5x5 array of cells is constructed for each representative fuel pin with the isolated pin being at the center of the lattice. All non-fuel regions of the pin are given a constant source. Likewise, the remaining cells of the lattice are filled with moderator and also given a constant source. Finally, the boundary conditions of the lattice are reflective. This is done to utilize the "Neutron Current Method" approach of determining Dancoff correction factors using the MoC, which is described in detail within Reference 6 of the LTR. This approach requires the isolated fuel pin neutron current being determined in an infinite moderator medium where the moderator is replaced with a constant source. The use of a reflective boundary condition would emulate an infinite moderator medium within a small 5x5 lattice. The reflective boundary condition is also typical of a typical transport solution calculation using the MoC. As such, NRC staff finds this approach acceptable.

The primary assumption made in this calculation is that the fuel pins are black (fully absorbing) at resonance energies, which is reasonable since the Dancoff factors are calculated for neutrons at resonance energies where absorption cross-sections are expected to be very high. This approximation is common in industry lattice physics analyses, and the staff finds this approximation acceptable on these bases.

The Dancoff correction factor based on detailed transport calculations using a fixed source is also applied to the correction of the escape cross-sections for non-fuel elements. This represents an extrapolation of the base method for fuel elements where the other regions (e.g., channel box or control blade) [[ ]]. NRC staff inquired about the validity of this assumption, especially concerning regions of zirconium. GNF's response to RAI 9 explains that this assumption is valid on the basis previously mentioned,
[[

]]. NRC staff finds this acceptable.

The staff finds that the approximate treatment of the isolated fuel element escape cross-section incorporates a reasonably accurate correction to the simplistic Wigner rational expression. Further, the staff finds that the approximations necessary in the derivation of the Dancoff correction factors are reasonably justified and consistent with the industry state-of-the-art in lattice analysis. On these bases, the staff finds that the LANCR02 method includes an acceptable process for determining the background cross-section for fine group cross-section interpolation.

3.4.1.2 Resonance Interference

Once the fine group microscopic cross-sections are corrected for the background cross-section, the cross-sections may be further treated to account for the effect of resonance interference. Resonance interference refers to the situation where a material composition of several isotopes includes nuclides that have resonances at similar energies. The neutron flux at energies around these resonances will be more depressed if one considers the synergistic effect of the two or more resonances. Therefore, these resonance interference effects ought to be treated in the microscopic cross-section formulation.

Upon initial submittal of the LTR, the LANCR02 methodology addressed resonance interference. The resonance interference phenomenon was treated by adjusting the isotopic microscopic cross-section by a Resonance Interference Factor (RIF). The RIF was determined on a problem-specific basis for each fuel region of the problem. Specifically, the approach relied on an ultra-fine group method that utilized point-wise cross-section data for several energy groups at a resolution much greater than the fine-group structure. The flux spectrum over the ultra-fine group structure was treated as approximately 1/E, and two spectra were calculated. The first spectrum assumed a finite dilution for each isotope, while the second was calculated considering all of the materials in the mixture. These two spectra were then used to collapse the microscopic cross-section over the ultra-fine group structure within each fine energy group, and the ratio of the two condensed cross-sections is the RIF.

However, GNF's responses to RAIs 10 and 11, which requested further details on the RIF, stated that the RIF model is not currently used in production cases, is not used in any of the

results contained in the LANCR02 qualification, and that NRC approval of the model was not being sought. Additionally, GNF modified the LTR to redact discussion of the RIF. Therefore, the adequacy and acceptability of the RIF model is outside the scope of this safety evaluation report. However, NRC staff review of the LANCR02 methodology was conducted to discern if any significant deviations in results would manifest from the model's exclusion. Upon examination of the results, the staff found the exclusion of the resonance interference model from LANCR02 acceptable due to the overall successful performance of the methodology over the wide range of tests presented in the model qualification LTR.

3.4.2 Coarse Group Cross Section Generation

While LANCR02 supports a capability to perform MoC transport calculations using the fine-group neutron cross-section sets, for practical applications for production purposes, the energy structure of the cross-sections is further condensed to a coarse group representation. The primary purpose for this coarse group condensation is to improve the run-time of the analysis.

The coarse group condensation must account for the neutron spectrum based on problem-specific considerations such as the lattice geometry. To account for these geometric effects, LANCR02 performs two simplified calculations to determine the fine-group-wise flux spectrum for cross-section condensation. One calculation is performed on the pin-cell level to characterize the local flux spectrum; and another calculation is performed to couple the local pin-cell to its lattice neighbors in order to account for spectral interaction on the spectrum at the local pin locations. These spectral interaction effects are pronounced for neighboring fuel elements of significantly different composition (e.g., gadolinia bearing rods near normal fuel rods), or where the local slowing down power may be significantly different (e.g., normal fuel rods near water rods).

3.4.2.1 Pin-Cell Calculation

Pin-cell calculations are performed to characterize the local flux spectrum. The basis for these calculations is a collision probability method (simplified transport) using a Wigner-Seitz geometric approximation. To perform these calculations, LANCR02 defines a multi-region one-dimensional (radial) problem.

The geometry is transformed into a co-axial cylindrical geometry. The transformation is performed according to the Wigner-Seitz approximation with modified boundary condition probabilities. These boundary conditions are often referred to as "white" boundary conditions, and they correct for inaccuracies introduced in the moderator region flux magnitude when perfectly reflective boundary conditions are assumed for the modified geometry. This approach has been widely adopted in standard applications, is reasonable for standard production purposes, and is acceptable.

The cell is comprised of the fuel, cladding, local coolant, and a surrounding buffer zone. The staff observed the gas gap between the fuel and cladding was not included. According to GNF's response to RAI 14, the gas gap is not explicitly modeled in LANCR02. Instead, the fuel material is defined to include the gas gap, filling the entire region inside the cladding inner radius, and preserving mass through density adjustment. GNF states that this approach is justified because the fuel expands to fill the gas gap during normal operation. Additionally, if the

fuel is assumed not to swell, the error introduced by the slight redistribution of mass is insignificant. On these bases, the staff finds this approach acceptable.

The spectral calculation is performed by driving the flux solution with a distributed fission source in the buffer zone outside the fuel-pin-specific cell region. The buffer zone is comprised of an average material composition representative of the entire lattice. The staff inquired as to the rationale behind using a material composition representative of the entire lattice as opposed to nearest neighbor. GNF's response in RAI 13 described that using a composition representing nearest neighbors is reasonable if the neutron mean-free-path does not extend beyond neighboring cells. However, the entire lattice is represented here, as opposed to nearest neighbors, due to the increase in neutron mean-free-path when dealing with voided conditions.

Additionally, this treatment better accounts for lattice-wide effects such as control blades or vanished pins. The approach can provide a more realistic fission source to drive the calculation and can therefore provide a better representation of the actual flux spectrum. On this basis, the staff finds the response acceptable.

The model description LTR states that this approach assists in rapidly converging the flux solution for pins without sources (i.e., water rods) or strong absorbers (i.e., gadolinia bearing rods). The staff agrees that this approach of providing a lattice-average external driving source would accelerate the calculation of the flux spectra under these conditions. For normal fuel pins, the inclusion of the buffer zone may improve the prediction of the flux spectra by incorporating, at the pin-cell level, some approximation of the external conditions. However, the precise magnitude of such an improvement is difficult to quantify, as two-dimensional calculations performed at later stages are intended to take into account the spectral interaction effects at a higher degree of detail. Therefore, the staff finds that the inclusion of the buffer zone assists in flux convergence for non-fuel pins or highly absorbing fuel pins and that its inclusion may improve the accuracy of the pin cell flux spectrum calculation. However, this may be lost in the correction to the pin-cell flux spectra that is performed at a latter point in the calculation.

Treatment of non-fuel pins, plenum regions, and other features such as control blades, water rods, water gaps, the channel box, water wings, and detectors is explained in GNF's response to RAI 15. Small water rods, fuel rod plena, and detectors are treated explicitly in the same way as fuel pins, only with the materials in the fuel region changed appropriately to represent the cell being modelled. Large water rods and control blades are split into arrays of individual small water rods or control pins and are treated similar to fuel pins. For control pins, however, an additional region between the outermost material and the buffer region is added containing the outer water gap cross-sections. Other features (i.e., channel box, water gaps, water wings) are treated with homogenized volume average cross-sections and are given the spectrum of the coolant of the corner pin cell. The geometric simplifications used preserve material volumes and emulate the lattice surroundings reasonably. Therefore, they are expected to provide flux spectra sufficient for the purpose of this calculation.

Additionally, the staff notes that the approximations made in this calculation do not apply to the final MoC solution and only serve to provide an estimated distribution used to weight the collapsed coarse group cross-sections. On these bases, the staff finds the approaches for modelling these non-fuel features acceptable.

Sources within the pin-cell calculation are assumed to be isotropic. However, the cross-sections are transport-corrected according to anisotropic expansion in Legendre polynomials to the first order. This is commonly referred to as the P1 approximation and is widely applied and accepted for standard reactor problems. The staff finds this to be an acceptable means for developing relationships to determine scalar flux magnitudes in the one-dimensional pin-cell calculation.

The P1 approximation allows for the determination of transmission kernels in one-dimension. The LANCR02 formulation of the transmission probabilities is based on a standard collision probability approach and consistent with general industry practice. When combined with the boundary conditions, a linear set of consistent group-wise equations for the scalar flux is generated. The scalar fluxes are then solved using standard matrix inversion techniques. The staff finds that the approximations and the solution technique at the pin-cell level are acceptable.

The LANCR02 formulation is substantially similar to previously approved pin-cell spectral calculation methods based on the collision probability method. In many ways, the LANCR02 pin-cell calculation is similar to the previously approved TGBLA06 (Reference 6) calculation with the novel feature that a buffer zone is included in the calculation. As stated above, the inclusion of the buffer zone improves convergence of the flux solution for non-fuel or highly absorbing pin-cells, and is likely a small improvement in the convergence of the coupled two-dimensional problem as it considers, in a lattice-average sense, the impact of spectral interaction at the pin-cell level. On these bases, the staff finds that the pin-cell calculations performed by LANCR02 are acceptable.

3.4.2.2 Two-Dimensional Coupling Calculation

A two-dimensional coupling calculation is performed using a simplified approach in order to account for the coupling between cells in the lattice to better represent the true surroundings of each pin-cell. The purpose of the calculation is to more accurately calculate the local flux spectrum in each lattice location.

To perform this calculation, LANCR02 homogenizes each pin cell based on the volumetric information from the geometry input and the flux spectra calculated during the pin-cell calculation. From these homogenized regions, a transmission probability technique is used to solve for the interface current coupling between the cells.

The transmission kernels and collision probabilities are tabulated and solved numerically using a ray tracing scheme. The current and flux solution is iteratively determined using a checker-board iteration scheme. The solution to the equations yields the group-wise scalar fluxes at each homogenized cell in the simplified two-dimensional geometry.

The one-dimensional fluxes are then updated to reflect the spectral interaction between surrounding cells (the coupling effect). The local flux spectrum is approximated by weighting the pin-cell spectrum for each region in the pin cell (fuel, cladding, coolant) based on the two-dimensional flux calculation. The weighting is performed by normalizing the group-wise region dependent pin-cell flux by the ratio of the product of the homogenized flux and volume to the summation of the product of the flux and volume of each region within the homogenized cell. This approximation of the corrected group-wise flux is given in Equation (27) of the model description LTR.

As stated in the LTR, several assumptions are required to perform this analysis. Particularly, to perform the simplified calculation, the surface currents are assumed to be isotropic. This assumption would not yield accurate results for a detailed transport calculation; however, the LTR concludes that this approximation is reasonable for the purposes of developing a condensation spectrum. To this end, the staff agrees that the current simplified approach provides at least a first order estimate (if not a more robust estimate) of the coupling effect between neighboring cells within the lattice. The use of transmission probability methods for spectral calculations on homogenized two-dimensional lattices is a common feature of modern lattice physics methods. The volumetric averaging approach to determine the pin-cell spectral correction factors is another simplifying assumption. However, with a sufficiently detailed multi-group structure, the errors introduced by these assumptions are expected to be minor.

GNF's response to RAI 16 addresses staff concerns about how the channel box and control blades are treated in the two-dimensional coupling calculation. Within the coupling calculation, the channel box is homogenized with the intra-assembly water gap (as opposed to the inter-assembly) between the box wall and the first row of pin cells. For control blades, homogenization is not performed between the controlled and moderator materials; any mesh cell that contains a portion of a control blade is assumed to be entirely controlled. Because a full transport calculation is performed without any homogenization, the overall effect of the channel box and control blade homogenization approaches on transmission probability calculation is negligible. On this basis, and because the channel box is optically thin, the staff finds the homogenization approaches acceptable.

For non-fuel regions, such as the inter-assembly water gap or channel box, the fluxes are determined directly from the two-dimensional calculation. For a control blade, a similar approach is used to correct the flux spectrum based on absorber tube pin-cell calculations and the two-dimensional flux solution for the homogenized cells representing the control blade (for controlled lattice calculations).

The staff finds that the approach is reasonable in terms of adjusting results to account for the prevalent phenomena associated with inter-cell coupling and has a reasonable technical basis. The accuracy of such an approximation, however, must be determined through a comprehensive qualification process. The staff review of the solution technique qualification is documented in Section 3.5 of this SE. Insofar as the method basis is acceptable, the staff finds that the approximations and assumptions are reasonable and that the calculational method is consistent with common practice within the industry.

3.4.2.3 Condensation

Once the pin-cell fine-group spectra are corrected for coupling effects, these spectra are used to condense the 118 fine-group cross-sections to a 23 coarse group (or multi-group) structure. The condensation is a standard flux weighting scheme and is therefore acceptable.

3.4.3 Void Distribution Model

LANCR02 includes a model for calculating the local void conditions for a given lattice average void fraction. The staff has previously reviewed this model during its review of TGBLA06. However, the staff sought confirmation of the model's proper implementation within LANCR02. According to GNF's response to RAI 12, the void distribution model is implemented differently in

LANCR02 than it was in TGBLA06. [[

]]. GNF also indicates that this approach accelerates the calculation as a whole. To validate the model's implementation in LANCR02, tables comparing distributed void fraction differences between TGBLA06 and LANCR02 for controlled and uncontrolled cases of a representative lattice on a pin-by-pin basis were presented. The root mean square (RMS) difference in void distributions between the controlled and uncontrolled cases was found to be [[ ]], respectively. Of note is that there is also an inherent difference in pin powers between TGBLA06 and LANCR02. The RMS of relative pin powers was quantified as [[ ]] for the controlled and uncontrolled cases, respectively. As a result, while the void distribution RMS differences are already reasonably small, the staff anticipates that these differences would be even smaller if the inherent difference in pin powers was not present. Therefore, the staff finds the void distribution model, as implemented in LANCR02, acceptable on the basis that the model itself has previously been reviewed and approved and that the implementation difference does not significantly impact the results of the model, as demonstrated by the RMS differences in void distribution and relative pin powers.

3.4.4 Detailed Two-Dimensional Transport Calculation

The basis for performing the detailed flux distribution calculation is the MoC. The method is based on solving the transport equation along characteristic rays that traverse the lattice geometry. The MoC is a standard approach for solving the flux distribution in modern lattice physics codes and is widely used for this purpose.

3.4.4.1 Method of Characteristics

The MoC technique is a sophisticated transport method whose accuracy is mostly constrained by the fineness of the spatial mesh and angular quadrature used. In this method, streaming rays are spaced along specific angular trajectories across the lattice and the Boltzmann equation is solved along these rays. Spacing these rays close together and having fine increments in both the azimuthal and polar angles in the quadrature closely approximates an explicit solution to the Boltzmann transport equation.

The staff reviewed the formulation of the Boltzmann equation along a particular ray to verify that the formulation was fully described and accurately discretized. Similarly, the staff considered the ray spacing to ensure that sufficient spatial resolution was afforded by the spacing to represent the model geometry with a modest error introduced due to geometric approximation. The staff also reviewed the azimuthal and polar quadrature to ensure that sufficient angular resolution has been provided.

3.4.4.1.1 Ray Tracing

The group wise, direction dependent flux solution is determined for each mesh cell according to a solution of the Boltzmann equation along particular rays. This formulation is provided in Equation (30) of the model description LTR. This equation calculates the change in the angular flux from the entry point of a ray to a mesh to the exit point of the ray through that mesh. The equation considers the transport corrected attenuation of the flux as well as source neutrons

produced within the mesh that contribute to the angular flux at the exit face of the mesh. The form of this equation is accurate when the source term is appropriately calculated.

Assuming isotropic sources, the source term is calculated as the sum of the group in-scattering term and the production term. According to GNF's response to RAI 17, the in-scattering term also includes the effects of (n, 2n) reactions. The production term is the summation of the fission neutrons born in the particular energy group considered.

The meshes are generated by the intersection of the rays with geometric features that define the detailed two-dimensional geometry, such as cylindrical interfaces defining the fuel pellet regions. The meshes are then approximated according to volume preservation to ensure that the mesh edges are treated as normal to the rays. With sufficiently fine ray spacing, volume preservation provides an appropriate basis for the geometric approximation without introducing significant error in the calculation.

The ray spacing must consider the conservation of particles in the problem. Since the infinite lattice calculations are performed with reflective boundary conditions, the MoC solution may be inaccurate if rays that approach the problem boundary do not have accompanying rays in the reflected direction. To address this concern, the problem-specified azimuthal angles and ray separation distance are adjusted to ensure perfect reflection. Section 3.1.3 of the model description LTR provides the equations that show how these angles and separation distances are slightly modified. The staff has reviewed these equations and determined that the method is appropriate and acceptable for determining a perfectly reflecting ray tracing scheme.

The flux solution is iterative in nature. Once the angular flux has been converged, integration is performed to calculate the scalar flux. This calculation uses transport-corrected cross-sections to account for anisotropic effects. The staff has reviewed this formulation and found that it considers appropriate sources and uses standard methods to correct for the assumption of isotropy. Therefore, the approach is acceptable to calculate the scalar flux distribution across the lattice meshes.

3.4.4.1.2 Quadrature

In the MoC solution, the quadrature represents a discrete set of angular directions (defined by both azimuthal and polar angles). The discrete directions are weighted according to their individual solid angle. The greater the number of discrete directions considered in the quadrature set, the more closely the MoC solution approximates the exact solution of the Boltzmann equation.

The quadrature is defined according to angular spacing in the azimuthal direction with corresponding angles specifying the polar directions that are explicitly treated in the numerical solution. The staff has previously discussed the fine adjustment to the azimuthal angular separations to address perfect reflection in Section 3.4.4.1.1. This section addresses the determination of the base azimuthal angular separation, which is slightly adjusted to ensure alignment of tracks along the problem boundaries. The azimuthal directions must span an entire circle (2π). This is intuitive for a reflected problem where the ray traces may encounter every boundary of the problem. [[

]]. For a

[[ ]], the directions are specified as the mid-point between (nearly) equally spaced angles and the weight is determined according to the solid angle.

The default number of [[ ]] was provided by the applicant in response to RAI 19. These defaults were determined through a sensitivity analysis on typical fuel configurations to produce results near the saturated solution. This default set is used for all production cases, but can be specified differently through input, if necessary. The staff finds the use of the default azimuthal angles acceptable for predicting an accurate solution to the Boltzmann transport equation via the MoC ray tracing scheme.

To complete the quadrature set, polar directions of motion, evaluated for each azimuthal angle, must also be specified. LANCR02 has multiple options for the polar quadrature sets. The Tabuchi-Yamamoto (TB) optimal quadrature sets, given in Table 3-1 of the model description LTR, provide three such options. Additionally, six separate Legendre quadrature sets are made available for use. [[

]]. NRC staff sought confirmation of this. Analyses provided by the applicant in response to RAI 20 show that the [[ ]] quadrature set most efficiently provides acceptable accuracy in the solution to the problem. It is demonstrated that only a slight increase in accuracy (on the order of 0-10 percent milirho (pcm) difference) is gained by using Legendre quadrature sets or by increasing the number of polar divisions in the TB sets. On these bases, the staff finds the use of the [[ ]] quadrature set (or one of the more detailed sets providing the same or better level of accuracy) acceptable for use in the MoC ray tracing scheme.

3.4.4.2 Fundamental Mode Calculation

LANCR02 includes a module that adjusts the MoC calculated flux in the lattice to simulate critical conditions. This adjustment is necessary because the MoC solution for the flux is obtained assuming reflective boundary conditions. In reality, leakage between assemblies occurs. The LANCR02 method includes a step to calculate the fundamental mode flux distribution using an artificial buckling that “leaks” neutrons into or out of the lattice such that the effective multiplication factor is unity. This simulates the flux that the bundle would contain in a critical reactor environment. By performing this correction to the flux, the exposure calculations more realistically emulate the expected power shapes for in-reactor operation.

To perform the fundamental mode flux calculation, the MoC coarse group cross-sections are expanded back to the fine-group energy structure. This allows for the calculation to more accurately account for energy dependent phenomena such as scattering and mean-free paths. Next, the fine group cross-sections are spatially homogenized over the lattice in order to maintain the radial flux shape, found by the MoC solution, throughout the fundamental mode adjustment. An energy independent buckling value is then found through iteration of Equation (34) in the LTR such that the obtained group-wise fluxes result in an effective multiplication factor that converges to unity in Equation (35). Upon convergence, the resulting fluxes are used to adjust the lattice-averaged MoC calculated flux spectra to develop the final lattice flux distribution. These final fluxes are used to perform power and depletion calculations.

In this sense, the fundamental mode flux is a spectral correction to the MoC solution to account for the propensity of higher energy neutrons to preferentially “leak.” The staff has considered

the effects of the fundamental mode flux calculation on cold shutdown margin cross-sections and controlled lattices. From GNF's response to RAI 21, it has been noted that the fundamental mode calculation is implemented on volume-averaged group fluxes and cross-sections. Therefore, the radial power shape determined from the MoC solution is retained throughout the calculation, ensuring that no biases will be introduced in cold shutdown evaluations as a result of power shape changes. The radial power shape will similarly be maintained for controlled lattices, making the calculation suitable for these cases as well.

The staff has reviewed this formulation and found that it considers appropriate sources and uses standard methods to correct for the assumption of reflective boundary conditions. The approach allows for more accurate cross-section generation for critical lattices while maintaining the underlying flux shape detail determined in the MoC solution. Therefore, NRC staff finds that the approach is acceptable to adjust the volume-averaged group fluxes and cross-sections.

3.4.4.3 Kinetics Parameters Calculation

LANCR02 also generates kinetics parameters for downstream calculations. The model description LTR describes the calculation of the delayed neutron fraction and prompt neutron lifetime. These calculations require the use of the adjoint flux. The adjoint flux is calculated by using the standard adjoint diffusion equation and the collapsed fundamental mode lattice-averaged cross-sections.

The adjoint flux is used to determine the effective delayed neutron fraction and prompt neutron lifetime. These calculations rely on the use of the adjoint flux to perform reactivity importance weighting of isotopic delayed neutron yields to determine the effective delayed neutron fraction for the system (in this case lattice). The formulation of the equations to determine these parameters is consistent with typical adjoint weighting schemes.

NRC staff has considered the effects of the fundamental mode adjustment on the formulation of the adjoint flux used to weight the kinetics parameters. In particular, a study in NUREG-CR-7164 (Reference 7) showed that appreciable error may be introduced in the calculation of beta effective when using the formulation in LANCR02. The NUREG suggests that the infinite (reflective boundary conditions) adjoint flux be used when calculating the kinetics parameters rather than the critically adjusted adjoint flux. In response to RAI 51, the applicant ensured that both methods of calculating the adjoint flux and weighting the kinetics parameters are available for use in downstream methodologies and the resulting flux distribution and cross-sections from LANCR02 are not affected by using one or the other. Additionally, similar variances between the two methods for calculating the kinetics parameters were seen when comparing LANCR02 and the NUREG report.

NRC staff has reviewed this formulation and found that it considers appropriate sources and uses standard methods to calculate the delayed neutron fraction and prompt neutron lifetime. The approach allows for use of either the critical or infinite adjoint flux weighting of the kinetics parameters, ensuring that the appropriate option is available for downstream applications. Additionally, the differing effects observed between the adjoint flux weighting methods are consistent with those observed in NUREG-CR-7164. Therefore, the staff finds the approach acceptable.

3.4.4.4 Power Distribution

The power distribution is calculated for the lattice using the neutron and gamma flux solutions. The staff review of the gamma flux solution and gamma smearing is documented in Section 3.4.6 of this SE. The power distribution calculation considers heat sources from: (1) kinetic energy deposition from fission products, (2) gamma heat deposition, and (3) beta particle energy deposition. Energy deposited due to the loss of epithermal neutron kinetic energy during capture was considered, but was found to be insignificant when compared to the three sources listed above, and is therefore not treated in LANCR02.

NRC staff found that these sources comprehensively treat all significant sources of recoverable fission energy and take into direct account heat contributions from delayed sources. The staff considered the formulation of each power calculation. The equations are derived from first principles and are straightforward once the flux solution has been determined. Therefore, the staff finds these reasonable and acceptable.

3.4.4.5 Detector Response

LANCR02 incorporates two different methods by which a physical detector can be modeled: "phantom" and "explicit." [[

]]

The staff generically reviewed the detector models noting that the solution techniques to determine the respective flux solutions are largely the same with variations arising from differences in the cross-section data used. [[

]] This pin-cell calculation is the same as the methodology used to analyze plena pin cells except that the number of regions is expanded to account for the detector materials (anode/cathode, insulation, and cover tube). This approach is as equally acceptable as the pin-cell homogenization technique for intra-lattice cells.

NRC staff noted that in the LANCR02 methodology, [[

]] A comparison between the J-factors obtained from both detector models for each detector type (thermal traversing incore probe (TIP), gamma TIP, gamma thermometer) is given in GNF's response to RAI 26 to address this concern. The response indicates that, [[

]] Plots of normalized relative difference in J-factor between the phantom and explicit models versus burnup show that [[

]]. Due to this consistency, the staff finds the use of either physical representation of the detector acceptable.

The MoC calculation and gamma transport calculation (Section 3.4.6) determine the detailed neutron and gamma flux distributions for the instrument location, in the southeast corner cell of the lattice. [[
]].

[[

]]. This approach is substantially similar to the J-factor methodology employed by the previously approved TGBLA06 code to perform instrument response calculations and therefore, the staff finds the approach acceptable.

[[

]] This calculation is substantially similar to the approach adopted in the previously approved TGBLA06 method for calculating gamma-sensitive detector J-factors. Additionally, the gamma energy deposition comparison presented in Test Suite 17 of the model qualification LTR can serve as verification of LANCR02's gamma flux determination ability due to the parameter's direct relationship. The staff review of the comparison is detailed in Section 3.5.3.17. On these bases, the staff finds the approach acceptable.

3.4.5 Depletion

The depletion model in LANCR02 is used to simulate the evolution of the lattice nuclide inventory during a set of historical conditions input in by the user. These historical conditions may include continuous or variable control state and void conditions. The basis for this model is to use power distributions calculated by the transport method along with power density to calculate the destruction and production rate of various nuclides in the fuel.

The destruction mechanisms include neutron absorption, (n,2n) reactions, and radioactive decay. The production mechanisms include lower isotope absorption, radioactive decay, and fission yield. The destruction mechanisms include branching ratios to account for the relative probability of neutron capture to (n,2n) reactions. Based on these mechanisms, a system of differential equations for each isotope can be written. These equations are then discretized and solved using a 4^{th} order Runge-Kutta-Gill algorithm. The Runge-Kutta-Gill algorithm is commonly employed for performing the depletion calculations in lattice physics analysis and is consistent with the TGBLA06 code, previously approved by the staff. On this basis the staff finds that the solution method is acceptable to determine the change in the nuclide inventory for a sufficiently small time step.

The depletion chains provided in the model description LTR illustrate the various production and destruction mechanisms for the actinides, fission products, and absorbers considered in the LANCR02 calculation. LANCR02 considers a significant number of important isotopes. Where isotopes are not explicitly modeled, they are tracked as part of a pseudo fission product. This approach is common, as it is not feasible to consider every isotope in these calculations. The

staff reviewed the depletion chains to confirm that the tracked nuclides include the major actinides, strong neutron absorbers (and their precedent nuclides) and relevant absorbers, such as gadolinium and boron. The staff compared the detailed depletion chains to those previously approved by the staff for use in TGBLA06 and found that the LANCR02 included greater detail in the number of isotopes that are explicitly tracked. On these bases, the staff finds that the depletion chains are sufficient and acceptable.

A time step must be specified for the depletion analysis. This time step is calculated based on a given incremental exposure step, power density, and fuel density. Generally, the model description LTR states that an exposure interval of 2 GWd/ST is acceptably small to simulate the nuclide evolution for normal fuel pins. However, for fuel pins loaded with considerable burnable poison absorber, very small exposure increments are required.

The smaller exposure interval is required because the gadolinia depletion rates are sensitive to the spatial distribution of the flux within the gadolinia bearing pin, and the spatial distribution, is likewise very sensitive to the concentration. Gadolinia bearing fuel rods tend to exhibit an "onion skin" effect where the outer regions of the fuel pin shield the inner regions. Calculating the depletion rates without taking this spatial coupling into account can result in significant errors.

[[

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However, this approach requires the use of very short exposure intervals (0.2 GWd/ST).

LANCR02 includes a dual time step model to address burnable poison rods. The LANCR02 methodology iterates between the depletion model and the flux solution typically at each 1 GWd/ST to establish the change in flux distribution resulting from nuclide depletion. This exposure increment is sufficiently small that the flux distribution in normal fuel pins are expected to be relatively constant and any errors introduced by the time discretization are minimal. However, gadolinia bearing pins require special treatment to allow the larger exposure increment in the analysis methodology.

The gadolinia bearing pins are depleted using a second, smaller exposure increment (0.2 GWd/ST). However, at the end of each depletion calculation, according to this smaller exposure increment, a full lattice transport calculation is not performed. Therefore, five gadolinia pin depletion calculations are performed for each whole lattice transport calculation. To account for the evolution of the spatial flux during the intermediate exposure increments, however, LANCR02 performs one-dimensional pin-cell calculations for the updated gadolinia bearing rod compositions. This calculation generates updated pin-cell fluxes. The updated pin-cell fluxes are combined with the two-dimensional coupling and MoC solutions of the previous detailed transport calculation to adjust the pin-wise fluxes for the gadolinia rods. This approximates the local variation in the flux over the short exposure interval assuming that the overall lattice distribution is not significantly perturbed. This is a reasonable approximation, as the other fuel pins do not exhibit significant changes in nuclide inventory over these short exposure intervals and the gadolinia bearing pins remain strongly shielded. Therefore, the staff

finds that the dual time step model is sufficient to allow for long exposure intervals of 1 GWd/ST between lattice MoC calculations while still accurately treating the gadolinia bearing fuel pins with finer detail.

The dual time step model is deactivated once the strongly absorbing gadolinium isotopes have been sufficiently depleted from all of the pins in the lattice. Control blade depletion capabilities are present in LANCR02, but GNF stated in response to RAI 29 that these capabilities are not currently used and approval for them is not being sought. Approval to use LANCR02 for control blade worth calculations or other applications that rely on control blade depletion would require a future submittal.

3.4.6 Gamma Transport Calculations

LANCR02 includes a detailed gamma transport model. The gamma transport calculation references the same MoC solution technique to determine the gamma flux distribution in the mesh. The primary difference is that the equations are solved based on the group-wise gamma interaction cross-sections. The gamma sources are calculated according to gamma rays emitted as a function of neutron capture and fission. The staff notes that the calculations are performed at steady state conditions so the delayed and prompt gamma sources are treated simultaneously.

The gamma transport model is used to calculate local gamma fluxes for two purposes: (1) gamma smearing of the radial power distribution, and (2) calculating the detector response.

3.4.6.1 Gamma Smearing

Gamma smeared power distribution is calculated by predicting the distribution of the gamma flux over the lattice. The model then calculates the energy deposition in each problem region by combining the region volume, local gamma flux, and the gamma energy deposition cross-section and integrating over energy. The gamma energy deposition rates are combined with other heat sources, such as the fission product kinetic energy, to determine the smeared power shape. The method is reasonable and the staff finds it acceptable to account for heat deposition in fuel structures due to gamma radiation.

3.4.7 Acceleration Techniques and Numerical Convergence

LANCR02 employs spatial and energy acceleration techniques. These techniques utilize rebalancing approaches to update the fluxes at the end of the inner iteration. The basis for the spatial technique is a coarse mesh rebalance and the basis for the energy technique is a fundamental mode rebalance. In either approach, a simplified calculation is performed based on the results of the most recent iteration to generate an updated flux for the next inner iteration and is intended to allow the calculation to approach the converged flux solution with fewer outer iterations. The staff has reviewed these acceleration techniques and found that they do not adversely affect the execution of the basic models and equations, but do assist in updating the flux at each iteration to, essentially, improve the “guess” of the converged flux solution. On these bases, the staff finds that the use of these acceleration methods is acceptable.

The LANCR02 convergence criteria are specified as a tolerance of change in the angular and scalar flux and eigenvalue at the end of each iteration. When the change is less than this

specified tolerance, it is determined that the solution has converged. The default convergence criteria, specified in GNF's response to RAI 31, was used in all of the model qualification analyses, demonstrating its validity. Additionally, this criteria is recommended to the user for use in production work. Use of convergence criteria greater (more relaxed) than the default recommended value would require demonstration that the accuracy of LANCR02 is not unreasonably affected throughout its range of applicability. The staff has reviewed the default convergence criteria and found that it is adequate, as is shown in the model qualification. Additionally, the staff believes that the convergence criteria remains adequate for production cases. The staff finds the use of lower (stricter) convergence criteria acceptable, if required. The use of a greater convergence criteria requires justification and adequate demonstration that accuracy is not unreasonably affected.

3.5 Model Qualification

The LANCR02 model qualification LTR describes the approach for qualifying the LANCR02 method for lattice physics analysis. The LANCR02 lattice physics methodology performs infinite lattice calculations. Therefore, these calculations are not compared directly to critical experiment data or in-reactor measurements. In the current approach, the LANCR02 method is qualified using an intermediary code, MCNP-05P. MCNP-05P is based on the MCNP5 code developed by Los Alamos National Laboratory (LANL) with some modifications to account for delayed gamma and beta energy deposition.

The Monte Carlo N Particle (MCNP) approach is based on direct simulation of individual particle transport trajectories. Several simulated trajectories are used to establish the detailed neutron flux distribution and eigenvalue of arbitrary geometries. The accuracy of the prediction is a function of the number of simulated trajectories.

According to the qualification LTR, the MCNP code that is referenced provides a means of qualifying the LANCR02 solution technique. The qualification approach is to perform qualification of the MCNP-05P code against critical experiments using the general geometric modeling capability of MCNP. Then the LANCR02 infinite calculations are compared to equivalent infinite calculations performed with MCNP-05P.

The MCNP method, with sufficient particle trajectories, provides a robust and detailed solution. The accuracy of MCNP has been justified, given its qualification against the relevant critical experiments, as discussed at greater length in Section 3.5.2 of this SE. Therefore, the staff finds that this approach is an acceptable means of bridging the available experimental qualification data to the infinite calculational method in LANCR02.

3.5.1 Qualification Codes

MCNP-05P is the primary code used in the LANCR02 qualification. This version of MCNP is largely similar to the MCNP code developed by LANL with some modifications made by GNF. The primary differences are in the calculation of delayed gamma sources and the calculation of beta energy deposition. The staff reviewed these model enhancements and found them to be straightforward modifications to improve accuracy in the calculation of heat deposition. Therefore, the staff finds that these models are acceptable.

With the exception of the depletion calculations, the comparisons were carried out using the MCNP-05P code. The depletion calculations were performed using the MCNPX code, which is another version of the MCNP code released by LANL. The staff finds that the modifications made in MCNP-05P are peripheral to the calculation of the flux distribution. For the depletion capability, the purpose of the qualification essentially tests the efficacy of the code to accurately predict the neutron flux distribution and subsequently the reaction rates that allow for the calculation of the isotopic evolution during irradiation. Therefore, the staff finds that either code version is acceptable for this purpose.

Depletion qualification was performed using comparative calculations between LANCR02 and MONTEBURNS. MONTEBURNS is a code that couples ORIGEN zero-dimensional depletion calculations with MCNP transport calculations. The MONTEBURNS code used by GNF has been modified slightly to include 200 depletion regions. The depletion calculations are performed by utilizing flux distribution and cross-section data generated using MCNP calculations to drive material depletion calculations using ORIGEN and associated half-life data. MONTEBURNS calculations are widely used to qualify depletion analysis methods, on this basis, the staff finds the approach acceptable.

3.5.2 Comparison to Critical Experiment Data

The LANCR02 model qualification LTR describes the various critical experiments that were used to qualify MCNP with the ENDF-B/VII.0 nuclear data library. The experiments serve not only to demonstrate the computational efficacy of the MCNP transport solution, but also to demonstrate the adequacy of the shared nuclear data library. The staff reviewed the scope of the critical experimental data to ensure that it was representative of the expected application range for lattice physics analyses for boiling-water reactors (BWRs). Additionally, the staff reviewed the robustness of the MCNP solution and the ENDF-B/V.II.0 nuclear data to represent realistic reactor configurations in terms of reactivity and power distribution.

The critical benchmark data presented in the qualification LTR considers a variety of data. In the case of the Jersey Central critical experiments, the staff has previously reviewed these data, as they were referenced in the initial qualification of the GEBLA lattice physics code (Reference 8). The KRITZ-75 experimental data were collected during the 1970s at the KRITZ reactor in Studsvik, Sweden and data from these reactor critical experiments have been referenced in other LTR reviews (e.g., Reference 9). In addition to these data, the model qualification LTR considers more recent data collected at the Toshiba Nuclear Critical Assembly (NCA) tests in the mid-1990s for 8X8 and 9X9 fuel designs and in early 2000 for 9X9 fuel, and a series of international benchmark data. The staff considered each experimental data set separately.

3.5.2.1 Jersey Central

The Jersey Central critical experiments were performed at the GE Vallecitos Nuclear Center and consist of a 4X4 bundle arrangement of 7X7 fuel. Two tests were performed. In either case criticality was achieved by adjusting the water level in the experiment. One test considered an uncontrolled configuration while the second test included the presence of a borated stainless steel curtain. Both tests were performed under cold conditions. The comparison of the critical eigenvalue to MCNP calculations indicates very good agreement. The average bias is [[ ]].

3.5.2.2 KRITZ-75

The KRITZ-75 critical experiments were similar to the Jersey Central experiments in that the critical assembly was comprised of a 4X4 bundle array. The fuel assembly design was based on an 8X8 fuel rod array. Unique features of the KRITZ-75 tests include the presence of various loadings of gadolinia poison in a number of fuel rods. The gadolinia loading in the poisoned rods varied between 3 weight-percent (w/o) and 5 w/o. In addition to the inclusion of gadolinia bearing fuel rods in the assemblies, the KRITZ-75 tests simulated hot zero power conditions by varying the critical assembly temperature between room temperature and 516 K (243 °C).

The temperature range considered temperatures that are near the saturation temperature for water at typical BWR operating pressures (at 7 megapascal (MPa) the saturation temperature is 286 °C). A total of nine experiments were performed. These tests considered the three temperatures and three variations in the gadolinia poison loading.

The results of the MCNP comparisons are provided in Table 4.2-1 of the model qualification LTR. The average multiplication factor calculated by MCNP indicates a small bias of approximately [[ ]], while the standard deviation of the results is very small [[ ]], particularly considering inherent limitations on the accuracy of the MCNP results (maximum MCNP uncertainty for these cases was 53 pcm).

3.5.2.3 Toshiba NCA

Critical tests were performed at the Toshiba NCA in the mid-1990s for the Step II (8X8 fuel) and Step III (9X9) fuel designs and in early 2000 for the GNF1 (9X9) fuel design. The NCA consists of a driver region surrounding a central test section. The test section is a 2X2 bundle array. The NCA facility includes the capability to simulate the presence of in-channel water voids through the inclusion of hollow aluminum tubes in the regions between the fuel rods. Figure 4.3-2 of the qualification LTR provides a schematic illustrating the locations of the test assemblies, the driver assemblies, and the aluminum tubes. The Step II, Step III, and GNF1 fuel designs include internal water rods, making these more representative of modern fuel lattice geometries.

In addition to the critical measurements performed at NCA, several gamma scan measurements were performed for the GNF1 fuel. These gamma scan measurements were performed axially over several fuel rods to characterize the local axial power distribution. Radial gamma scans were also performed over a short axial section for several rods at the NCA mid-plane. These data allow for direct comparison of the MCNP predicted local power distributions to measurements in a critical facility. This feature of the NCA critical tests makes these data particularly useful in the qualification of the MCNP method in terms of its prediction of power distribution.

In total, 15 critical measurements were performed at the NCA for the Step II and Step III designs and 37 measurements for GNF1. The results for the 52 critical measurements and the MCNP calculations are provided in Table 4.3-1. The results indicate a mean bias in the MCNP results of approximately [[ ]] with a narrow standard deviation of [[ ]]. The results of the MCNP calculations are consistent over a wide range of water rod configurations, void conditions, and temperatures. No biases were observed for the various critical conditions. The eigenvalue bias is consistent in magnitude with the biases observed for the ENDF-B/V.II.0

nuclear data for the earlier critical tests and the standard deviation is quite small, considering inherent variation in the MCNP results.

Axial measurements were made for a subset of pins for two of the GNF1 critical tests. For these comparisons the MCNP calculations were within [[ ]]. This indicates excellent agreement between the calculations and the measurements. This difference includes associated measurement uncertainty and MCNP uncertainties (statistical and nuclear data), and therefore, the staff considers this degree of agreement to be very good.

Radial measurements were made at the mid-plane. The results for pin-wise gamma scans on a large number of pins (approximately [[ ]] for 3 separate critical tests) indicate very good agreement in the pin power distribution. The fission rate RMS differences were [[ ]]. Again, considering measurements, cross-section, and MCNP uncertainties, the staff considers this agreement to be very good.

The NCA critical tests were performed for fuel designs with modern features, such as gadolinia bearing fuel pins and water rods, and considered conditions of void fractions up to 40 percent, as well as higher temperatures. The tests included pin-wise axial and radial gamma scans. Therefore, the staff considers the NCA critical tests to constitute a robust test of the MCNP capabilities in terms of eigenvalue and power distribution calculations with particular relevance to the anticipated application range for BWR lattice calculations.

3.5.2.4 International Benchmarks

The International Criticality Safety Benchmark Evaluation Project (ICSBEP) includes numerous critical experimental data for the purpose of benchmarking critical safety analysis tools. In the case of the MCNP qualification, GNF identified a set of ~~270~~ 127 critical evaluations from the ICSBEP critical benchmarks relevant to BWR applications. MCNP comparisons were performed using the ENDF-B/V.II.0 nuclear data set. The results indicate an average bias of approximately [[ ]] in the MCNP calculated multiplication factor and a standard deviation of [[ ]]. The standard deviation is slightly larger than those for the other critical data sets. However, some degree of increased variability is expected based on the wide range of spectral conditions considered in the ICSBEP critical benchmarks (the water to fuel ratio ranges from 1.50 to 4.22).

The ICSBEP critical benchmarks provide an industry accepted set of benchmarks for computational efficacy assessment for criticality codes and is widely used for this purpose. Therefore, the staff finds that this provides a basis for the qualification of the MCNP eigenvalue calculations for low enriched uranium, water moderated reactors. The results of these MCNP comparisons indicate small biases and standard deviation relative to the critical benchmarks. No strong discernable biases were observed as a function of the neutron spectrum, indicating robustness in the solution method and cross-section data.

3.5.2.5 Critical Benchmarking Conclusions

The MCNP transport solution is a highly robust computational approach to solving the neutron transport problem. The solution accuracy is essentially limited only to the cross-section uncertainties and the number of particle trajectories simulated in the problem. The MCNP code and the associated nuclear data library were tested against relevant critical assembly data and international benchmarks. The results of these comparisons indicate good agreement between

MCNP and the data. Some biases in the multiplication factor were observed when MCNP was exercised with the ENDF-B/V.II.0 nuclear data library. However, these biases were relatively small (on the order of 100 to 300 pcm). Standard deviations relative to the critical measurements for Jersey Central, KRITZ-75, and NCA Step II, Step III, and GNF1, were small. These results indicate consistent accuracy in the solution technique and an acceptable range of application for the nuclear data library. These results are consistent with the performance of MCNP with the ENDF-B/V.II.0 nuclear data set against the ICSBEP benchmarks.

The staff has reviewed these qualifications and agrees that the ENDF-B/V.II.0 is well suited for application to BWR fuel lattice analyses. The biases observed in the eigenvalue determinations are sufficiently small that the staff is reasonably assured that downstream infinite lattice eigenvalue predictions based on these data are acceptable for use in neutronics codes. The staff, of course, notes that uncertainties and biases in the calculational methods will be quantified and considered in the development of appropriate operational and safety limits.

The staff agrees with the overall assessment of MCNP insofar as that the bias is sufficiently small that results of the MCNP calculations with the ENDF-B/V.II.0 can be reasonably considered best-estimate when compared to typical biases and uncertainties in lattice methods.

Therefore, the staff finds that the qualification provides a reasonable basis for the acceptance of MCNP with ENDF-B/V.II.0 cross-section data for qualification of the LANCR02 infinite lattice code solution technique. The qualification also provides a reasonable basis to justify the selection of the ENDF-B/V.II.0 cross-section library as the nuclear data set for LANCR02 calculations.

3.5.3 Computational Benchmarking

The computational benchmarking between LANCR02 and MCNP serves to test the capability of the LANCR02 solution method. Both MCNP and LANCR02 reference the same nuclear data. The qualification of MCNP indicates that the solution technique is highly accurate and confirms the applicability of the nuclear data set. The computational benchmarks, therefore, serve to illustrate the capability of the LANCR02 solution technique relative to the highly robust MCNP technique. This approach quantifies any source of uncertainty attributable to assumptions or approximations in the LANCR02 methodology.

The basic approach in the computational benchmarking analysis is to test the capability of LANCR02 to analyze a variety of fuel lattices, to identify subgroups of lattices that characterize particular conditions, such as high void fraction, control state, boron concentration, etc., and to develop a trend criterion. The trend criterion is a measure of calculation differences between MCNP and LANCR02 that would indicate an adverse trend in the performance of LANCR02 to analyze a particular lattice type or condition. The staff has reviewed this approach and found that with sufficient subgroup categories it provides an acceptable means to identify any adverse trends in the LANCR02 solution performance.

To establish the baseline performance of LANCR02, a baseline test suite was developed that considers the standard application of LANCR02 to the primary fuel products developed by GNF. This test suite serves as a baseline to characterize the LANCR02 performance and to judge the relative performance for lattice subgroups. In determining the trend criterion of [[ ]], GNF compared this criterion to the performance of LANCR02 for the baseline test suite. The

baseline test suite results show a standard deviation in the eigenvalue of [[ ]]. Therefore, GNF has conservatively selected a trend criterion that is less than two standard deviations. The staff reviewed this basis and the magnitude for the trend criterion and determined that the trend criterion is conservative and acceptable to identify any applications where LANCR02 performance is not consistent with the baseline results.

GNF developed 17 test suites. These suites address lattice subgroups with particular lattice perturbations. These perturbations within the test suites allow for comparison of the testing results against the trend criterion to determine if any applications indicate degraded performance of the LANCR02 solution methodology. These suites consider variation in: lattice type, edge part length rods, mixed oxide fuel, depletion, alternative product lines, gadolinia concentration, placement of gadolinia pins, enrichment, boration, alternative control blade designs, Doppler, off-normal void concentration, channel bow, and fuel rod variation. In addition to these perturbation type suites, two additional test suites were considered to address neutron balance and gamma transport. An additional benchmark was provided to independently compare the depletion solution against the MONTEBURNS method.

The staff reviewed the scope of the test suites and found it to be sufficiently comprehensive to test for trends in the application range for LANCR02. In some cases, the staff found that the test suites considered ranges beyond the scope where the staff expects LANCR02 to be applied. For example, the enrichment test suite considers high uranium enrichments (above 5 percent).

Within each test suite, the qualification benchmark analyses also compare the fission density differences between LANCR02 and MCNP on a pin-wise basis and also consider the differences in pin peaking factor. These parameters are often relied upon in the nuclear analyses to reconstruct the detailed core power distribution and are important in assessing thermal margin in the core simulator. To this end, the staff finds that the eigenvalue and pin power distribution comparisons provide an acceptable qualification approach to demonstrate the capability of LANCR02 to predict those nuclear parameters that are relevant in downstream safety analysis. On these bases, the staff finds that the qualification approach is comprehensive and acceptable.

As each test suite is designed to evaluate the potential for calculational biases (trends) with given lattice conditions, the staff reviewed the results of each test suite separately, as described in the following sections of this SE. The staff reviewed each suite to determine if pin power distribution calculations were in good agreement and if calculational results for any subgroup exceeded the conservatively established trend criterion.

3.5.3.1 Baseline

The baseline test suite considered primary GNF fuel product lines: GE9, GE11, and GE14. To generate the test suite, GNF considered prototypical bundles and analyzed each lattice comprising the bundle design. Normal parameter variations were considered that are analyzed to generate data for downstream core simulator analyses. These variations consider: control state, xenon state, temperature, and void fraction. All of the calculations were performed for a C-lattice configuration with a D100 control blade design. Variations in control blades and lattice configuration were considered as part of other test suites.

The staff reviewed the results including the figures presented in the model description LTR to indicate any trends with void fraction or lattice size (8X8, 9X9, or 10X10). A summary of the comparisons is provided in Section 5.1.5 of the model description LTR. The overall baseline test suite indicates a very small eigenvalue bias [[ ]] relative to MCNP, indicating no meaningful bias. The standard deviation in the entire baseline suite was [[ ]]. This indicates very good agreement between MCNP and LANC02 for eigenvalue determination.

The staff reviewed the summary of the subgroup comparisons and agrees with GNF's determination that no trends are observed in the baseline test suite. Further, the staff reviewed the fission density and pin peaking comparisons. The staff found that the agreement between LANCR02 and MCNP was excellent. The maximum fission density RMS difference remained below [[ ]], which is considered excellent agreement. For the pin peaking factors, only the hot cases were considered. The hot cases are consistent with the potentially limiting conditions for high pin power, and the staff found that this basis was appropriate for the consideration of the pin peaking statistics. The results also indicate excellent agreement with very small biases and standard deviations amongst the entire data set, as well as all of the subgroups that were considered. Table 5.1-10 of the qualification LTR summarizes the pin peaking factor statistics and demonstrates very small biases [[ ]] and very small standard deviations [[ ]]. Again, the staff finds that these comparisons indicate excellent agreement between LANCR02 and MCNP.

Doppler worth comparisons indicate agreement within approximately [[ ]] and control blade worth comparisons indicate agreement within approximately [[ ]]. These comparisons show very good agreement.

The staff review of the baseline suite confirms the applicability of LANCR02 to the primary GNF fuel products over the standard range of application in terms of control state, void fraction, temperature, and xenon concentration. The biases were demonstrated to be very small for both eigenvalue and pin power distribution. The standard deviations were also small, in the range of [[ ]] for eigenvalue and [[ ]] for pin power. Overall, the staff agrees with GNF's assessment that the agreement is excellent. On this basis, the staff is reasonably assured in the adequate applicability of LANCR02 to analyze standard GNF fuel products.

3.5.3.2 Variation in Lattice Type

This test suite was developed to examine potential trends in LANCR02 for various lattice configurations. The baseline test suite considered only a C-lattice configuration. Comparative calculations were performed for various nodal conditions for the higher powered regions of the prototypical bundle designs for D-lattice and N-lattice configurations.

Rod worth and Doppler worth comparisons indicate consistent performance across the various lattice types with the performance of LANCR02 demonstrated for the baseline test suite.

The staff reviewed the summary statistics and trend plots. The staff agrees with GNF's assessment that no trends are observed for any subgroup. The statistics indicate excellent agreement between LANCR02 and MCNP and demonstrate consistent performance with the baseline test suite.
On these bases, the staff finds that the LANCR02 capability to analyze various lattice configurations has been adequately established.

3.5.3.3 Edge Part Length Rods

This test suite was developed to examine potential trends in LANCR02 for various fuel designs with different numbers of part length rods on the edge of the lattice. The test suite considered GNF1 and GNF2 bundle designs with different part length fuel rod configurations.

Rod worth and Doppler worth comparisons indicate consistent performance across the various edge part length rod arrangements with the performance of LANCR02 demonstrated for the baseline test suite.

Figure 5.3-3 of the model ~~description~~ qualification LTR illustrates some test suite cases with pin peaking factor differences [[

]] the staff notes that these observed differences are minor and essentially negligible. The overall agreement between MCNP and LANCR02 remains excellent with pin peaking factor statistics indicating agreement [[ ]].

The staff reviewed the eigenvalue statistics provided in Table 5.3-8 and determined that no observable biases were found across the subgroups. The staff notes that the [[ ]]. However, this set represents only a portion of the larger subgrouping (either cold and uncontrolled, or vanished lattice).

Differences slightly greater than [[ ]] are expected for subsets of the subgroups. The staff, however, notes that the bias here still is small considering uncertainties in the MCNP solution (approximately 60 pcm), and uncertainties associated with the cross-sections when compared to critical experiments (100 to 300 pcm). Therefore, the staff judges the agreement over the test suite to be very good and essentially consistent with the LANCR02 performance for the baseline suite.

On the basis of the comparisons, the staff found that the capability of LANCR02 to analyze lattices with varied edge part length rod configurations has been adequately demonstrated.

3.5.3.4 Mixed Oxide

This test suite was developed to examine potential trends in LANCR02 for various plutonium loadings in the form of Mixed Oxide (MOX) fuel. The suite considered prototypical 8X8 and 9X9 MOX bearing fuel assemblies. One assembly was designed to be a high-power assembly while the other was intended to capture any trends at lower powers. However, as MOX fuel is not used in domestic LWR applications, the staff did not perform a review assessment of LANCR02's MOX capabilities.

3.5.3.5 Reactivity Worth of Depleted Lattice

This test suite was developed to examine potential trends in LANCR02 for increased exposure. The suite considered both ~~urania~~ uranium oxide (UOX) and MOX fuel. Exposures considered in the test suite included Beginning Of Life (BOL), 40 gigawatt days per short ton (GWd/ST) and 80 GWd/ST. This range covers the exposure range of interest for current reactor applications.

The process by which the fuel composition was generated for the MCNP calculations was unclear to NRC staff. In their response to RAI 43, GNF stated that LANCR02 was used to deplete the fuel lattices to the desired exposure step, and the nuclide densities calculated in the depletion are used to create the MCNP input deck. Thus, the eigenvalue and power distribution comparison were performed using a consistent inventory between the two codes. However, the staff noted that this approach is contingent upon the sufficiency of LANCR02's depletion capabilities. These capabilities were independently qualified through comparison to MONTEBURNS calculations as discussed in Section 3.5.3.18 of this SE. Therefore, the staff finds the approach to generating the fuel composition acceptable.

3.5.3.5.1 Uranium Oxide Results

[[

]] Therefore, the staff did not consider these differences to demonstrate any meaningful deficiency in LANCR02.

[[

]]

Comparisons of the Doppler and rod worth calculations over the exposure range indicate consistent performance for the exposed lattice relative to the baseline test suite [[

]].

Eigenvalue statistics provided in Table 5.5-13 indicate the same performance for UOX fuel under exposed conditions as for BOL conditions for the baseline test suite. [[

]]

Using the trend criterion, two trends were identified based on the UOX depletion cases. The first is between the hot uncontrolled case at BOL and 40 GWd/ST. The difference in the eigenvalues between these subgroups is [[ ]]. The second is at the 40 GWd/ST exposure. The difference between the hot controlled and hot uncontrolled subgroup average eigenvalues is [[

]]. The staff reviewed the results of the power distribution comparisons at this exposure for hot controlled and uncontrolled conditions and found that the agreement in the peak pin power and the fission density RMS was excellent. Therefore, while these trends have been identified, the staff finds that the trend itself is relatively minor and will not likely have a significant impact on downstream calculations.

On these bases, the staff concludes that GNF has demonstrated consistent performance for UOX fuel over the exposure range of interest for current reactor applications.

3.5.3.5.2 Mixed Oxide Results

Section 5.5.6 of the qualification LTR presents comparison results between LANCR02 and MCNP for MOX analyses. Just as with the UOX analyses, the results encompass hot and cold eigenvalues, fission rate RMS, relative pin peaking factor, eigenvalue difference grouped by void, and Doppler and rod worth. However, as MOX fuel is not used in domestic light-water reactor (LWR) applications, the staff did not perform a review assessment of LANCR02's MOX capabilities.

3.5.3.6 Alternative Product Lines

The purpose of this test suite is to identify any potential trends or biases in LANCR02 for calculations of alternative fuel products. This test suite considers fuel product lines with features that are not deployed in the current GNF fuel product lines. In particular, this relates to the arrangement of internal water channels. The test suite considers offset water box internal water channels and water wing water channels. The qualification LTR states that the geometry input for these fuel products is based on publicly available information about alternative fuel designs and are approximate. The staff finds that the approximated geometry is acceptable to test the overall capability of LANCR02 to analyze fuel with alternative geometric features.

Figure 5.6-1 of the qualification LTR provides a description of the alternative fuel products. These fuel designs include: (1) water box geometries, or square arrays of fuel pins with a large square internal water channel, and (2) water wing geometries, or square arrays of fuel pins with a large central diamond shaped water channel with four other narrow channels that bridge the span between the channel wall and the central diamond channel. Geometries such as these are consistent with alternative fuel product lines currently deployed in the operating fleet.

The staff reviewed the hot eigenvalue comparisons and found the hot eigenvalue differences to be consistent with the performance demonstrated for the baseline test suite. [[ ]] The staff noted that for the cold conditions, [[ ]]

[[ ]] These power distribution comparisons indicate consistent performance for the alternative fuel products as was demonstrated for the baseline test suite.

The staff noted that the rod worth tended to be [[ ]] However, the staff notes that the differences were all very near [[ ]]; thus, demonstrating essentially equivalent performance for the calculation of the rod worth for the three alternative fuel designs.

When all of the test cases are considered the average eigenvalue bias is [[ ]] with a standard deviation of [[ ]]. These results are consistent with the baseline results. [[

]]. Therefore, the staff found that the trend was sufficiently small that no significant impact would be expected for downstream analyses.

Overall, the agreement between the LANCR02 and MCNP results is excellent. The staff finds that GNF has adequately demonstrated acceptable performance for the LANCR02 method to address alternative fuel product design features commonly deployed in the current operating fleet.

3.5.3.7 Gadolinium Concentration

The purpose of this test suite is to assess the capabilities of LANCR02 to analyze lattices with various gadolinia loadings. Two lattices were considered for the GE14 fuel design with gadolinia loadings between 0 and 12 weight percent (w/o). The cases considered also include enriched gadolinia cases where gadolinium-155 and gadolinium-157 (the principle absorbers) are tested at increased concentrations, up to an equivalent worth of 12 w/o natural gadolinia loading. These cases allow for the separate qualification for both of the principle gadolinia isotopes. The arrangement of the gadolinia bearing fuel pins is the subject of test suite 8, so only one of the arrangements for the GE14 dominant and vanished zones are considered in the subject suite.

The staff reviewed the range considered and finds that this range (from 0 to 12 w/o) is acceptable as it bounds the current industry practice in terms of gadolinia loadings. The test cases for the principle isotopes are acceptable to verify that both of the major absorbers are treated properly in LANCR02. Uncontrolled, hot, BOL cases were considered as part of this test suite.

Figures 5.7-1, 5.7-2, and 5.7-3 of the qualification LTR illustrate the results for the eigenvalue, fission density distribution, and pin peaking factor, respectively. Observation of these figures demonstrates that the performance of LANCR02 for these cases is wholly consistent with the performance for the baseline test suite. Eigenvalue differences remain [[ ]], fission density RMS differences [[ ]] and pin peaking factor differences [[ ]]. There were no observed trends for the test suite.

As mentioned above, the staff noted that all the cases within this test suite were performed at the BOL. As a result, the staff could not completely review the capability of LANCR02 to simulate the depletion modeling for gadolinia bearing fuel pins. The staff therefore requested the test suite be supplemented with information comparing gadolinium-155 and gadolinium-157 pin-wise inventories between LANCR02 and MONTEBURNS at various burnups. In their response to RAI 44, GNF elected to compare pin-wise inventories between LANCR02 and MCNP6 instead of MONTEBURNS because of the straightforward way in which MCNP6 can supply the user with pin-wise isotopic data. MCNP6 combines the full functionality of previous versions of MCNP, but also incorporates the ability to perform nuclide depletion calculations.

Comparing results from LANCR02 to MCNP6 necessitated an evaluation of the consistency between the new qualification tool MCNP6 and the original tool MONTEBURNS. For a representative lattice, eigenvalue results versus exposure were compared between LANCR02, MONTEBURNS, and MCNP6. The range of exposure was [[ ]], which the staff finds is representative of the range over which gadolinia is depleted. The staff reviewed the eigenvalue results between the various codes over the exposure range and found

they were nearly identical. The only observable difference was a slight increase in MCNP6 predictions by about 0.4 percent near 25 GWd/ST. The staff considers this excellent agreement between all three codes. The staff also performed a cursory evaluation of MCNP6's capabilities in conjunction with the eigenvalue comparison results and concluded that there is sufficient confidence that the depletion results provided in the test suite remain valid. Therefore, pin-wise isotopic inventories between LANCR02 and MCNP6 can be reasonably compared.

[[

]].

The depletion results were plotted as percent of initial concentration versus exposure for both LANCR02 and MCNP6. In all cases, the maximum absolute difference between the predictions of the two codes was [[ ]]. This translates to a difference in calculated gadolinia concentration of [[ ]], which the staff considers excellent agreement. Additionally, examination of the differences between both codes for all burnup steps showed no significant biases. On these bases, the staff finds the ability of LANCR02 to perform depletion modeling of gadolinia has been adequately demonstrated.

During review of the test suite results in the qualification LTR, the staff noted that only one temperature was considered. As a result, there was no basis on which to compare the Doppler worth, and the staff requested additional cases be considered to demonstrate the Doppler worth calculations. GNF supplied additional test cases in their response to RAI 44. The standard BOL, uncontrolled cases of the test suite were re-evaluated at an increased fuel temperature of 1,500˚C, and the differences between the LANCR02 and MCNPC predicted Doppler eigenvalues were plotted. Observation of the results demonstrates that the performance of LANCR02 for these cases is wholly consistent with the performance for the baseline test suite; the Doppler eigenvalue differences between subcases [[ ]]. Thus, there were no observed trends for the test suite. When considering the tests as a whole, the average predicted Doppler eigenvalue bias was [[ ]] with a standard deviation less than [[ ]], which is consistent with the baseline results and demonstrates excellent agreement.

Therefore, the staff finds LANCR02 can acceptably calculate Doppler worth for gadolinia bearing lattices.

Lastly, the staff also noted during its review that no controlled cases were presented in order to demonstrate control blade worth calculations in the presence of gadolinia bearing lattices. The staff requested additional test cases be considered, and GNF provided the test results in their response to RAI 44. The BOL, 560˚C fuel temperature cases at various void levels were re-evaluated at controlled conditions and the differences between the LANCR02 and MCNP eigenvalue results were plotted. As with other results presented for the test suite, the performance of LANCR02 for these cases is wholly consistent with the performance for the baseline test suite. The controlled eigenvalue differences [[ ]]. The average predicted eigenvalue bias for the controlled tests as a whole was [[ ]] with a standard deviation less than [[ ]], which is consistent with the baseline results and demonstrates excellent agreement. When comparing the controlled case results to the uncontrolled, no trends were observed. In light of these results, the staff finds LANCR02 can acceptably calculate control rod blade worth in the presence of gadolinia bearing lattices.

Overall, the agreement between the LANCR02 and MCNP results is excellent. The staff finds that GNF has adequately demonstrated acceptable performance for the LANCR02 method to analyze lattices with various gadolinia loadings.

3.5.3.8 Gadolinium Pin Placement

The purpose of this test suite is to compare the performance of LANCR02 for various patterns in the placement of gadolinia bearing fuel rods. The test suite considers hypothetical cases that are expected to challenge the performance of LANCR02. In particular, the test suite considers the edge and corner gadolinia bearing rods as well as lumped gadolinia bearing rod configurations. These tests were compared against the nominal gadolinia bearing rod locations for the GE14 dominant and vanished zones.

The test suite considered hot, uncontrolled cases at the BOL for the standard void fractions (0, 40, 80, and 100 percent). Figures 5.8-1 and 5.8-2 of the qualification LTR illustrate the locations for the gadolinia bearing rods. The specific configurations tested include: [[ ]].

The comparison between MCNP and LANCR02 eigenvalues is illustrated in Figure 5.8-4 of the qualification LTR. The eigenvalue differences, for the most part, [[ ]].

Power distribution comparisons indicate very good agreement. For the majority of cases the fission density RMS [[ ]] and the pin peaking factor differences [[ ]].

When the subgroups are considered for comparison to the baseline results and the nominal gadolinia rod configuration, the pin power distribution results are in very good agreement for all of the subgroupings with pin peaking factor average differences and fission density RMS differences less than [[ ]].

For the eigenvalue comparisons, the results were generally in agreement with the baseline test suite results. [[ ]] However, the staff notes that [[ ]] is only slightly larger than the trend criterion and still remains within a two standard deviation range based on the baseline test results.

Overall, the staff finds that the capabilities of LANCR02 to predict the lattice eigenvalue and power distribution has been adequately demonstrated for a range of gadolinia bearing rod arrangements. The staff reviewed the configurations considered and found these to be hypothetical challenging cases that are unlikely to be deployed in operating reactors. However, on the basis that challenging cases were considered in the study, the results are expected to bound those cases likely selected as candidate configurations for deployment.

3.5.3.9 Enrichment

Test suite ~~8~~ 9 was developed to assess the performance of LANCR02 over a range of uranium enrichments. Test cases included uniform enrichments ranging between 2 and 10 w/o. A GE14 dominant zone was considered in the test suite at hot, uncontrolled conditions with void fractions of 0, 40, 80, and 100 percent.

While typical fuel bundle designs include natural uranium lattices at the upper and lower axial extremes of the bundles, these nodes tend to have very large relative power errors. These large errors are due, in part, to the low power in these assemblies and assumptions regarding the axial leakage in core simulator methods. Considering the low importance of these nodes in the overall core nuclear analyses, the staff finds that the range of enrichments considered in the test suite is appropriate as it addresses the applicability of the LANCR02 cross-section generation for high importance regions of the core.

[[ ]] Figure 5.9-2 of the qualification LTR illustrates the eigenvalue results. The figure shows [[

]]

The highest enrichment cases are hypothetical cases and are not expected to be representative of deployed designs (currently enrichments are limited to 5 w/o). Therefore, GNF also considered the subset of comparisons at 6 w/o and lower to bound the current range of application. For these cases, the results are consistent with the baseline test results. Over the full range of enrichments, the trend with enrichment tends to be self-compensating; the average eigenvalue bias is [[ ]] with a standard deviation of [[ ]].

For current enrichment levels (5 w/o and less), the staff has found that LANCR02 has been adequately qualified. The staff notes that the eigenvalue differences at 6 w/o are the smallest, which would be indicative of higher code accuracy for those lattices containing the highest allowable enrichment loadings, which will also tend to be the higher power (thus most limiting) regions of the core. Therefore, the staff finds that the capability for LANCR02 to analyze potentially limiting conditions in terms of nodal and pin power has been adequately demonstrated. At the lowest enrichments, some biases were observed. However, these biases are relatively small and acceptable.

The power distribution comparisons indicate consistent performance with the baseline test suite. It would appear from Figure 5.9-~~2~~ 3 from the qualification LTR that [[

]]. Therefore, the staff finds that the power distribution calculations have been adequately qualified over an extensive range of fuel enrichment.

3.5.3.10 Borated Lattices

The purpose of this test suite is to assess the capability of LANCR02 to analyze borated conditions. The test suite considers GE14 dominant and vanished zone lattices at cold conditions with coolant temperatures ranging between 20 and 286 ºC. Boron concentrations of 1000 parts per million (ppm) and 2000 ppm (natural boron equivalent) were considered and compared against the results for unborated lattices at cold conditions.

The highest concentration considered (2000 ppm) is large compared to the quantity typically required to achieve cold shutdown for the current operating fleet. The staff, therefore, finds the range of concentrations considered to be representative, if not bounding, of those conditions analyzed as part of the reload licensing process.

Borated conditions are assessed for the purpose of calculating the Standby Liquid Control System shutdown margin. Therefore, consideration of the power distribution is ancillary to those calculations performed with borated conditions. Therefore, the qualification LTR only considered the eigenvalue qualification. The staff finds that this is appropriate, given the use of the boron cross-sections in standard nuclear design analysis.

The eigenvalue results provided in Table 5.10-4 of the qualification LTR do not demonstrate any trends in the comparison. The eigenvalue statistics are consistent with the baseline test suite, as well as the cold uncontrolled subgroup of the baseline test suite. Therefore, the qualification LTR demonstrates that LANCR02 performance is consistent across a wide range of boron conditions up to 2000 ppm (natural boron equivalent). On these bases, the staff finds that the LANCR02 capability of analyze borated conditions has been adequately justified.

3.5.3.11 Alternative Control Blade Designs

The purpose of this test suite is to assess LANCR02 for alternative control blade designs. Nine different designs were considered in the test suite. The [[

]] and a design similar to the ASEA Brown Boveri control blade designs with boron carbide absorber. The designs included are highly varied in terms of geometric design and absorber material. The staff finds the test suite to be comprehensive relative to current control blade designs and include novelties that have not been deployed in the current operating fleet. The tests were performed for lattices at BOL at hot conditions with void fractions varying from 0 to 100 percent.

The results were compared on the basis of eigenvalue and power distribution. [[

]] The staff compared these average differences with the hot controlled 10X10 results from the baseline suite. [[

]]

[[

]]

The eigenvalue comparisons indicate a good degree of agreement. The eigenvalue bias is larger than for the baseline results ([[ ]]) and the standard deviation is slightly larger ([[ ]]). [[ ]] However, the staff notes that these results include [[ ]] The staff notes that the results depicted in Figures 5.11-2, 5.11-3, and 5.11-4 of the qualification LTR indicate, [[

]]

Further, the staff notes that the fission density RMS differences and the pin peaking factor differences remain consistent for all of the designs and the results are consistent with the baseline results [[

]]

The staff noted during its review that the test suite for alternative control blade design did not consider cold eigenvalue comparisons. Because the LANCR02 output will be utilized to assess reactivity under cold controlled conditions, the staff requested the subset of 0 percent void fraction cases be examined in order to make conclusions regarding the applicability of LANCR02 to provide cross-section data for cold shutdown margin calculations. GNF's response to RAI 47 stated that for all 0 percent void fraction cases, the average eigenvalue difference is [[ ]] with a standard deviation of [[ ]]. This is consistent with the 0 percent void fraction cold controlled conditions in the baseline test suite. When compared to the baseline test suite to identify trends, the staff observed an average difference in eigenvalue of [[ ]], which is well within the [[ ]]

Overall, the agreement between the LANCR02 and MCNP results is very good. The staff finds that GNF has adequately demonstrated acceptable performance of the LANCR02 method to model alternative control blade designs.

3.5.3.12 Alternative Doppler Analysis

The purpose of this test suite is to assess LANCR02 for predicting Doppler worth at various temperature conditions. Hot and cold coolant temperatures are considered (286 and 20 ºC, respectively). The fuel temperatures considered include 20, 286, 560, 1500, and 2300 ºC. These cases encompass cold shutdown, hot standby, hot, and even higher temperatures. The staff finds that the temperature ranges considered in the test suite is appropriate to demonstrate the adequacy of LANCR02 to predict the Doppler worth.

The Doppler Coefficient Parameter (CDOP) for all cases agrees [[ ]]. This is consistent with the baseline results. Generally, the agreement between MCNP and LANCR02 for the dominant zone lattice considered is much better. [[

]] These results indicate excellent agreement and are adequate to demonstrate the applicability of LANCR02 to predict the Doppler reactivity over a wide range of temperature conditions.

3.5.3.13 Off-Nominal Voiding

The off-nominal void tests refer to cases where the bypass water regions are voided. Cases were considered for 80 percent and 100 percent in-channel void fraction where the bypass void fractions range from 5 to 20 percent for the intra-assembly bypass and 20 percent to 40 percent for the internal water channel. Additionally, cold condition cases were analyzed in which the water rod was assumed to have 20 percent void. These were all performed for uncontrolled conditions for the GE14 dominant zone lattice. The staff finds that the scope of the test suite is sufficient to test the LANCR02 capabilities over a reasonable range of bypass void conditions.

Eigenvalue and power distribution comparisons indicate consistent agreement with the baseline results (eigenvalue agreement within [[ ]], fission density RMS and pin peaking factor agreement within [[ ]]). The overall agreement between LANCR02 and MCNP is excellent in terms of reactivity and power distribution. The comparisons amongst subgroups indicates that there are no meaningful trends based on the trend criterion. Therefore, the staff finds that the adequacy of LANCR02 to analyze conditions of off-nominal void conditions has been adequately demonstrated.

3.5.3.14 Channel Bowing

Tests were performed with bowed channels to assess the LANCR02 capability to analyze asymmetric cases consistent with the phenomenon of channel bow. In the test suite the channel was asymmetrically bowed towards the control blade by 2 millimeters (mm) (approximately 79 mils). Both controlled and uncontrolled hot cases were considered at the standard void fractions (0 through 100 percent) for a GE14 dominant zone lattice.

The nominal channel geometry and bow channel geometry results are compared in Figures 5.14-2, 5.14-3, and 5.14-4 on the basis of eigenvalue, fission density, and pin peaking factor. The results depicted demonstrate essentially equivalent performance of LANCR02 to analyze the nominal and bowed geometries. These comparisons provide the staff with reasonable assurance that the LANCR02 solution technique is sufficiently generalized and that it has the capability to analyze asymmetric channel geometries consistent with the phenomenon of channel bow.

3.5.3.15 Fuel Rod Variations

The purpose of the fuel rod variation test suite is to assess the capability of LANCR02 to analyze slightly different fuel rods. While the baseline test suite included 8X8, 9X9, and 10X10 lattices, and therefore, included variations in the design of the fuel rod in terms of dimensions, the subject test suite considered the capability of LANCR02 to analyze fuel lattices where the rod geometry varied within the lattice. Three scenarios were considered: [[

]]. The geometries considered are provided in Figure 5.15-1 of the qualification LTR.

The test cases included hot and cold cases, both controlled and uncontrolled. The power distribution comparisons and eigenvalue comparisons indicate consistent performance relative to the baseline test suite. No meaningful trends were observed.

The staff concludes that the capability of LANCR02 to analyze lattices with different fuel rod geometries has been adequately demonstrated. When considered with the baseline test suite and the variation in rod geometry inherent between the different lattices (8X8, 9X9, and 10X10), the staff finds that the capability of LANCR02 to analyze rods of different dimensions has been adequately demonstrated.

3.5.3.16 Neutron Balance

The purpose of the neutron balance test suite is to compare the group-wise production and absorption rates for the principle isotopes. The cases considered cover the full range of void fractions for hot, BOL conditions for the GE14 dominant zone lattice. The principle isotopes considered and tabulated are uranium-235, and -238, gadolinium-155 and -157, water, and zirconium. LANCR02 has capability to output three group cross-sections, so the results are provided for the three energy groups.

The results are provided in Tables 5.16-3 through 5.16-4 of the qualification LTR. For these principle isotopes, for all three groups, the agreement between MCNP and LANCR02 is excellent.

The staff observed that a small contribution to the overall production comes from zirconium isotopes and requested confirmation that this is a result of (n, 2n) reactions. In the response to RAI 50, GNF stated that the production terms seen for zirconium did include the (n, 2n) reaction but only in MCNP. In LANCR02, (n, 2n) reactions are combined with the scattering matrices and are therefore not considered separately as a source for the neutron balance results. Based on the results presented in Tables 5.16-3 through 5.16-4, the staff observed that this approximation results in a small difference in the total production source of less than 0.2 percent. The staff does not believe this small difference is significant enough to invalidate the comparison between the two codes.

The neutron balance assessment cases provide assurance that LANCR02 adequately solves the transport and slowing down equations to generate the correct group-wise production and absorption rates. These results are consistent with the overall agreement in the power distribution and eigenvalue calculations as demonstrated in the other test suites. These cases provide an alternative and more fundamental basis to demonstrate consistent performance between LANCR02 and MCNP. The staff finds that there is adequate reasonable assurance that the LANCR02 methodology is acceptable for generating three group constants for downstream analysis.

3.5.3.17 Gamma Transport Solution

The purpose of test suite 17 is to assess the gamma transport solution in LANCR02. The basis for the comparison is to calculate the gamma heat deposition in the various sub-components

(fuel, control blade, etc.) between LANCR02 and MCNP. Accurate calculation of the gamma heat deposition signifies accurate calculation of the local gamma flux. The test cases considered address controlled and uncontrolled lattices for the GE14 dominant zone and vanished zone. By selecting these cases, GNF has included variation in the relative mass of cladding to fuel (vanished vs. dominant) and consideration of the presence of gamma absorbing materials in the bypass (control blade). The staff finds that these cases are sufficient to provide a reasonable test of the LANCR02 gamma transport calculations.

GNF compared the LANCR02 and MCNP predicted gamma energy deposition in the fuel pins. For all of the cases the agreement was fairly consistent. This is expected since the gamma flux is only subtly sensitive to the presence of in-channel water. The results indicate agreement in the gamma energy deposition within [[ ]]. The staff agrees that this degree of agreement between the two methods is excellent.

When comparing the fission heat deposition per component (fuel, clad, channel, water rod, moderator, and control blade), the differences between LANCR02 and MCNP were generally small. The fuel (as it contains high atomic number nuclides) absorbs the predominance of the gamma heat. Differences between LANCR02 and MCNP for fuel heat deposition [[

]]. This indicates a high degree of agreement and well within expected performance for gamma transport methods based on generally large uncertainties. The agreement for the apportionment between the various components between LANCR02 and MCNP is good. The staff, therefore, finds that the adequacy of the gamma transport methods in LANCR02 have been adequately demonstrated.

3.5.3.18 Depletion

The depletion assessment is provided in Section 6 of the qualification LTR. It represents an independent calculational assessment of the depletion capabilities of LANCR02. To perform this independent assessment, MONTEBURNS calculations were performed. The staff has reviewed MONTEBURNS for this purpose and finds that it provides a robust means for providing a computational assessment of the LANCR02 depletion calculations (see SE Section 3.5.1).

For MONTEBURNS to accurately predict the depletion, a large number of neutron histories are required to adequately converge the spatial and energy flux distribution for the gadolinia bearing fuel rods. The qualification LTR states that 4 million active neutron histories are simulated for each exposure step to ensure adequate convergence of the spatial and energy distribution of the flux. The eigenvalue uncertainty associated with this number of active histories is approximately 50 pcm. The staff finds that this number of histories is reasonable for the degree of accuracy sought.

Three void histories were considered in the assessment. A GE14 dominant zone lattice was selected for the basis for comparison. This is a reasonable candidate from the baseline test suite. When considered with the results of the test suite perturbations, the GE14 dominant zone lattice depletion comparison is adequate to generally assess the overall capability of the LANCR02 methodology.

The lattices were depleted with their associated void histories to approximately 55 GWd/ST exposure. This is sufficiently long that the behavior of the reactivity trend with exposure becomes linear. Further comparison of the LANCR02 and MONTEBURNS trends beyond this exposure do not provide additional insight into the depletion capability for either code. Therefore, the staff finds the range of exposure considered to be reasonable.

In all of the depletion cases, LANCR02 is run without the leakage correction model. This is to ensure consistent flux boundary conditions between MCNP and LANCR02. The staff finds this approach reasonable and notes that the purpose of the test is to compare the depletion capabilities on a consistent basis to establish any errors or trends with depletion.

The results for all three void histories are consistent. The exposure dependent eigenvalues are generally predicted within [[ ]], which is consistent with all of the BOL assessments provided in the other test suites. For all void histories, the standard deviation between LANCR02 and MCNP predicted eigenvalues is less than [[ ]]. This indicates excellent agreement, especially considering statistical uncertainty in the MNCP result of approximately 50 pcm.

No trends in eigenvalue error were observed over any period during the depletion. In the case of the depletion analyses, any errors would tend to be cumulative through the depletion and manifest as larger systematic errors at high exposure. It is on this basis that the independent MONTEBURNS qualification provides a truly robust test of the LANCR02 depletion capability. The consistently accurate prediction of the eigenvalue through the entire exposure range for various void conditions provides assurance that the LANCR02 transport and depletion methods are highly accurate.

The staff finds that the use of MONTEBURNS provides for a rigorous test of the LANCR02 method and the results indicate a high degree of accuracy with performance essentially rivaling the robust MCNP and ORIGEN methods. Therefore, the staff finds that the LANCR02 depletion capabilities have been adequately demonstrated.

3.5.3.19 Summary of Trends and Conclusions

Table 3.5.3.19.1 provides a summary of the trends observed between subgroups in the comprehensive LANCR02 test suites. [[

]] The other observed trends based on the trend criterion were small. In most cases the magnitude of the eigenvalue trend remained within two standard deviations. The identification of the trend in these cases is likely attributable to the conservatism in the trend criterion as opposed to any true indication of degraded performance.

[[

]]

[[

]]

The staff further notes that the depletion comparisons between LANCR02 and MONTEBURNS indicate excellent agreement for various void histories to relatively high exposures. As errors in the depletion model assumptions would accumulate through exposure, the staff finds that the depletion qualification provides a rigorous and robust demonstration of the LANCR02 method.

When all of the test suites are considered, the staff finds that the capability of LANCR02 to analyze lattices has been demonstrated over a wide range. The consistent accuracy demonstrated for the LANCR02 methodology illustrates the robust nature of the solution technique. On these bases, the staff finds that the accuracy of the LANCR02 method is acceptable and that the performance of the LANCR02 method is consistent with the state of the art.

Table 3.5.3.19.1: Summary of Observed Trends

Subgroup Comparison	Eigenvalue Difference (pcm)	SE Section
[[		3.5.3.5.1
		3.5.3.5.1
		3.5.3.6
		3.5.3.8
		3.5.3.9
		3.5.3.11
	]]	3.5.3.11

3.6 Code Updates

Section 9 of the model description LTR provides for updates that may be made to LANCR02 to ensure that it remains consistent with the state-of-the-art. The staff has reviewed the proposed types of potential updates to determine if NRC review and approval of such updates are required prior to the implementation of the updated code for licensing evaluations. Two types of update classifications were determined: (1) nuclear data updates, and (2) implementation updates.

3.6.1 Nuclear Data Updates

In terms of the nuclear data updates, the proposed allowable updates include incorporating new basic nuclear data. The staff finds that these updates meet the intent of SRP 4.3 to ensure that the nuclear methods utilize state-of-the-art basic data to perform these calculations. Therefore, such proposals to update the nuclear data are consistent with the staff's review criteria and may be implemented without NRC review and approval so long as the relevant testing requirements are executed and shown that such updates do not result in unintended adverse impacts on the accuracy of the method.

Also, the proposed nuclear data updates include the potential [[ ]]. In principle, this type of model update serves to improve the accuracy of the [[ ]]. However, the staff must note that such an increase in resolution, while potentially benefitting the prediction of nuclear data for core simulation applications, must not be utilized for other purposes unless reviewed and approved by the staff. [[ ]] These materials could have a negligible impact on the calculation of reactivity or power distribution; however, the accuracy of the use of nuclear methods for this purpose would need to be assessed for peripheral purposes. An example of a peripheral purpose is the calculation of radiological inventory to determine dose consequences of potential accidents in core designs including target or absorber rods.

The staff finds potential nuclear data updates that [[ ]], within the approved scope of analyses and calculations of LANCR02, acceptable on the basis that such a change would serve to [[ ]] as described above. Therefore, such a nuclear data update may be implemented, within the approved range of applicability of LANCR02, without prior NRC review and approval.

The final proposed nuclear data update relates to [[ ]]. Therefore, when considered with the testing requirements, the staff finds that model updates [[ ]] acceptable and may be performed without prior NRC review and approval.

It should be noted that although the staff finds potential nuclear data updates acceptable as stated above, the updates may only be made so long as these updates are not made and utilized in conjunction with an expansion of the usage of LANCR02 to perform additional safety analyses beyond the scope of the initial staff review. Any increased capabilities resulting from

code updates should not be used for alternative safety analyses beyond those reviewed by the staff without prior NRC review and approval.

3.6.2 Model Updates

The second classification of proposed code updates involves the LANCR02 methodology implementation. Two of the proposed implementation updates address [[
]]. The staff finds that these types of changes are only expected to increase calculational accuracy, and as before, when considered with the testing requirements, may be performed without prior NRC review and approval.

Additional proposed model updates are presented in the area of [[
]]. Therefore, the staff finds that the proposed [[ ]] may be made without prior NRC review and approval, so long as they are executed within the framework of the specific testing requirements described in the model description LTR and that they satisfy the conditions set forth in Section 4 of this SE.

[[
]]

The model updates specify that aspects of the LANCR02 methodology may be updated to [[
]] would require significant changes to many methodologies such as the one dimensional pin-cell calculational method and Dancoff factor. Such a significant departure from the reviewed and approved LANCR02 methodology would require NRC review and approval prior to implementation.

3.6.3 Testing Requirements

The testing requirements discussed by GNF in the model description LTR to be undertaken when performing proposed updates include either: (1) [[

]], or (2) [[
]]. In either case, the impact of any code change or implementation change may be quantitatively assessed. Therefore, code changes may be explicitly tested such that the [[

]].

[[

]] The staff finds that performing similar qualification to that presented in the model qualification LTR forms a sufficient basis to justify the application so long as the biases and uncertainties show consistency with the approved method (or are improved).

4.0 LIMITATIONS AND CONDITIONS

The NRC is limiting their approval to the LANCR02 methodology as presented in the model description LTR (Reference 2) and qualification report (Reference 3). Approval for use of the LANCR02 methodology and update process that has been outlined and reviewed by NRC staff in this Safety Evaluation (SE) is contingent upon the satisfaction of the following conditions and limitations:

4.1 Range of Applicability

1. The LANCR02 methodology is only applicable to LWR fuel designs of Cartesian geometry with enrichment at or below ~~the maximum permissible enrichment of~~ [[ ]] weight percent U-235.

2. LANCR02 may only be considered applicable for analyses up to an [[.

]] average lattice exposure

3. The LANCR02 methodology is not to be applied to mixed oxide (MOX) fuel designs.

4. Convergence criteria greater than the default value specified in response to RAI 31-1 (0.00005) should not be used in LANCR02 without demonstration that the results are not unreasonably affected.

5. [[

°

]]

4.2 Methodology and Data Source Updates

The NRC is limiting their approval to the LANCR02 methodology as presented in the model description LTR (Reference 2) and qualification report (Reference 3). Any changes to the calculational methodology, numerical methods, underlying principles, bases, assumptions, range of applicability, etc. are not permitted without prior NRC approval excepting those updates to nuclear data and model implementation presented below. These changes and their associated justifications are found to be acceptable. Any increased capabilities resulting from code updates should not be used for alternative safety analyses beyond those reviewed by the staff without prior NRC review and approval.

1. Updates to basic and processed nuclear data without prior NRC review are limited to the following:
   [[

   ]]

2. Updates to implementation of approved models or methodologies without prior NRC review and approval are limited to the following:
   [[

]]

5.0 CONCLUSIONS

The staff reviewed the LANCR02 model description LTR and the LANCR02 model qualification LTR against the three acceptance criteria specified in SRP 4.3, III.7 for analytical lattice physics methods as discussed in Section 2 of this SE.

The staff review of the model description LTR confirmed that the LANCR02 code is based on appropriate and acceptable theoretical models and approximations. The MoC solution technique is consistent with the industry state of the art in lattice transport methods. Therefore, the staff has found that the LANCR02 methodology meets the first criterion.

The staff reviewed the qualification of the MCNP-05P transport code with ENDF/B-VII.0 cross-sections against various critical experiments. The MCNP05-P code is based on the LANL MCNP5 code which in turn is based on a very robust Monte Carlo transport solution. The critical experiment qualification considered power distribution and eigenvalue calculations. The results of these qualification analyses justified the applicability of the ENDF/B-VII.0 cross-sections to analyze BWRs. The cross-section data are based on the most recent ENDF release and are therefore state of the art. On these bases, the staff found that the basic nuclear data are acceptable and that the second criterion has been met.

The staff reviewed the model description LTR and the qualification LTR and concluded that adequate models were included in the code to predict the relevant physical processes important for lattice physics analyses. These models include neutron thermalization, spatial coupling, and cross-section condensation. The performance of these models has been demonstrated through code-to-code comparisons against MCNP. MCNP is considered very accurate and has been directly qualified against relevant critical experiments. The staff review of the qualification LTR provides the demonstration of the code performance to simulate the important physical processes. Figures of merit were established and comparisons were made over a very extensive and comprehensive series of test suites.

The staff reviewed the results of the code-to-code comparisons for each test suite. In some cases small trends were observed that indicated subtle changes in the performance of

LANCR02 to analyze particular lattice types. The staff reviewed these trends and found that while these trends exceeded the trend criterion, they were not significant. The staff therefore concluded that the performance of LANCR02 was consistent over a wide range of applications. Also, these tests were sufficiently robust in scope to provide reasonable assurance that all of the important physics characteristics were acceptably treated by the methodology. Therefore, the staff concludes that the qualification adequately establishes that the models are sufficiently accurate and that the third criterion has been met.

While future deployment of LANCR02 in a code system including core simulator methods will require an uncertainty analysis and further review of the core simulator models, the staff has found that there is reasonable assurance that the use of LANCR02 methods to generate lattice physics information for downstream methodologies will be acceptable in such a future code system.

6.0 REFERENCES

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2. NEDC-33376P, Revision 1, "LANCR02 Lattice Physics Model Description," Global Nuclear Fuel, June 2009 (model description LTR) (ADAMS Accession No. ML091820498 / ML091820508 (Publicly Available / Non-Publicly Available)).

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