FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT ANP-10297P-A, REVISION 0, SUPPLEMENT 1,

"THE ARCADIA® REACTOR ANALYSIS SYSTEM FOR PWRs METHODOLOGY

DESCRIPTION AND BENCHMARKING RESULTS,"

FRAMATOME, INC.

DOCKET NO. 99902041

1.0 INTRODUCTION

By letter dated June 26, 2015 (Reference 1), Framatome, Inc. (Framatome) (formerly AREVA Inc.) submitted Topical Report (TR) ANP-10297P-A, Revision 0, Supplement 1, "The ARCADIA® Reactor Analysis System for PWRs [Pressurized Water Reactors] Methodology Description and Benchmarking Results" (Reference 2) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. ANP-10297P-A, Revision 0, Supplement 1 (Supplement 1) is the first supplement to the ARCADIA® Code System (ARCADIA®) TR, ANP-10297P-A, Revision 0 (Reference 3). The main purpose of Supplement 1 is to seek approval for changes that have been incorporated into the ARCADIA® methodology to support the development of Non-Loss-of-Coolant Accident methodologies, reduce uncertainties, and extend the ARCADIA® range of applicability to include fuel designs using enriched reprocessed uranium (ERU). To that end, Supplement 1 details new models and methods and updates to existing models and methods while providing expanded benchmarks. In addition to seeking approval for these changes, Supplement 1 also addresses Limitation 3 of the of the ARCADIA® safety evaluation (SE) that restricted use of ARCADIA® to fuel types with non-Inconel grids unless additional verification of uncertainties is conducted, quantified, and accounted for in licensing calculations on a plant-specific basis.

1.1 The ARCADIA® Code System

ARCADIA® is a code system developed by Framatome for neutronic and thermal-hydraulic core design and SE. To realize the neutronic and thermal-hydraulic aspects, ARCADIA® is comprised of three main codes: APOLLO2-A – the lattice physics code, ARTEMIS™ – the three dimensional (3D) core-simulator code, and COBRA-FLX™ – the 3D sub-channel code. Several ancillary modules are also incorporated to facilitate data transfer between the main codes.

Description and benchmarking of ARCADIA® was presented in TR ANP-10297P-A, Revision 0 (the original ARCADIA® TR). The original ARCADIA® TR was approved for use by the NRC in February 2013. The NRC staff's SE allowed ARCADIA® to be referenced in submittals supporting PWR neutronic design analyses and imposed a number of limitations and conditions. In order to inform the technical discussion and evaluation that is to follow for Supplement 1, it is beneficial to briefly discuss the original ARCADIA® TR and the limitations and conditions associated with it. The limitations and conditions presented below are numbered as they were in the original SE.

Enclosure

(1) The range of applicability of the ARCADIA® methodology is restricted to the fuel data provided in the TR, unless additional analysis and benchmarking is conducted to validate ARCADIA® to a fuel type not mentioned in the TR.

(2) The benchmarks provided in the ARCADIA® TR include uncertainty verification for plants that use moveable incore, rhodium fixed incore, and Aeroball incore detectors. Framatome will evaluate at least three cycles of data relative to these criteria prior to licensing the first cycle with Framatome fuel with ARCADIA®. Additionally, application of ARCADIA® to a new uncertainty measurement system(s) would require review and approval by the NRC staff prior to implementation.

(3) ARCADIA® is limited to fuel types with non-Inconel grids unless additional verification of uncertainties is conducted to account for any peaking biases due to grid type or other plant effects.

(4) For any changes made to the stand-alone version of COBRA-FLX™ that is implemented in ARCADIA® (the COBRA-FLX™ module), Framatome will revalidate ARCADIA® output using measured data from multiple plants and cycles.

1.2 Supplement 1 to the ARCADIA® Code System

As discussed in the previous section, ARCADIA® is comprised of three separate codes: APOLLO2-A – the lattice physics code, ARTEMIS™ – the 3D core-simulator code, and COBRA-FLX™ – the 3D sub-channel code. All of the new and enhanced methods and models presented in Supplement 1 are modifications made to the APOLLO2-A and ARTEMIS™ codes only. The various modifications made to these codes are listed below.

APOLLO2-A Modifications:

- Adjustments to promethium branching ratio
- Adjustments to cross sections
- New spacer grid form function model
- New large core model support
- New spectral history model

ARTEMIS™ Modifications:

- New search algorithms
- Enhanced dehomogenization model
- Incorporation of spectral cross-section model
- Updated steam tables
- Incorporation of spacer grid model
- New control rod cusping model
- Incorporation of optimal time-step control
- Enhanced coarse-fine mesh coupling
- New excore detector model
- New decay heat model
- Modification of the MEDIAN power distribution reconstruction methodology
- Enhancements to support future coupling with S-RELAP5

In addition to presenting new and enhanced models and methods, Supplement 1 seeks to remove Limitation No. 3 in the NRC staff's SE for the original ARCADIA® TR.

2.0 REGULATORY EVALUATION

The ARCADIA® TR presented a description of ARCADIA® for application to neutronic design analyses. Supplement 1 provides updates to models and methods for application to the same neutronic design analyses. In light of this, regulations that are applicable to the steady state and anticipated operational occurrences (AOOs) neutronic analysis methods presented in Supplement 1 are found in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities:"

- 10 CFR 50.34, "Contents of Applications; Technical Information," which provides the requirements for the Final Safety Analysis Report required for each plant, and includes the requirements for licensees to perform analysis of normal operation, transients, and postulated accidents to demonstrate safety of their facilities.

- 10 CFR 50.36, "Technical Specifications," which requires the Technical Specifications (TSs) include items in the following specific categories: (1) Safety limits, limiting safety system settings, and limiting control settings; (2) Limiting conditions for operation; (3) Surveillance requirements; (4) Design features; and (5) Administrative controls.

- 10 CFR Part 50 Appendix A, "General Design Criteria," which establishes the minimum requirements for principal design criteria for the facility. In particular, General Design Criterion (GDC) 10, "Reactor Design," requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs.

- 10 CFR Part 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," which requires that transient and accident analysis methods that are important to the safety of nuclear power plants be maintained under a quality assurance program.

When performing safety analyses to determine safety limits in accordance with 10 CFR 50.34 and 10 CFR 50.36, a licensee may use a variety of methods to evaluate normal operation and transients that are postulated in the licensing basis for its nuclear power plant. The NRC staff reviews these methods to ensure that they provide a realistic or conservative result and that they adhere to the applicable regulatory requirements of 10 CFR. Framatome has submitted Supplement 1 for review by the NRC so that licensees may reference its methods in safety analyses performed to support licensing without incurring additional NRC review of the methods.

Licensees perform simulations to demonstrate that the applicable regulatory criteria have been met, and as part of its regulatory oversight, the NRC reviews these simulations. To assure the quality and uniformity of NRC reviews, the NRC created NUREG-0800, "Standard Review Plan [(SRP)] for the Review of Safety Analysis Reports for Nuclear Power Plants" (Reference 4) to guide the NRC staff in performing their reviews. Guidance for the NRC staff in their review of Supplement 1 is provided in SRP Chapter 4.3, "Nuclear Design" and SRP Chapter 15.0.2, "Review of Transient and Accident Analysis Methods."

The areas of review for nuclear design and the associated acceptance criteria are defined in Chapter 4.3 of the SRP. In particular, SRP Chapter 4.3 provides criteria for ensuring that the requirements of GDC 10 are met. Satisfying these criteria ensures the models and methods presented in Supplement 1 satisfy GDC 10 and other applicable regulatory requirements.

However, as explained above in Section 1.2, Supplement 1 is an update to ARCADIA® that includes enhancements to existing models and methods. As such, additional guidance for the evaluation of these specific aspects of Supplement 1 may be found in SRP Chapter 15.0.2, Section III.5, which includes provisions for the review of submittals that constitute small changes to existing approved models and methods. When only small changes are made, the following attributes should be considered when determining the extent to which the full review process may be applied:

(1) Novelty of the revised model compared to the currently acceptable models: Small changes to a component may only need evaluation against a small subset of the entire code assessment matrix to adequately test the phenomena that are affected rather than a full review of the entire methodology.

(2) The complexity of the event being analyzed: The level of effort involved in the review process should be commensurate with the complexity of the methodology.

(3) The degree of conservatism in the methodology: The review process may be simplified if there is a large documented degree of conservatism in the methodology.

(4) The extent of any plant design changes or operational changes that would require a reanalysis: If the changes to plant design or operations are small, then the required scope of the review process is also likely to be small. This is because most of the changes to plant equipment or operations do not cause the plant to operate outside the range of validity of the approved methodology.

In summary, the NRC staff used the review guidance in SRP Chapter 4.3 and Chapter 15.0.2 in conducting its review of Supplement 1. Where applicable, as per SRP Chapter 15.0.2 Section III.5, the level of effort applied to the review of these areas was commensurate with the magnitude of the changes made to certain models and methods in support of the present application relative to the previous individual approvals for ARCADIA® and their impact on relevant phenomena.

3.0 TECHNICAL EVALUATION

Supplement 1 is the application for the use of new and/or enhanced models and methods to an existing NRC-approved code, ARCADIA®. The submittal is therefore comprised of a variety of smaller sections where each new or improved model is detailed in turn. A high-level breakdown of these sections is:

(1) APOLLO2-A Methodology Changes
(2) ARTEMIS™ Methodology Changes
(3) APOLLO2-A Revalidation
(4) ARCADIA® Revalidation of Core Benchmarks
(5) Validation for Transient Model
(6) Additional Parameter Benchmarks

(7) Power Distribution Uncertainties
(8) Safety Parameter Uncertainties

Each of the sections within Supplement 1 will be discussed and evaluated in the following subsections.

3.1 APOLLO2-A Methodology Changes

As mentioned in Section 1.2, nuclear data generation within ARCADIA® is realized through the lattice physics code APOLLOA2-A. APOLLO2-A is a deterministic code developed by Framatome for its industrial applications. APOLLO2-A solves the two dimensional (2D) transport equation for **[ ]** using Collision Probability (CP), the Integro-Differential Transport (IDT) Solver and the Method of Characteristics (MOC) solution methods in stages. The main purpose of this code is to provide fuel-assembly-averaged microscopic cross sections, heterogeneous form functions, and kinetic parameters for the 3D neutronics core simulator ARTEMIS™.

The following subsections discuss the changes made to the APOLLO2-A methodology.

3.1.1 Promethium Branching Ratio Modification

Analyses performed using ARCADIA® since its NRC approval in February 2013 has shown **[ ]**. Additionally, Framatome observed that **[ ]**. Framatome analyzed cross section data to determine the parameters that could potentially influence this behavior and concluded the promethium-148m production and branching ratio parameters qualified.

Radiative capture in promethium-147 can produce either promethium-148 or the metastable promethium-148m. The promethium-148m branching ratio is the probability that radiative capture in promethium-147 will produce promethium-148m instead of the ground state, promethium-148. Since promethium-148m has a much higher capture cross section than the ground state, **[ ]**. Additionally, promethium-148m has a 41.3 day half-life, and **[ ]**.

APOLLO2-A utilizes cross section data input from JEFF3.1.1 library. Framatome conducted a review of the promethium-148m branching ratio in this library and concluded the ratio of 0.467 **[ ]**. As such, Framatome has chosen to modify the promethium-148m branching ratio and selected a value of **[ ]**, making the corresponding promethium-148 branching ratio **[ ]**. These branching ratios were also implemented in ARTEMIS™.

Framatome indicated the promethium-148m branching ratio of **[ ]** was selected based on its presentation within the literature, citing a paper by A. J. Koning and D. Rochman in Nuclear Data Sheets (Reference 5). The paper does not directly present or reference the new branching ratio, and instead discusses nuclear data evaluation using the TALYS nuclear reaction code system for the development of the TALYS Evaluated Nuclear Data Library, or TENDL. However, NRC staff confirmed the selected branching ratio exists within the TENDL library, specifically **[ ]**.

The TALYS nuclear reaction code system is software developed and maintained by the Nuclear Research and Consultancy Group for the analysis and prediction of nuclear reactions that involve neutrons, photons, protons, deuterons, tritons, helium-3, and alpha particles in the 1 kiloelectron volt – 200 mega electron volt (MeV) energy range. Three of the primary requirements of the TALYS code system development are robustness of the nuclear reaction models, consistency of data evaluations, and reproducibility of results. The ultimate goal of this approach is that, the resulting TENDL library will contain mutually consistent reaction information for all isotopes with a quality that continually increases as new experimental data and data from existing evaluations are incorporated.

Framatome indicates that, with the change in branching ratio, **[ ]**. Validation of this assessment is provided in Figure 2-1 of the TR, which shows how the predicted k^{∞} between the prior and modified versions of APOLLO2-A varies **[ ]**. For **[ ]**. This behavior is not unexpected; a change in branching ratio for production of a fission product poison will directly affect the predicted eigenvalue **[ ]**. Given a long enough **[ ]**, such as in Figure 2-1 of the TR, the change in predicted eigenvalue will level off as the production and removal of the fission product poison reaches a new equilibrium.

Having been published in a peer-reviewed journal, the TALYS nuclear reaction code system and the generation of the associated TENDL library have undergone peer-review by industry experts. Given this peer review; the code system requirements of robustness, consistency, and reproducibility; and the above discussion of validation results, the NRC staff finds the updated promethium-148m branching ratio acceptable.

3.1.2 Cross Section Adjustments

As mentioned in Section 3.1.1, analyses performed using ARCADIA® since its NRC approval in February 2013 has shown **[ ]**. To correct the latter behavior of **[ ]**, Framatome adjusted the promethium-148m branching ratio. In order to affect a change in the former behavior of **[ ]**, Framatome concluded adjustments to several types of cross-sections were necessary for a number of different isotopes. Specifically, Framatome made the following changes:

- **[ ]**
- **[ ]**;
- **[ ]**

- **[** **]**
- **[** **]**.

Each of these is discussed in turn.

3.1.2.1 Inclusion of **[** **]**

For the first of the changes listed above, Framatome indicates that **[** **]** improves calculations in the epi-thermal resonance region. It is not uncommon for a lattice physics code to implement a form of **[** **]**. Including an **[** **]**, which is more realistic, will result in a greater number of epi-thermal energy neutrons subject to possible resonance capture. This will have an impact on reactivity prediction and calculation of the Doppler temperature coefficient (DTC).

Framatome did not supply information in the TR regarding the nature of the **[** **]**. However, Framatome supplemented the response (Reference 5) to the NRC staff's RAI regarding the formulation of excore weighting factors (see Section 3.2.13 of this SE) with information on the **[** **]**. The nature of the model was verified during the audit (Reference 7). The information provided by Framatome in the RAI response supplement indicates **[**

]. This was confirmed during discussions with Framatome representatives during the audit. **[** **]**, which then proceeds to process the data normally.

Given the inclusion of **]** the NRC staff examined the RAI response supplement to determine whether any adjustments were made to **[** **]**. The RAI response supplement identifies that the **[** **]**. This was confirmed during discussions with Framatome representatives during the audit. **[** **]**.

Partial validation for the adequacy of the **[** **]** is provided in the form of Doppler power coefficient (DPC) benchmarks. These benchmarks are discussed in

Section 3.5.3 of this SE. Additional validation is provided via pin power predictions and is discussed in Section 3.1.2.5 of this SE.

In summary, the explicit consideration of **[**

]. Based on this and the validation results discussed in Section 3.5.3 and Section 3.1.2.5 of this SE, the NRC staff finds the **[** **]** to be acceptable.

3.1.2.2 Higher **[** **]**

Within a commercial light water reactor, the vast majority of neutron thermalization occurs as a result of neutron scattering interactions with the water moderator. **[** **]**, any adjustments to these cross-sections will directly impact the reactivity prediction.

Framatome did not supply information in the TR regarding the magnitude of the change in **[** **]**. However, Framatome supplemented the response (Reference 5) to the NRC staff's RAI regarding the formulation of excore weighting factors (see Section 3.2.13 of this SE) with information on the **[** **]**. The NRC staff also performed an audit to verify the **[** **]** (Reference 7). The information provided by Framatome in the RAI response supplement indicates the **[**

], which then proceeds to process the data normally.

Validation for these adjustments in cross-section is provided via pin power predictions and is discussed in Section 3.1.2.5 of this SE. Based on the above discussion and the validation results discussed in Section 3.1.2.5 of this SE, the NRC staff finds the **[** **]** to be acceptable.

3.1.2.3 Modification of **[** **]** Cross-Section

Framatome indicated the modifications made to the **[** **]** cross-section were done based on presentations within the literature, **[** **]**. This paper discuses re-estimation of **[** **]** in the JEFF3.1.1 library using continuous energy Monte Carlo calculations for fast critical experiments and large reactor geometries. Within the paper, comparisons between predicted and measured k_∞ and k-effective (k_{eff}) for each experiment indicate that, when the JEFF3.1.1 library is used, the Monte Carlo calculations consistently **[** **]** reactivity, and the reactivity predictions can be **[** **]** cross-section in several energy ranges. The magnitude of the **[** **]** for each of these energy ranges is determined based on a non-linear regression technique, and **[ ]**.

The adjustment brings the JEFF3.1.1 **[ ]** cross-section closer to that of the ENDF B-VII, **[ ]**, across the full energy spectrum. The paper also indicates the trends seen in these cross-section adjustments were later confirmed by the Institute for Reference Materials and Measurements as new differential measurements. Of specific note is the uncertainty associated with the adjusted cross-section; while the cited paper calculates a reduction in uncertainty at 1 sigma (σ) from the nominal JEFF3.1.1 value of **[ ]**, which is conservative. Validation for the adjusted **[ ]** cross-section is provided via pin power predictions and is discussed in Section 3.1.2.5 of this SE.

Having been published in a peer-reviewed journal, the magnitude of the JEFF3.1.1 **[ ]** cross-section adjustment and the manner of its determination has undergone peer-review by industry experts. Given this peer review, the alignment the adjusted cross-section has with the ENDF B-VII library, and the validation results discussed in Section 3.1.2.5 of this SE, the NRC staff finds the updated **[ ]** cross-section acceptable.

3.1.2.4 Modification of **[ ]** Cross-Sections

The Framatome modifications to the **[ ]** and **[ ]** cross sections are based on presentations within the literature. The first of two publications cited by Framatome discusses sensitivity and uncertainty studies that were performed in international collaboration by a group of research institutes to evaluate the impact of neutron cross-section uncertainty on integral parameters related to the core and fuel cycle, such as k_{eff}, Doppler reactivity coefficient, and void reactivity coefficient, among others (Reference 9). The international effort identified **[ ]** and **[ ]** as among the main contributors to reactivity uncertainty at **[ ]**, respectively. The second of the two publications cited by Framatome acts as a follow-up to the first by performing a re-estimation of nuclear data and the JEFF3.1.1 uncertainty through the development of covariance matrices calibrated on integral measurements (Reference 10).

In alignment with the results of the aforementioned publications, Framatome modified the magnitudes of the **[ ]** and **[ ]** cross sections by **[ ]** and **[ ]**, respectively. Both of these adjustments will result in an **[ ]**. While the publications cited by Framatome also discuss uncertainty reductions, **[ ]**, which is conservative

The identification of **[ ]** and **[ ]** as among the main contributors to reactivity uncertainty and the re-estimation of these nuclear data has undergone peer-review by industry experts. Given this peer review and the validation results discussed in Section 3.1.2.5 of this SE, the NRC staff finds the updated cross-sections acceptable.

3.1.2.5 Validation of Cross-Section Adjustments on Results

As validation of the cross-section adjustments, and to demonstrate an **[ ]**, Framatome provided maps of pin power differences between the approved and modified versions of APOLLO2-A in Figure 2-2 through Figure 2-4 of the TR. These maps of pin power differences are at burnups of **[ ]**:

1. **[ ]**,
2. **[ ]**,
3. **[ ]**.

The effects of modifying the promethium branching ratio are also included in these pin power maps.

The observed pin power differences across all the supplied maps are within **[ ]** and the maximum root-mean square (RMS) of the differences is **[ ]**. These results demonstrate that the pin power differences between the approved and modified versions of APOLLO2-A are small, which is the expected result given the minor cross-section adjustments discussed in previous sections of this SE. Additionally, the pin-to-pin distribution of power differences within each of the lattices is reasonable and as expected, which demonstrates the cross-section adjustments did not introduce any new behavior.

While the maps of pin power differences demonstrate the cross-section adjustments yield small changes between the approved and modified versions of APOLLO2-A, they do not serve to demonstrate the cross-section adjustments had the intended effect of **[ ]**. To assess whether the cross-section adjustments achieved the desired result, the NRC staff compared the updated reactivity predictions and the updated predicted fission rate distribution maps for the Babcock & Wilcox (B&W) -1970's and B&W-1980's critical experiment configurations with those of the approved APOLLO2-A. The updated reactivity predictions are presented in Table 4-2 through Table 4-3 of the TR and the updated predicted fission rate distribution maps are provided in Figure 4-1 through Figure 4-10 of the TR.

Comparison of the reactivity predictions shows the modified version of APOLLO2-A predicts a slightly **[ ]** with respect to the approved version of APOLLO2-A. Similarly, comparison of the two sets of fission rate distribution maps shows the modified version of APOLLO2-A predicts a slightly **[ ]** with respect to the approved version of APOLLO2-A for nearly all pins. These results demonstrate the expected **[ ]** due to the cross-section adjustments. The predicted fission rate distribution maps are also consistent with the maps of pin power differences; the observed changes are small and the pin-to-pin distribution within each critical experiment configuration lattice is as expected. Additionally, the fission rate distribution relative errors RMS for 7 of the 10 critical experiment configurations exhibits **[ ]**, indicating closer agreement with the measured data than the approved version of APOLLO2-A. For the remaining three critical experiment configurations, the relative errors RMS **[ ]**. In all instances, the relative errors RMS never exceed the **[ ]** against which the approved version of APOLLO2-A was assessed.

Comparisons of predicted and measured critical boron concentrations at the beginning of cycle and throughout the cycle for various plants were also provided. These results demonstrate an **[** **]**, while continuing to meet relevant acceptance criteria. These results are discussed in further detail in Section 3.4 of this SE.

Based on these validation results and the discussions in the previous subsections, the NRC staff finds the cross-section adjustments acceptable.

3.1.3 Enhancements to Gamma Calculation

The gamma transport model presented in Supplement 1 was initially presented in the request for additional information (RAI) responses (Reference 11) of the original ARCADIA® review and subsequently approved by the NRC staff in Reference 3. Therefore, no additional assessment of the model was performed as part of the review effort for Supplement 1. The discussion that follows is to ensure documentation of the model's development and approval is retained and that the gamma models presented in the Supplement 1 TR and the response to RAI-16 of the original ARCADIA® are identical.

On the average, approximately 12 percent of a fuel rod's power is a result of gamma energy deposition. Therefore, gamma energy is an important part of the pin power calculations. However, gammas have a large mean free path, resulting in a smoothing impact on the power density distribution within fuel assemblies. Consequently, it is important to both accurately determine the gamma sources within a fuel lattice and model gamma transport to capture the gamma energy deposition contribution to power densities.

When determining the gamma sources, the original version of APOLLO2-A utilizes the converged flux solution from the neutron transport calculations to calculate the neutron-induced prompt gammas (from fission, radiative capture, and inelastic scattering) and the delayed gammas (from fission product decay). The gamma production cross-sections are read from the gamma production library using a fine energy group structure; the incident neutron energy mesh is **[** **]**. To calculate the gamma transport, the original version of APOLLO2-A uses what Framatome calls "a simplified gamma transport model" (RAI-16 in Reference 11). The simplified gamma transport model takes advantage of **[** **]**. This approach was taken as a good compromise between accuracy and speed.

Supplement 1 indicates that, to improve the calculation of power densities, Framatome has replaced the "originally implemented model for determining the gamma distribution based on smearing" with that of "**[** **]**." However, the original version of APOLLO2-A does not make use of the smearing approach. Examination of the TR for the approved version of ARCADIA® indicates the smearing approach was initially intended for use, but was updated to the simplified gamma transport model discussed in RAI-16 of Reference 11. Additionally, examination of the gamma model presented in Supplement 1 suggests this model and the simplified gamma transport model of RAI-16 are one-and-the-same. On a clarification teleconference with Framatome on November 20, 2017, the NRC staff confirmed that this is indeed the case. Framatome further indicated that the description of the gamma transport model in Supplement 1 was included for purposes of completeness and enhancement of the discussion initially presented in RAI-16 of Reference 11.

3.1.4 Enhancements to Delayed Neutron Data

Delayed neutron data in the JEFF3.1.1 library are presented **[ ]**. Within the approved version of APOLLO2-A, this **[ ]** data is **[ ]**. While this approach is typical of lattice physics codes, it does not fully capture the effect of the nodal information that is actually available to ARTEMIS™.

Therefore, to improve data flow, Framatome implemented a change that **[ ]** from APOLLO2-A to ARTEMIS™. Framatome indicates that by **[ ]**. The NRC staff agrees with this assessment; by **[ ]**, information is preserved and ARTEMIS™ is then able to **[ ]**. Therefore, the NRC staff finds this acceptable.

3.1.5 Generation of Spacer Grid Form Functions

The approved version of APOLLO2-A does not consider the effects of spacer grids on axial power density. To improve upon this, Framatome implemented a new model in APOLLO2-A for the generation of spacer grid form functions for the treatment of spacer grids in the 3D core simulator ARTEMIS™. These form functions are informed by a number of spacer grid factors, which are generated for application in both steady state and transient core calculations. The spacer grid factors are also **[ ]**.

The TR did not present sufficient information on how the grid factors are generated or on the nature of the form functions themselves. Therefore, NRC staff performed an audit (Reference 12) of the material and requested additional information in RAI-1 (Reference 13) to assess the adequacy of the model. Both the material that was examined and the Framatome response indicate that, in addition to spacer grid factors, there are spacer grid *depression* factors. The form functions are dependent upon **[ ]** spacer grid depression factors, and each of these spacer grid depression factors is determined from one of the aforementioned spacer grid factors. Additionally, the Framatome response indicates that **[ ]** spacer grid factors are calculated, **[ ]**. Thus, **[ ]** form functions are ultimately derived.

The NRC staff examined the equations describing the spacer grid form functions during the audit and observed they are **[ ]**. These **[ ]** terms are applied to the grid depression factors to create a smoothly varying curve at all points in between the factors. This approach has a higher fidelity than **[ ]**, and it is also more realistic because it **[ ]**. However, the effectiveness of this approach ultimately depends on the adequacy with which the grid depression factors are calculated.

According to the Framatome response to RAI-1, the grid depression factors are a simple conversion of the grid factors to represent the percentage by which the grids depress **[ ]**. The grid factors themselves are calculated using a **[ ]**. The **[ ]** geometry is composed of [ ] that represent the spacer grid and the span between grids. Each of the [ ]. The transport calculation uses the standard APOLLO2-A methods of CP and the IDT MOC [ ]. At this point, [ ]. The disparity in [ ] due to [ ] between [ ]. The [ ] is also used to calculate the power response. All of these calculations ([ ]) are performed at a variety of [ ] levels [ ]. The spacer grid factors are taken as the [ ] at [ ] different [ ] locations. These locations are [ ]. The NRC staff notes this approach is substantially similar to that discussed in Reference 14.

The MOC is known to be robust and accurate for neutron transport and the calculation of a reconstructed neutron flux fine-group energy structure. The methodology accounts for the effects of self-shielding by [ ], and repeated execution of the methodology [ ], which affects the production of cross-sections [ ]. The accuracy of the methodology was demonstrated for the approved version of APOLLO2-A via critical experiments, fission rate distribution comparisons, and isotopic analyses of spent fuel. In all cases, predictions fell within 2σ of the total uncertainties. As discussed in Section 3.3 of this SE, the modified version of APOLLO2-A also exhibits the same degree of accuracy. The use of the 1D axial flux [ ] ensures a smooth [ ] response is obtained that is representative of a critical lattice.

Given the accuracy of the MOC, the calculation of a critical lattice [ ] ensures the spacer grid effects [ ] responses are adequately captured for the axial locations where the grid factors are determined. Based on this and the discussion above, the NRC staff finds the presented approach for the generation of grid form functions within APOLLO2-A to be acceptable. The application of the grid form functions, and the resulting consideration of their effects on pin burnups and in-core detector signals, is discussed in Section 3.2.5 of this SE.

3.1.5.1 [ ] Spacer Grid Factors

The spacer grid form function model discussed in Section 3.1.5 of this SE makes use of the MOC [ ] in order to calculate grid factors representative of that grid's specific effects on the [ ] response. Spacer grid types can vary from one fuel product to another, and Framatome has therefore chosen to generate [ ].

However, Framatome did not justify **[ ]**, nor did Framatome identify whether the validation results presented in the TR were calculated **]**. Framatome also did not specify the nature of the variance (e.g., variance in **[ ]** response) nor the magnitude of the variance **[ ]**. The NRC staff therefore issued RAI-2 requesting this information. Framatome's response to RAI-2 stated that all validation results in the TR were calculated **[ ]** and indicated the nature of the variance is with respect to the heat flux hot channel peaking factor (F_Q). Framatome's response outlined the approach used in the development **[ ]** to justify their use and to illustrate the variance in F_Q.

During development of the spacer grid form function model, the sensitivity of F_Q with respect to spacer grid type was explored via a total of **[ ]** different sets of grid factors generated using **[ ]** different fuel assemblies. The resulting combination of evaluation cases comprised spacer grids of various materials (**[ ]**), heights, and spans and also fuel assemblies of various fuel types, enrichments, and gadolinia content across a range of cycle burnup values. Variations in **[ ]** were also considered. The NRC staff observed that the ranges of burnups, enrichments, grid types, etc. are typical of commercial PWR fuel and operating conditions.

Examination of the predicted F_Q at the beginning of cycle (BOC), middle of cycle, and end of cycle (EOC) across the range of fuel assemblies for the various spacer grid coefficient sets **[ ]**. Although the magnitude in predicted F_Q varies substantially from one evaluation case to the next (which is to be expected given a change in assembly and burnup state), the NRC staff observed that **[ ]**. This demonstrates there is a **[ ]** for the presented spacer grid form function model and that **[ ]** reasonably representative of a **[ ]**. Based on these results, Framatome chose **[ ]**. These grid factors yield the highest predicted F_Q values at nominal conditions across the vast majority of the evaluation cases, which is conservative. For the few instances where higher F_Q values are predicted **[ ]**, the evaluation cases are at off-nominal conditions (**[ ]**), and the differences in F_Q are very small, **[ ]** at most. Therefore, the NRC staff finds this approach reasonable.

The Framatome response to RAI-2 clarified that **[ ]**. In the process, the predicted F_Q values **[ ]**. However, Framatome did not directly specify the magnitude of the deviation from the evaluation cases at which **[ ]**. Framatome clarified via teleconference on January 10, 2018, that the portion of the RAI-2 response discussing comparisons and deviations of F_Q values **[ ]** should be interpreted literally; if the F_Q value **[ ]** on an evaluation case-specific basis, then **[ ]**. The NRC staff finds this response acceptable because, based on an analysis of the evaluation cases, it is more restrictive than using two standard deviations as the limit.

Based on the above discussion, the NRC staff finds the **[ ]** to be acceptable for the presented spacer grid form function model and existing spacer grid designs. This acceptance does not preclude Framatome from **[ ]** in analyses.

3.1.6 Large Core Problems

Framatome has modified APOLLO2-A to allow for the modeling of large 2D geometries. Specifically, the capability to allow for configurations up to 1/4 core pin-by-pin geometry including radial reflectors has been added. The information provided in the TR is sparse, but it suggests the existing CP and IDT solvers are still being applied on an assembly level calculation while the APOLLO2-MOC solver can now be applied to either an assembly-level geometry (as normally done) or a larger geometry comprised of multiple assemblies. The NRC staff sought clarification of its understanding on this subject from Framatome via teleconference on January 10, 2018. In this teleconference, Framatome confirmed the NRC staff's understanding; both the CP and IDT solvers continue to solve for the multi-group neutron flux on the assembly level, which is then utilized by the APOLLO2-MOC solver on the larger core-level geometry.

The MOC is a robust calculational method for solving the neutron transport equation on an unstructured geometry. The unstructured geometry is composed of calculation regions delimited by a combination of segments and arcs as opposed to the rectangular grid of a Cartesian geometry, allowing for a very fine spatial mesh and the explicit treatment of a lattice's heterogeneity. The parallel path "ray-tracing" approach (calculation of the differential equation solution along a characteristic curve) implemented by the MOC to solve for the neutron flux is not constrained to a specific size of geometry. Rather, the accuracy of the MOC solver is dependent upon the resolution of the spatial mesh employed and the level of detail modeled within the unstructured geometry. From this perspective, the primary limitations are computational power and time.

Based on the continued use of the CP and IDT solvers to determine the assembly-level multi-group flux and the robust nature of the MOC on unstructured geometries, the NRC staff finds the modeling of configurations up to 1/4 core geometry to be acceptable.

3.1.7 Surface Spectral History Model

Modifications were made by Framatome to APOLLO2-A in order to support the surface spectral history model in ARTEMIS™. These modifications allow APOLLO2-A to **[ ]**. Within the present discussion, the neutron flux spectrum **[ ]**. According to Section 3.2.1 of the TR, the change in **[ ]** is tabulated alongside the change **[ ]**. The tabulated changes are then passed to ARTEMIS™.

The generation of the tabulated changes in **[ ]** are determined using the existing, approved calculational capabilities of APOLLO2-A. The modifications made to APOLLO2-A amount to the execution of those calculational capabilities **[ ]** to catalog the results in a look-up table. No modifications to existing models or the introduction of new models were made. Therefore, the NRC staff finds the modifications made to APOLLO2-A for the collection of tabulated values and their passing to ARTEMIS™ to be acceptable.

The ARTEMIS™ surface spectral history model itself is discussed in Section 3.2.2.1 of this SE.

3.2 ARTEMIS™ Methodology Changes

As mentioned in Section 1.2, the neutronic aspect of ARCADIA® is realized through the steady-state/transient 3D reactor core simulator code ARTEMIS™. ARTEMIS™ solves the 3D neutron diffusion equation using Nodal Expansion Method (NEM) or Semi-Analytical Nodal method. Both methods are well-known industry standard methodologies used in similar codes (such as SIMULATE-3, NESTLE, and AETNA02). Since the Semi-Analytical Nodal method is known to be more accurate for steep flux gradients, the semi-analytical method is the default solver mode for ARTEMIS™.

When solving the 3D neutron diffusion equation, ARTEMIS™ calculates macroscopic cross-sections for each solution node using the assembly-average microscopic cross-sections (determined at various base conditions with branch perturbations) from APOLLO2-A. After the flux solution is calculated, ARTEMIS™ performs a heterogeneous pin power reconstruction (termed dehomogenization) and provides a detailed thermal hydraulic coupling and kinetics capability.

The following subsections discuss the changes made to the ARTEMIS™ methodology.

3.2.1 Additional Search Algorithms

Framatome implemented a new search algorithm in ARTEMIS™ to allow for determination of the critical condition based on a specified axial offset through adjustment of control rod position or through the imposition of a xenon distribution. The Framatome purpose for implementing this algorithm is for the determination of the axial flux difference target bands that are used when calculating the penalty function in a power distribution control analysis.

Search algorithms are utilized in 3D nodal codes to determine core conditions based on a variety of different parameters. Typical search algorithms include boron, power, and control rod position, and these all currently exist in ARTEMIS™.

The TR did not indicate if this search algorithm would be used in a process to set TS safety limits or to simply generate the axial offset for a given set of conditions for comparison purposes. If such a search algorithm were to be used to set TS safety limits, a greater level of complexity may be needed than what is currently employed. The NRC staff therefore sought clarification from Framatome on the application of the search algorithm via teleconference on January 10, 2018. During the teleconference, Framatome clarified that the algorithm would be used to determine axial offsets for a given set of conditions and compared to TS safety limits, not to set them. The NRC staff finds this response acceptable and therefore finds the axial offset search algorithm acceptable.

3.2.2 Enhanced Dehomogenization Model

The pin reconstruction process in ARTEMIS™ is used to calculate pin power, pin burnup, and detector reaction rate using node-average and surface-average fluxes for each node. These fluxes are obtained from the ARTEMIS™ nodal flux solution and APOLLO2-A data constants obtained during the generation of the cross-section. In extending the process for application to mixed-oxide (MOX) fuel assemblies, Framatome observed that improvements to the "dehomogenization" (DHO) of the nodal data into pin powers were required. Pin powers at **[** **]**.

The NRC did not review ARCADIA® for applications to MOX fuels. However, Framatome indicated in the TR that, while the impetus for the improvements to the DHO process was application to MOX, they are also applicable to, **[** **]**, UO_2 fuel assemblies. Therefore, the NRC staff has reviewed the enhanced DHO model for application to UO_2 and ERU fuel types.

The modifications implemented in the DHO include a surface spectral history model and a multi-group form factor model. Each of these models is discussed in the following subsections.

3.2.2.1 Surface Spectral History Model

The few group nuclear data (cross-section, surface and corner discontinuity factors) used in ARTEMIS™ are derived by APOLLO2-A from single assembly calculations in an infinite lattice. Given the infinite (perfectly reflective) nature of these calculations, **[** **]**. The ARTEMIS™ surface spectral history model **[** **]**.

The treatment considers the local change in the cross-section due to the **[** **]**. More specifically, the treatment specifies a spectral history parameter that is defined for each nodal surface and corner as the **[** **]**. This captures how the **[** **]**.

Section 3.1.7 of this SE discusses how APOLLO2-A generates a surface spectral history look-up table that tabulates the change in **[** **]**. This table is passed from APOLLO2-A to ARTEMIS™ for use in the spectral history model. When used in conjunction with the definition of the spectral history parameter, the table allows for a functionalized "mapping" of change in **[** **]** to change in **[** **]**. The assumption is that, a change in **[** **]**. In this way, the spectral history parameter becomes analogous to a discontinuity factor, but it corrects for **]**. The NRC staff finds this approach reasonable.

During the DHO calculations, Framatome uses the relationship discussed above to determine the change in **[**

]. The [ ] is then utilized in the existing pin power reconstruction process to effectively yield pin-powers [ ] that more accurately reflect the [ ].

Based on the above discussion, the NRC staff finds the development of the surface spectral history model and its application to be reasonable. Validation of the model is discussed as part of the DHO model validation in Section 3.2.2.3 of this SE.

3.2.2.1.1 [ ] Spectral History Values

Within the TR, Framatome indicates that, ARTEMIS™ has [ ].

Framatome also states that, the [ ]. However, Framatome did not provide any justification that [ ], nor did Framatome indicate [ ] were used in generating the results presented within the TR.

The NRC staff performed an audit (Reference 12) of the material, and Framatome clarified that all the results within the TR were generated [ ]. This includes the validation results for the DHO model. As discussed in Section 3.2.2.1.1 of this SE, the NRC staff finds these results to be acceptable. Based on this, the NRC staff finds the use of [ ] to be acceptable.

Because [ ] would be developed using approved methods within APOLLO2-A and ARTEMIS™ that [ ], there is little reason to doubt their adequacy. It is anticipated that the [ ] will yield improved results in comparison to the results [ ]. However, Framatome did not request approval for [ ]. Therefore, the NRC staff did not review the adequacy of [ ], particularly with regard to the impact these values would have on predicted results and the determination of power distribution uncertainties.

3.2.2.2 Multi-group Power Form Functions

The ARTEMIS™ pin power reconstruction methodology is very similar to that of other industry codes. As a first step in this methodology, [ ].

In the approved version of ARCADIA®, power form functions are condensed to one energy group. While most of the fissions occur in the thermal energy region, a significant contribution is also made by fast fission in U-238. Therefore, Framatome modified the method to provide power form functions in two energy groups. Framatome indicates that, [

] The NRC staff agrees with this assessment.

While the description provided by Framatome in the TR indicates the updated two energy group power form functions are consistent with the current application scheme, Framatome did not explicitly present the equations for them. Therefore, the NRC staff requested this information in RAI-3. Framatome's response presents the updated power form functions and demonstrates they now have an energy group dependence. Additionally, the NRC staff verified they are modified versions of the power form functions discussed in Section 3.7 of the approved ARCADIA® TR (Reference 3). Specifically, the NRC staff observed the equations determining the intra-nodal homogeneous powers and local pin powers were appropriately adapted to accommodate two energy group power form functions. The updated local pin powers are then utilized in the previously approved methodology. Therefore, the NRC staff finds Framatome response acceptable.

Based on the above discussion, the NRC staff finds the modifications made to the power form functions for two energy group dependence to be reasonable. Validation of the model is discussed as part of the DHO model validation in Section 3.2.2.3 of this SE.

3.2.2.3 Validation of Enhanced Dehomogenization Model

As discussed in the preceding sections, the DHO model with ARTEMIS™ was enhanced through the introduction of a surface spectral history model and modification of the existing power form functions to account for two energy groups. Framatome's intent with introducing these enhancements is to produce more accurate pin powers, especially at [

].

Validation for the enhanced DHO model was provided in the form of colorset predicted pin power comparisons between ARTEMIS™ and APOLLO2-A for the approved and modified versions of ARTEMIS™. As a higher order method, the APOLLO2-A pin power predictions are taken as the reference, and the relative error between the two codes is presented. The colorset examined, contains [
]. At this burnup, [ ] and influencing the hardening of the neutron spectrum. [

] Observing the colorset for the modified version of ARTEMIS™, there is a noticeable improvement in the pin power predictions, [ ], exhibiting a reduction in magnitude from [ ] to [ ]. [
] the NRC staff observed that the enhancements to the DHO model result in a reduction in the relative error magnitude, never an increase.

Based on these validation results and the discussions presented in Section 3.2.2.1 and Section 3.2.2.2 in this SE, the NRC staff finds the enhanced DHO model in the ARTEMIS™ 3D nodal code to be acceptable. This acceptance includes the surface spectral history model and the modified two energy group power form functions discussed in the aforementioned sections of this SE.

3.2.3 Spectral Cross-Section Model

In addition to the surface spectral history model discussed in Section 3.2.2.1 of this SE, Framatome has incorporated a spectral cross-section model into ARTEMIS™. Unlike the surface spectral history model, which corrects **[**

], the spectral cross-section model corrects **[**

]. The corrections are considered for **[** **]**. The purpose of this model is to ensure the results from ARTEMIS™ match APOLLO2-A, considered a higher-order code, as closely as possible.

The treatment specifies a spectral adjustment factor that is defined as the **[**

]. Therefore, the **[** **]**. As in the surface spectral history model, the neutron flux spectrum **[** **]**. Spectral adjustment factors are computed for **[** **]**.

The NRC staff observed that, given the **[** **]** and the use of the **[** **]**, the spectral adjustment factor acts as a simple multiplicative correction coefficient that, when applied to a node, brings the ARTEMIS™ **[** **]**. Additionally, because the **[** **]** are computed by APOLLO2-A for each fuel type in the core, the spectral adjustment factors are applicable to all PWR fuel that can be acceptably modeled by APOLLO2-A.

Given the discussion above, the NRC staff finds the spectral cross-section model to be a reasonable method for bringing ARTEMIS™ results into closer alignment with APOLLO2-A. Framatome did not provide any direct validation of the model within Section 3.3 of the TR for the NRC staff to assess its acceptability. Therefore, the NRC staff assessed the model's acceptability through examination of the ARCADIA® revalidation results presented in Section 5.0 of the TR. These revalidation results, which are discussed in Section 3.3 of this SE, demonstrate consistency with, and a slight improvement over, the approved version of ARCADIA®. The NRC staff therefore finds the spectral cross-section model acceptable.

3.2.4 Update of Steam Tables

The approved version of ARTEMIS™ uses the American Standards of Mechanical Engineers (ASME) 1967 water properties (IFC-67 formulation). Framatome has updated the water properties in ARTEMIS™ to the latest version of the International Association for the Properties of Water and Steam (IAPWS) industrial formulation number 97 (IAPWS-IF97). IAPWS-IF97 is designed for speed in industrial applications. The NRC staff observed that this speed comes with only slightly less accuracy than its previous formulation, IAPWS-95. However, IAPWS-IF97 is quite comparable, is representative of the current state of the art, and represents an improvement over the ASME 1967 water properties. The NRC staff therefore finds the update of the water properties to be acceptable.

3.2.5 Spacer Grid Model

The spacer grid model Framatome is incorporating into ARTEMIS™ uses the grid form functions generated by the APOLLO2-A lattice code, as discussed in Section 3.1.5 of this SE. The grid form functions are applied after pin reconstruction on power density and group fluxes. Specifically, they are applied to **[ ]**. During their generation, the grid form functions are **[ ]**. As such, **[ ]**.

Framatome supplied validation of the spacer grid model in the form of comparison plots of calculated and measured axial power distributions. These comparisons are supplied in the TR appendices as part of the ARCADIA® revalidation, which is discussed further in Section 3.4 of this SE, and cover the BOC, middle of cycle, and EOC, for a wide variety of plant types (Westinghouse, CE, Siemens, and B&W), fuel designs (17x17 and 15x15), and fuel types (UO_2, and ERU). In all instances the plots show excellent agreement; all of the major trends and the vast majority of minor trends due to spacer grid effects are captured. Additionally, Table 5-24 through Table 5-32 of the TR present the core average axial power distribution RMS differences of these plots for both the prior version of ARTEMIS™ and the modified version. These tables demonstrate a noticeable improvement in the ability of the modified version of ARTEMIS™ to predict axial power distribution, with an average reduction in RMS of **[ ]**.

Adjustment of the power density and group fluxes to reflect the presence of spacer grids will subsequently have an effect on the determination of pin burnup and detector response. These effects should be considered. Within the TR, Framatome indicates these effects are properly accounted for when incorporating the spacer grid model. In the case of pin burnup, **[ ]**. Because this approach utilizes the adjusted pin power density **[ ]**, the effects of the spacer grid on **[ ]** are inherently captured. The NRC staff therefore finds this approach acceptable.

For the effects on detector response; **[ ]**. The NRC staff examined the detector model discussed in Section 3.7 of the approved ARCADIA® TR (Reference 3) and observed that **[ ]** will pass the effects of the spacer grid into the detector response calculation. Based on this, the NRC staff concludes the effects spacer grids will have on the detector response are acceptably captured.

Based on the above results and discussion, the NRC staff finds the spacer grid model incorporated into ARTEMIS™ acceptable.

3.2.6 Control Rod Cusping Model

Control rod movement within PWRs is performed in small discrete steps (e.g., 0.64 inches). Depending on the nodalization scheme used when representing the reactor core in a nodal code, it is possible for an inserted control rod to occupy only a portion of a node's volume. During execution of the nodal coarse mesh calculations, the axial heterogeneity of these partially rodded nodes can, depending on how the nodal cross sections are handled, lead to what is known as a 'cusping' effect. The cusping effect manifests as an exaggerated oscillating differential control rod worth, the amplitude of which increases with increasing rod insertion. The cusping effect appears when performing a pure volume weighting of the cross-sections of the rodded and unrodded portions of the node.

Within the approved version of ARTEMIS™, **[** **]**. To improve the accuracy **[** **]**, Framatome implemented a control rod cusping model. The implemented model is **[** **]**.

The **[** **]** cross-section is used in the nodal balance equation. Specifically, its application is in Equation 3-13 of the approved ARCADIA® TR, which describes the treatment for heterogeneous nodes. The NRC staff observed that **[** **]**. Substitution of **[** **]** for this term should allow for a more accurate treatment **[** **]**.

While the NRC staff finds the approach reasonable, Framatome did not supply validation for the model within the TR. The NRC staff therefore issued RAI-4 requesting this information. Framatome's response supplied plots of differential control rod worth and critical boron concentration versus control bank insertion for three different test cases. The three test cases make use of a 3D core with Cartesian nodes of similar dimensions with a gradually inserted control rod bank. The second test case makes use of a finer axial mesh (half as tall) than the first, while the third test case utilizes a control rod comprised of multiple axial material regions.

As anticipated, the differential control rod worth plots demonstrate a noticeable reduction in the amplitude of the oscillatory saw-tooth cusping effect; the largest swings were reduced from **[** **]**. The average reduction in saw tooth magnitude is **[** **]**.

The critical boron concentration plots also demonstrate improvement. In all three test cases there is an increased linearity in the critical boron concentration, which is indicative of a reduction in the cusping effect. This is especially noticeable for the third test case where the control rod is comprised of multiple axial material regions. Trendlines potted to these data for the third test case before and after application of the cusping model show an increase in the R^2 value from **[** **]**, demonstrating the increased linearity.

Based on these results and the discussion above, the NRC staff finds the control rod cusping model acceptable.

3.2.7 Incorporation of Additional Uniform Input Parameters

In order to utilize the ARCADIA® code system for transient thermal-hydraulic analyses (e.g., development of an evaluation model for transient analyses that makes use of the coupled codes presented within the ARCADIA® TR), it is necessary to incorporate data from an NRC approved system code (e.g., S-RELAP5). In preparation for the development of such an evaluation model, Framatome modified ARTEMIS™ to accept the following data: time dependent inlet temperature, inlet mass flux, and core exit pressure. These data will be used to perform 3D neutronics, sub-channel wise thermal-hydraulic, and pin-wise thermal-mechanical calculations for the transient (e.g., locked rotor or complete loss of flow).

The modifications made to ARTEMIS™ in order to support the passing of the data discussed above essentially amount to the "opening of ports" within the code to accept incoming information. No modifications to the existing models and equations are being performed. Additionally, Framatome is not seeking, through this supplement, approval for the coupling of ARTEMIS™ to an approved system code. As such, the NRC staff does not take issue with the modifications made to ARTEMIS™ to incorporate additional inputs, but neither does the NRC staff approve the coupling of ARCADIA® to a system code or its application to the aforementioned transient analyses. The acceptability of coupling ARCADIA® to an approved system code and its application to these transient analyses would need to be reviewed and approved in a separate submittal.

3.2.8 Incorporation of Non-Uniform Input Parameters

As a subset to the time-dependent inlet temperature discussed in Section 3.2.7 of this SE, Framatome has also modified ARTEMIS™ to accept non-uniform core inlet temperature distributions that may arise in steady state or transient conditions. These data are essential for the modeling of asymmetric events (e.g., main steam line break).

The modifications made to ARTEMIS™ in order to support the passing of this data essentially amount to the "opening of ports" within the code to accept incoming information. No modifications to the existing models and equations are being performed. Additionally, Framatome is not seeking, through this supplement, approval for the coupling of ARTEMIS™ to an approved system code. As such, the NRC staff does not take issue with the modifications made to ARTEMIS™ to incorporate additional inputs, but neither does the NRC staff approve the coupling of ARCADIA® to a system code or its application to the aforementioned transient analyses. The acceptability of coupling ARCADIA® to an approved system code and its application to these transient analyses would need to be reviewed and approved in a separate submittal.

3.2.9 Parameter Penalization

The penalization of sensitive parameters is necessary for applying appropriate biasing and the implementation of uncertainties, both of which are essential for conservative safety analyses. To further enhance parameter penalization, Framatome introduced a set of modifications to ARTEMIS™ that allow for the adjustment of reactivity parameters (e.g., moderator temperature coefficient, DTC, and rod worth), thermal-hydraulic parameters, and thermal-mechanical parameters.

However, Framatome did not supply information on the nature of these modifications or justification for the adjustment/penalization of the identified parameters. Therefore, the NRC staff issued RAI-5. Framatome's response supplied a brief description of the classes of parameters that would be adjusted/penalized, but emphasized that the intent of the discussion in the Supplement 1 TR regarding parameter penalization was to inform the NRC that the penalization option exists within ARCADIA® and is intended for use in a future methodology submittal. The actual parameters that will be adjusted and/or penalized and the criteria for penalization will be addressed in the future submittal. Therefore, the NRC staff did not review the parameter penalization options briefly discussed in Supplement 1.

3.2.10 Optimal Time Step Control During Transient Calculations

Transient calculations are sensitive to time discretization. The use of a time step control (TSC) model can ensure the correct time step sizes are used during transient calculations. As such, Framatome **[**

].

While the NRC staff is familiar with the TSC method discussed above, the specific implementation of it, the various criteria used to determine an appropriate time step size, and the acceptable ranges of time step size were not presented within the TR. Therefore, the NRC staff issued RAI-6. Framatome's response to RAI-6 indicated that, the general principle of the ARTEMIS™ implementation **[**

].

The TSC method described above estimates the **[** **]** to reduce it to within a pre-specified acceptance criteria, which increases the accuracy of the **[** **]**. Additionally, the analyses for which ARCADIA® has been approved (e.g., PWR reload design, reactivity and power distributions, and startup predictions) are most sensitive to the neutronic response. Therefore, the neutronics module calculation within the code system becomes a limiting factor. Determining **[** **]** ensures the **[** **]** phenomenological modules function cohesively. Based on this, the NRC finds the implemented TSC method discussed above to be reasonable.

However, Framatome's response did not specify values or justification (e.g., sensitivity studies) for the range of allowable time step sizes, **[**

] will be addressed as part of a future methodology submittal.

Based on the above discussion, the NRC staff finds the transient TSC model implemented in ARTEMIS™ to be acceptable, but the NRC staff does not approve it for use in SRP Chapter 15 transient analyses. The NRC staff review and approval of the transient TSC model as part of a future evaluation methodology submittal, wherein the staff may assess the ranges and justification for pertinent parameters (e.g., allowable time step sizes), is necessary before it may be used in SRP Chapter 15 transient analyses. This has been captured in Section 4.0 of this SE as Limitation and Condition No. 5 of the ARCADIA® code system.

3.2.11 Non-Uniform Boron Nodal Distribution

Insertion or removal of boron from the core is important in certain transients (e.g., increase steam flow and boron dilution). In these transients, the ability to explicitly model the boron front as it progresses through the core provides a better representation of the change in core reactivity throughout the transient. **[**

].

Because it is intended for use in transient analyses as part of a coupled system code, the NRC staff did not perform a review of this model. Should a future evaluation model be submitted describing the coupling of ARCADIA® to an approved system code for transient analyses, the NRC staff will review this model as part of that methodology submittal.

3.2.12 Enhanced Coarse-Fine Mesh Coupling

Supplement 1 indicates Framatome implemented internal coupling in ARTEMIS™ that allows the efficient coupling of the 3D nodal solution to the reconstructed 3D pin-wise data at each time step. However, the description of the coupling appears to be that of what is presented in Section 3.8 of the approved ARCADIA® TR; **[**

]. It is not clear how the internal coupling described in Supplement 1 is different from that in the approved ARCADIA® TR. Therefore, in an audit of the material on January 10, 2018, the NRC staff asked for clarification on the coupling description.

Within the audit, Framatome clarified that **[**

] performs the standard nodal calculation on a 1/4 fuel assembly geometry and outputs pin powers, just as it has always done. But the pin powers **[**

].

This difference in **[**

].

Therefore, the nature of the coupling is such that the closed-loop, iterative execution to a converged solution for a given time step remains **[**

]. As such, the coupling does not lend itself to possible code convergence or instability issues. The NRC staff therefore finds this coupling acceptable.

3.2.13 Excore Detector Model

The new excore detector model Framatome implemented in ARTEMIS™ is intended for the determination of excore signals in either static or time dependent calculations. The excore detector signals are used to support setpoint analyses and for the determination of trip times for transient analyses. The model uses weighted averages of the neutron source terms, which are determined using discrete ordinate transport theory techniques, and are input by core location. The excore detector model also makes use of a temperature decalibration model for adjustment of the excore detector signals based on core average temperature.

Formulation of the weighting factors and the temperature decalibration model were not provided within the TR. To assess the adequacy of the excore detector model, the NRC staff requested this information in RAI-7. Framatome's response indicates that the weighting factors are determined independent of ARCADIA® using the discrete ordinates code, DORT. Using DORT, the adjoint flux solution to the Boltzmann transport equation for the excore detector is found. Activity values for each incore assembly are extracted from the solution and adjusted to represent the cross-sectional area of the actual assembly instead of that which is modeled. These are normalized and then integrated over the assembly volume to yield assembly-specific weighting factors. Additionally, according to Framatome, interior assemblies have small contributions to the excore detector response. Therefore, only those assemblies with higher weighting factors (i.e., greater than 0.01) are used. The final weighting factors are then determined by renormalizing the weighting factors of only these assemblies.

The adjoint flux solution represents the flux contribution (importance) of each incore assembly to the excore detector. As such, its use in determining assembly-specific weighting factors is apt. Also, the NRC staff agrees with Framatome's assessment of the contribution of interior assemblies to the excore detector response; the core peripheral assemblies will be dominant. Therefore, the NRC staff finds this approach to determining weighting factors to be reasonable.

Framatome's response also provided a description of the excore model itself. The model supports a top and bottom detector at four radial locations (i.e. one in each quadrant of the core), incorporates the assembly-specific weighting factors discussed above, **[**

]. A coolant temperature correction function and a calibration factor are also included in the model. The coolant temperature correction function adjusts the calculated excore detector response for **[**

]. The calibration factor calibrates the model for an accurate response to either the current core power or the core average axial flux difference (AFD).

The use of top and bottom detectors in each of the core's four quadrants allows for a realistic determination of the AFD **[** **]**, which is necessary for setpoint analyses. Additionally, the model is calibrated to the conditions within the core and also considers the effects of **[**

]. Therefore, the NRC staff finds the formulation of the excore model to be satisfactory.

Validation of the excore detector model is provided in the TR through plots of excore signal in percent core power (percent NP) versus time for a series of rod drop tests. These test results are also part of the ARCADIA® transient revalidation suite discussed in Section 3.6 of this SE. The predicted results show excellent agreement with the measured data; all major and minor plot trends are captured, and differences between the predicted and measured data, which amount to less than **[** **]**, only occur after reaching the point of minimum percent NP. The difference between predicted and measured minimum dP/dt (change in power) is also quite small, approximately **[** **]**.

Lastly, Framatome's response to RAI-7 stated that the temperature decalibration model option is not being requested for review and approval. This model is intended for use with online core monitoring applications. Therefore, the NRC staff did not review this model.

Based on the discussions above and the validations results, the NRC staff finds the manner of determining assembly-specific weighting functions and the excore detector response model to be acceptable.

3.2.14 Decay Heat Model

Framatome indicates ARTEMIS™ is intended for use as the core model in many transient analysis applications, either independently (e.g., the first part of a rod ejection), or in conjunction with a system code (e.g., S-RELAP5) via a future methodologies submittal. To accommodate this, a decay heat model was incorporated into ARTEMIS™ that can be used in either steady state or transient analyses. The model incorporated is ANSI/ANS-5.1-2005. This is a contemporary version of the 1979 decay heat model from the American National Standards Institute (ANSI) and American Nuclear Society (ANS) discussed in 10 CFR 50.46 and Appendix K to 10 CFR Part 50. The 2005 ANS standard is substantially similar to the 1979 standard while being slightly more accurate. Therefore, the NRC staff finds this model acceptable for the calculation of decay heat in transient analyses (i.e., AOOs and anticipated transients without scram). However, while the NRC staff accepts this model for the calculation of decay heat, this acceptance does not constitute approval of ARTEMIS™ for application to SRP Chapter 15 transient analyses. Performing such analyses requires NRC review and approval of an associated evaluation methodology. This has been captured in Section 4.0 of this SE as Limitation and Condition No. 5 of the ARCADIA® code system.

3.2.15 Modification of the MEDIAN Power Distribution Reconstruction Methodology

3.2.15.1 Overview of the MEDIAN Methodology

The Measured Dependent Interpolation Algorithm using NEM (MEDIAN) is a power distribution reconstruction methodology. It uses measured activity rates obtained from detectors in instrumented assemblies to infer the power in non-instrumented assemblies (or more specifically, all non-instrumented locations within all assemblies). The approved version of the ARCADIA® TR identified the methodology as being used in conjunction with the aeroball measurement system (AMS) in German reactors and planned for use in the United States evolutionary power reactors (USEPR) plants.

During plant operation, the MEDIAN methodology is used to derive 3D core power distributions from a combination of measured and calculated data. This reconstructed power distribution is referred to as an "inferred power distribution", and as mentioned above, provides powers for all core locations as opposed to what is referred to as a "measured distribution," which provides powers for only detector locations. Several steps are involved in determining the inferred power distribution.

First, **[**

].

Second, what Framatome refers to as **[** **]** is determined. **[**

].

Third, once the **[**

].

The result of the above steps is **[** **]**. From this distribution, other quantities such as **[** **]** can be derived using the existing models and methods within ARTEMIS™ (e.g., the DHO model). This approach to power distribution reconstruction was approved for use with the AMS as part of the original ARCADIA® methodology.

3.2.15.2 Modification of the MEDIAN Methodology

Framatome made modifications to the MEDIAN power distribution reconstruction methodology described in the approved version of the ARCADIA® TR (and discussed above in Section 3.2.15.1) to expand its capabilities. These modifications include the ability to treat axially segmented, fixed incore detectors and an update to the modelling of **[** **]**. This is intended to make the methodology applicable to fixed, movable, and aeroball type incore detector systems, both full length and axially segmented.

The first of the modifications addresses **[**

] This is an unrealistic response. **[**

]. This is performed during the first step of the MEDIAN reconstruction methodology as discussed in Section 3.2.15.1 of this SE.

The **[** **]** comes from application of the ATERMIS™ nodal solution homogeneous fluxes **[** **]** to the detector flux form factors and cross sections. The NEM and Semi-Analytical methods used in ATERMIS™ are well-known industry standards that produce accurate results, and the calculation of detector responses was found to be satisfactory during NRC staff review of the original ARCADIA® methodology. Additionally, Framatome's use of **[** **]** conserves the direct measurement rates of each detector, allowing for a direct comparison of measured and calculated rates. Based on this, the NRC staff finds this approach for the treatment of axially segmented detectors to be reasonable.

The second of Framatome's modifications addresses **[**

]. In such instances, **[**

].

The treatment of **[** **]** occurs between steps two and three of the MEDIAN reconstruction methodology as discussed in Section 3.2.15.1 of this SE. Framatome's new treatment for axially segmented measurement systems occurs in step one. Therefore, **[**

].

The viability of Framatome's **[** **]** treatment depends upon the assumption that **[**

]. The NRC staff concludes that this assumption reasonably holds for the immediate vicinity of the detector; **[**

]. This conclusion is based on the understanding that these nodes will most directly influence the detector response. Based on this, the NRC staff finds the treatment for [ ] to be reasonable.

Based on the above discussions, the NRC staff finds the modified MEDIAN power distribution reconstruction methodology to be acceptable. Uncertainty quantification of the methodology for fixed, movable, and aeroball type incore detector systems is discussed in Section 3.7 of this SE.

3.3 Revalidation of APOLLO2-A

Accurate simulation of the reactor core and core parameters is not possible unless accurate and robust nuclear data from APOLLO2-A is passed to ARTEMIS™ for use in the flux solver. Validation suites were provided by Framatome in the original ARCADIA® TR to demonstrate the capability of APOLLO2-A to accurately model reactivity, fission rate distribution, and isotopic concentrations of a wide range of different fuel lattices. However, the model and methodology changes Framatome has made in the updated version of APOLLO2-A have the potential to impact these original validation results. Therefore, Framatome updated the original validation suites with results obtained using the modified version of APOLLO2-A. These revalidation suites are presented in Supplement 1.

The APOLLO2-A revalidation consists of eigenvalue and fission rate distribution comparisons for five sets of cold critical experiments, eigenvalue comparisons for integral experiments, and spent fuel isotopic comparisons. In many instances, the revalidation results are presented alongside the original validation results to compare the existing and modified versions of APOLLO2-A. The results from the revalidation are also used as one component of the pin power uncertainties. This is further discussed in Section 3.7 of this SE.

3.3.1 Critical Experiments

During the review of the approved version of ARCADIA®, the NRC staff identified that, although the APOLLO2-A validation results contain a large set of critical experiment data, all the datasets presented are for fresh fuel at cold conditions. Given that APOLLO2-A pin power uncertainties are calculated from comparison with these cold critical experiments and that the accuracy of the calculated lattice physics parameters can change with depletion, the NRC staff at the time sought justification that the pin power uncertainties calculated at zero burnup cold conditions are conservative for all burnups. Framatome provided this justification in the form of a previously published APOLLO2-A validation paper (Reference 15) comparing pin fission rate distribution RMS values from APOLLO2-A depletions versus MCOR (MCNP5-KORIGEN). Following examination of this paper, the NRC staff concluded that pin fission rate uncertainties calculated from cold critical experiments are not necessarily conservative for all burnups, but they are conservative from BOC until approximately 50 gigawatt-days per metric ton (GWd/t). Since assemblies at this exposure are never expected to be limiting, the NRC staff found the use of cold critical experiments in determining pin power uncertainties to be acceptable.

Because the modifications Framatome has made to APOLLO2-A have the potential to impact prior results, the NRC staff re-examined this conclusion in light of the revalidation suites. Table 4-12 of Supplement 1 provides a comparison of pin fission rate distribution RMS values between the original and modified versions of APOLLO2-A for the five sets of cold critical experiments. For the B&W cold critical experiments, the modified version of APOLLO2-A demonstrates a slight improvement over the original version (e.g., [ ]). For other critical experiments, such as

EPICURE, RMS results from the modified version of APOLLO2-A vary from case to case, with some being slightly smaller or slightly larger compared to the original version. However, in each case, the change in magnitude remains small (e.g., **[ ]**).

Based on these results, the performance of the modified version of APOLLO2-A appears to be commensurate with that of the original, and the NRC staff concludes that a comparison of the modified version of APOLLO2-A versus MCOR is likely to produce results similar to those presented within the literature. Therefore, the NRC staff prior conclusion remains unchanged, the use of cold critical experiments in determining pin power uncertainties is acceptable.

3.3.1.1 Reactivity Measurement Comparisons

The revalidation suite cold critical experiments presented in Supplement 1 are the B&W-1970's, B&W-1980's, KRITZ KWU, EPICURE, and CAMELEON experiments. The measured k-effective for each critical experiment is compared with the calculated value from the modified version of APOLLO2-A. These results reveal that the trends observed during the original ARCADIA® review remain: eigenvalue swings exist as well as biases in different directions between different sets of the same critical experiment. The difference in magnitude of the eigenvalues has increased from an approximate average of **[ ]**, although this is still an acceptably small difference in reactivity prediction considering the reported total uncertainty for the critical experiments of **[ ]** at 1σ. Despite this increase in magnitude, the previously observed eigenvalue swings from one case to another within a given critical experiment remains consistent, having shifted only slightly from **[ ]** to **[ ]**. The KRITZ KWU experiments also continue to show a bias of up to **[ ]** between the rodded and unrodded cases. Based on these results, the performance of the modified version of APOLLO2-A appears to be consistent with that of the original version.

The larger difference in predicted eigenvalues is not unexpected given the modifications made to APOLLO2-A for the purpose of **[ ]**. In fact, the NRC staff observed a consistent **[ ]** for all cases within the B&W-1970's and B&W-1980's critical experiments. Within the EPICURE experiments and rodded KRITZ KWU cases, the NRC staff observed a **[ ]**. This appears to be counter to Framatome's stated intent. However, this **[ ]** brings the results closer to the experimental. In all instances, the eigenvalue differences remain within 2σ of the reported total measurement uncertainty of **[ ]**. Based on this and the consistent trends in performance between the modified and original versions of APOLLO2-A, the NRC staff finds the accuracy of the updated results to be acceptable.

3.3.1.2 Fission Rate Distribution Comparisons

Eigenvalue comparisons are based on a single calculated value for the whole lattice. Because of cancellation of errors, systematic code errors may not be discovered even with a wide range of critical experiments. Fission rate distributions, however, provide pin-to-pin comparisons with the experiments, allowing for the discernment of possible systematic code errors. Spatial accuracy of the lattice physics calculations can be evaluated and an important parameter, local pin power uncertainty, can be calculated. It is also important to know the location of the pin with the highest power.

Framatome provided a series of fission rate distribution maps and summary statistics for each critical experiment. The summary statistics show that no relative errors exceed **[ ]**

RMS, and the fission rate distribution maps show, with one exception, that pin-to-pin comparisons do not exceed **[ ]**. The exception is a single pin in the **[ ]** that exhibits an error of **[ ]**. This pin is **[ ]** and has a noticeable disparity in predicted pin power error compared to that of its two neighbors (**[ ]**), which suggests the result may be an outlier. In only a few instances, APOLLO2-A mispredicts the peak pin location. However, in such cases, the power at the mispredicted location is close to (**[ ]**) the peak pin power in the experiment.

It is a challenge to measure uncertainties in critical experiments. The B&W-1970's critical experiments have, on average, a **[ ]** uncertainty in pin fission rates while the B&W-1980's critical experiments have a **[ ]** uncertainty, on average. For the KRITZ KWU, the estimated measurement uncertainty is **[ ]**. The EPICURE and CAMELEON experiments possess an estimated measurement uncertainty of **[ ]**. As mentioned above, some of the pin power predictions have errors of approximately **[ ]**. These pins exist in the B&W-1970's configurations, and exceed the associated 2σ measurement uncertainty. However, as seen in Reference 3, some of these pins can have uncertainty as large as **[ ]** just from multiple measurements. Additionally, there are only 3 pins in the B&W-1970's experiments that exhibit the prediction errors of approximately **[ ]**, and the vast majority fall well within the 2σ measurement uncertainty. For the KRITZ KWU, EPICURE, and CAMELEON experiments, all pin power prediction errors fall within the respective 2σ measurement uncertainties. Therefore, the NRC staff concludes the modified version of APOLLO2-A can acceptably predict the results of cold critical experiments.

3.3.2 Integral Experiments

The integral experiments presented within the revalidation suites are cold critical experiments that possess a very homogeneous fuel pin configuration. Since all cells except the ones at the periphery are surrounded by similar pin cells, these experiments simulate the infinite homogeneous medium conditions. As such, they are useful for validating single pin cell calculations.

The revalidation of APOLLO2-A consists of the 1978 VALDUC critical integral experiments. The maximum eigenvalue difference for the four different experiments with different moderator to fuel ratios is approximately **[ ]**. Overall, the modified version of APOLLO2-A **[ ]** k_{eff} of the analyzed experiments by **]**. This is different from the prior version of APOLLO2-A, which would **[ ]** k_{eff} by **[ ]**. Regardless, the predictions exhibit small differences in reactivity, and the NRC staff finds the predictions demonstrate the modified version of APOLLO2-A can accurately model pin cells.

3.3.3 Spent Fuel Analysis

Spent fuel isotopic benchmark analysis provides information regarding the accuracy of decay chains, the depletion algorithm, and the cross-section data used in a lattice physics code. Although core simulator benchmark results will reflect the accuracy of the depletion methodology up to a degree, they can be insensitive to isotopic distributions at the pin level.

In spent fuel analysis, samples from depleted fuel pins are analyzed to measure the isotopic distribution, which is then compared with the calculated values from the lattice physics code. The revalidation cases compare fuel pin isotopic content from a total of nine experiments.

Seven of these experiments come from different operating power plants and two come from experimental cores. A variety of fuel types is examined within the experiments, including MOX and ERU. As mentioned earlier in this SE, the NRC did not review ARCADIA® for applications to MOX fuels. The NRC staff's review of the spent fuel analysis therefore focused on the results for UO_2 and ERU fuel types (ERU fuel is discussed in Section 3.4.3 of this SE). A summary of the experiments presented within Supplement 1 is shown in Table 3.1, below.

Table 3.1: Spent Fuel Analysis Benchmark Cases

Experiment	Fuel Type	Number of Samples	Enrichment (wt%)	Lattice Type	Exposure (GWd/t)
Bugey	UO_2	1	3.1 U-235	17×17	25
Gravelines	UO_2	7	4.5 U-235	17×17	25, 44, 60
Malibu	UO_2	1	4.3 U-235	15×15	71
Cruas	ERU	6	3.1 U-235, 1.2 U-236	17×17	14, 22, 35
Gedeon 1	UO_2-Gadolinium Oxide (Gd_2O_3)	6	5 Gd, 3.25 U-235		3, 7, 11
Gedeon 2	UO_2-Gd_2O_3	6	8 Gd, 0.2 U-235		3, 7, 11
Saint Laurent	MOX	7	2.9 - 5.6 Pu	17×17	28, 42
Dampierre	MOX	4	6.7 Pu	17×17	52, 57
Malibu	MOX	1	8.1 Pu	15×15	68

The revalidation cases provided by Framatome contain detailed isotopic distribution comparisons between the calculated and measured concentrations. Comparison of the revalidation cases to the original validation cases demonstrates the impact the modified version of APOLLO2-A has on predicted isotopic concentrations is very small, less than **[ ]** across all isotopes and less than **[ ]** for important reactivity contributors such as U-235, Pu-239, Pu-240, Pu-241, and americium-241. For the large majority of isotopes in the experiments, the observed change in predicted isotopic concentration manifests as a reduction in error. This indicates the isotopic depletion and tracking performance of the modified version of APOLLO2-A is slightly improved over that of the original version.

Framatome also provided available measurement and modeling uncertainties. Examination of the revalidation cases demonstrates the vast majority of isotopic concentrations are within 2σ of the total uncertainties. This is especially true for the aforementioned important reactivity contributors, all of which are within 2σ of the total uncertainties. Based on the revalidation cases, the NRC staff concludes the modified version of APOLLO2-A acceptably depletes and tracks fuel isotopes in UO_2 fuel from low burnups to high burnups.

3.4 Revalidation of ARCADIA®

As with APOLLO2-A, the model and methodology changes Framatome has made in the updated version of ARTEMIS™ have the potential to impact the original ARCADIA® validation results. Therefore, Framatome updated the original ARCADIA® validation suites with results obtained using the modified versions of APOLLO2-A and ARTEMIS™ (i.e., the modified version of ARCADIA®). In addition, Framatome expanded the revalidation suites with the inclusion of additional benchmarks for purposes of extending the range of applicability of ARCADIA® to ERU fuel.

The ARCADIA® revalidation suites presented in Supplement 1 are extensive. 1D axial and 2D radial core nodal powers, rod and boron worths are compared to measured values for 6 different PWR reactor types with 7 different fuel types for a total of 58 cycles of operation. The summary of the benchmarks cases is given in Table 3.2, below.

Table 3.2: ARCADIA® Benchmark Suite Summary

NSSS Vendor	No. Fuel Assemblies	Fuel Type	Plants	Number of Fuel Cycles	U-235 Enrichment Range (wt%)	Gd_2O_3 Enrichment Range (wt%)
Westinghouse	157	17×17	A, B	14,1	1.8 – 4.95	2 – 8
Westinghouse	193	17×17	S1, S2	3, 3	4.2 – 4.8	2 – 8
Westinghouse	157	15×15	V1	5	3.9 – 4.2	8 and 10
CE	217	14×14	C	9	2.7 – 4.7	1 – 8
Siemens	177	15×15	G1	5	3.5 – 4.4	0 – 5
Siemens	193	18×18	G2	5	1.9 – 3.5	3 – 7
B&W	177	15×15	T1	7	2.85 – 4.9	2 – 8
Westinghouse	157	17×17[a]	E	6	3.7 – 4.0	0

[a]This fuel type contains ERU

ARCADIA® benchmarks results are evaluated for startup physics test measurements and core follow measurements.

Because Framatome seeks to extend the range of applicability of ARCADIA® to ERU fuel, which requires further assessment of APOLLO2-A results, the additional benchmarks from Plant E (see Table 3.2, above) will be discussed separately in Section 3.4.3 of this SE.

3.4.1 Startup Physics Test Measurements

For each of the cycles shown in Table 3.2, the calculated critical boron concentration, individual control rod bank worths, total control rod bank worth, and isothermal temperature coefficient are compared to the measured values for the BOC. The acceptance criteria for the benchmarks were adopted from the ANS standard ANSI/ANS 19.6.1-2005 (Reference 16) test criteria. These acceptance criteria are listed in Table 3.3, below.

Table 3.3: ANSI/ANS 19.6.1-2005 Test Criteria for Startup Physics Tests

Parameter	Criterion
All rods out critical boron concentration	±50 ppm of measured or ±500 pcm equivalent
Individual control rod bank worths	±15 percent or ±100 pcm, whichever is larger
Total control rod bank worth	±10 percent of measured
All rods out isothermal temperature coefficient	±2 pcm/ºF of measured

The BOC hot zero power (HZP) ARCADIA®-predicted critical boron concentrations are compared to measured critical boron concentrations. Examination of these comparisons for Plants A through T1 (Plant E is discussed in Section 3.4.3) demonstrates the results meet the 50 ppm acceptance criterion for all cycles. This includes cycles 7 and 8 of Plant A, which were found in the original ARCADIA® validation suites to be just outside the 50 ppm criterion, necessitating, at the time of the review, assessment using the 500 pcm equivalent criterion (the cycles were found acceptable). This indicates a slight improvement in the updated version of ARCADIA® compared to the prior version. Additionally, compared to the original ARCADIA®

validation suites, the critical boron concentrations from the revalidation suites are **[ ]** for all cycles, affirming that the cross-section adjustments made to APOLLO2-A (discussed in Section 3.1.2 of this SE) have had the intended effect of **[ ]** core reactivity.

Plants G1 and G2 in the revalidation suite do not have HZP individual rod bank worth, total control rod bank worth, and isothermal temperature coefficient measurements. For the remaining plants in the revalidation suite, BOC HZP ARCADIA®-predicted individual rod bank worths, total control rod bank worths, and isothermal temperature coefficients are compared to measurements. Examination of these comparisons for Plants A through T1 (Plant E is discussed in Section 3.4.3) demonstrates all the results meet the relevant acceptance criteria listed in Table 3.3 for all cycles with one exception. The predicted total rod bank worth of cycle 5 for Plant A differs from the measured by **[ ]**, exceeding the 10 percent criterion.

Framatome identified this exception in the TR and indicated that the measured bank worth data is suspect. This assessment is based on the observation that, **[**

].

An approximate differential control rod bank worth of **[ ]** is **[ ]** larger than what the NRC staff has observed in similar startup physics testing, and it could be indicative an **[ ]**. In addition to this, the NRC staff observed that, in comparison to the total rod worth relative errors of all other cycles for Plant A, cycle 5 shows a large disparity, and this disparity also existed within the original ARCADIA® validation suite. An NRC staff analysis of the total rod worth of all cycles for Plant A demonstrated cycle 5 exceeds two standard deviations of the mean predicted total rod bank worth, while all other cycles fall well within two standard deviations. When this analysis is considered alongside the approximated differential bank worth from Framatome, the results suggest the cycle 5 datum is likely an outlier. Based on this discussion, the NRC staff finds Framatome's assessment that the cycle 5 datum is anomalous to be acceptable. Therefore, the NRC staff finds the modified version of ARCADIA® can acceptably predict startup physics tests measurements.

3.4.2 Core Follow Measurements

For each of the cycles shown in Table 3.2, the calculated critical boron concentration (as a function of burnup), assembly-average power distribution, and core-average axial power distribution are compared to measured plant data. Boron concentration is compared at each measurement point whereas power comparisons are performed at least at BOC, middle of cycle, and EOC. The acceptance criteria for the benchmarks are based on the recommended test criteria provided in ANSI/ANS 19.6.1-2005 (Reference 16). These criteria are listed in Table 3.4, below.

Table 3.4: ANSI/ANS 19.6.1-2005 Test Criteria for Core Follow Measurements

Parameter	Criterion
Critical boron concentration	±50 ppm of measured or ±500 pcm equivalent (comparisons performed a minimum of one measurement per 30 effective full power days)
Assembly-average power distributions	RMS * 100 of |C-M| assembly powers < 5.0 for each full-core comparisons map
Core-average axial power distributions	RMS * 100 of |C-M| core-average axial power < 5.0 for each comparisons map

The ARCADIA®-predicted critical boron concentrations, assembly-average power distributions, and core-average axial power distributions are compared to measured plant data. With one exception (discussed below), the comparisons for Plants A through T1 (Plant E is discussed in Section 3.4.3) demonstrate all the results meet the respective acceptance criteria listed in Table 3.4 for all cycles. Critical boron concentrations of the revalidation suite are slightly **[ ]** in comparison to the original validation suite, affirming that the cross-section adjustments made to APOLLO2-A (discussed in Section 3.1.2 of this SE) have had the intended effect of **[ ]** core reactivity with burnup. Comparisons of assembly-average power distributions to that of the original validation suite demonstrate a slight improvement of approximately **[ ]**, on average, indicating the performance of the modified version of ARCADIA® is consistent with that of the approved version. Comparisons of core-average axial power distributions to that of the original validation suite demonstrate a noticeable improvement, with an average reduction in RMS of approximately **[ ]**. This improvement is a result of the spacer grid model, which allows for a more accurate axial power profile prediction, and demonstrates the model's effectiveness. The spacer grid model is discussed in Section 3.2.5 of this SE.

The one exception to the revalidation results meeting the respective acceptance criteria is the core-average axial power distribution RMS for **[ ]**. In this particular instance, the predicted core-average axial power distribution RMS is **[ ]**, which exceeds the 5.0 percent acceptance criterion. As discussed in the NRC staff's SE of the original ARCADIA® review (Reference 3) and reiterated in Supplement 1, this plant was operated in "load follow mode" for the first half of the cycle, indicating the reactor core was not continually operated at steady state. In such a case, it is not unexpected for a steady state core simulator to produce less accurate results. This data point is thus considered an outlier.

Based on the above discussion, the NRC finds the modified version of ARCADIA® can acceptably predict critical boron concentrations with burnup, assembly-average power distributions, and core-average axial power distributions.

3.4.3 ARCADIA® Applicability to Enriched Reprocessed Uranium

Framatome seeks to extend the range of applicability of ARCADIA® to ERU fuel. To that end, Framatome included an isotopic burnup analysis of ERU fuel in the APOLLO2-A revalidation suite and expanded the ARCADIA® revalidation suites with the inclusion of an ERU fuel plant. This plant is Plant E in Table 3.2.

Accurate simulation of the reactor core and core parameters is not possible unless the core simulator receives accurate and robust nuclear data from the lattice physics code. Thus, extending the range of applicability to include ERU fuel requires examination of the performance of both APOLLO2-A and ARTEMIS™ with respect to the new fuel type to ensure it is being accurately modeled.

The performance of the updated version of APOLLO2-A with regard to ERU fuel is assessed via spent fuel analysis. In spent fuel analysis, samples from depleted fuel pins are analyzed to measure the isotopic distribution, which is then compared with the calculated values from the lattice physics code. This provides information regarding the accuracy of decay chains, the depletion algorithm, and the cross-section data used in a lattice physics code.

Comparison of predicted to measured results for the ERU spent fuel analysis in the APOLLO2-A revalidation suite demonstrates the isotopic concentrations of important reactivity contributors (e.g., U-235, Pu-239, Pu-240, Pu-241, and americium-241) are well-predicted by the modified version of the code, within **[ ]** across the burnup range of **[ ]**. Beyond these reactivity contributors, the vast majority of isotopic concentrations presented in the analysis also fall within **[ ]** relative error. Further examination of the ERU isotopic burnup analysis shows the vast majority of isotopic concentrations are within 2σ of the total uncertainties. This is especially true for the aforementioned important reactivity contributors, all of which are within 2σ of the total uncertainties. Additionally, as discussed in Section 3.3.3 of this SE, comparison of the revalidation cases to the original validation cases demonstrates the isotopic depletion and tracking performance of the modified version of APOLLO2-A is slightly improved over that of the original version; for the vast majority of isotopes in the ERU spent fuel analysis, the observed change in predicted isotopic concentration manifests as a reduction in error of approximately **[ ]**. Based on the revalidation cases, the NRC staff concludes the modified version of APOLLO2-A acceptably depletes and tracks fuel isotopes in ERU fuel from low burnups to high burnups.

The performance of the updated version of ARCADIA® with regard to ERU fuel is assessed via startup physics tests and core follow measurements of Plant E from the ARCADIA® revalidation suites. Startup physics test measurements consist of calculated BOC, HZP critical boron concentrations, individual control rod bank worths, total control rod bank worths, and isothermal temperature coefficient comparisons to measured plant data. Core follow measurements consist of calculated critical boron concentration as a function of burnup, assembly-average power distributions, and core-average axial power distribution comparisons to measured plant data. The acceptance criteria for these two sets of measurements were adopted from the ANSI/ANS 19.6.1-2005 (Reference 16) test criteria and are listed in Table 3.3 and Table 3.4, respectively.

Comparisons of the ARTEMIS™-calculated results to measured data for all startup physics test measurements and all core follow measurements for all cycles of Plant E demonstrate the respective acceptance criteria from Table 3.3 and Table 3.4 are met for all cases, with one exception, the **[ ]**. The relative error of **[ ]**, exceeding the criterion listed in **[ ]**.

As justification, Framatome indicates in Supplement 1 that, **[ ]**.

The response to RAI-27 of the original ARCADIA® review discusses how a single failure for this measurement leads to a reasonably expected and acceptable failure rate of **[ ]** for ARCADIA® predictions and that, when all plant data in the suite is considered, the 95/95 one-sided tolerance limit for **[ ]** meets the acceptance criterion. In the NRC staff's SE for the original ARCADIA® review, this justification was found acceptable. Given the lower failure rate with the expanded revalidation suite, it follows that the 95/95 tolerance limit is still met. However, the original validation suite consisted of only UO_2 fuel while the performance of ARCADIA® for ERU is the present focus.

The Plant E validation suite consists of **[ ]**, making a single failure point lead to a failure rate of **[ ]**. The response to RAI-27 of the original ARCADIA® review did not identify whether the 95/95 one-sided tolerance limit was an upper tolerance limit or a lower tolerance limit, so the NRC staff reproduced the results and identified a one-sided lower tolerance limit was used. Calculating the 95/95 one-sided lower tolerance limit for the Plant E data yields a value of approximately **[ ]** relative error, which meets the acceptance criterion. However, while performing this calculation, the NRC staff considered the ANSI/ANS acceptance criterion is ±15 percent relative error, not simply -15 percent. A 95/95 one-sided lower tolerance limit indicates 95 percent confidence that 95 percent of predictions will be larger than the lower limit, which includes predictions with relative errors greater than +15 percent. These values would not meet the acceptance criterion. This therefore suggests it is more appropriate to use a two-sided tolerance interval to demonstrate compliance with the criterion rather than a one-sided tolerance limit. The calculation of a 95/95 two-sided tolerance interval for the Plant E data yields an upper limit of **[ ]**, which just exceeds the acceptance criterion. The RAI-27 response of the original ARCADIA® review did not discuss the choice of a lower tolerance limit versus a tolerance interval.

The NRC staff conducted an audit with Framatome (Reference 7) and sought clarification on the decision to use a one-sided tolerance limit instead of a tolerance interval. Framatome responded that, **[ ]**. In the present case, under-predicting **[ ]**. Therefore, a one-sided tolerance limit was used to demonstrate conservative predictions and compliance with the criterion. The NRC staff agrees that it is conservative to under-predict **[ ]**, but conservative predictions may not meet the intent of the ANSI/ANS acceptance criterion for startup physics tests. The NRC staff therefore examined the bases for the acceptance criterion in ANSI/ANS 19.6.1-2005.

The bases indicate the primary purpose of assessing control rod bank worth is to ensure the ability to shutdown and maintain a shutdown condition with sufficient margin both nominally and from a reactor trip. This is emphasized throughout the standard. Therefore, for the present analysis of the Plant E data, the NRC staff finds the use of a one-sided tolerance interval to be acceptable. However, the NRC staff concludes that the use of a 95/95 one-sided upper tolerance limit, not a lower tolerance limit, is more appropriate for demonstrating conservative prediction of **[ ]** because it would indicate 95 percent confidence that 95 percent of all predictions will have relative errors less than the upper limit, which includes relative errors less than -15 percent. These negative relative errors indicate **[ ]**, which is conservative. Calculation of the 95/95 one-sided upper tolerance limit for the Plant E data yields **[ ]**, which meets the acceptance criterion.

As additional justification that the **[** **]** does not represent a failure of ARCADIA®, Framatome **[** **]**. Based on this result for **[** **]**, the 95/95 one-sided upper tolerance limit result for the **[** **]**, and the basis for the present startup tests being the accurate or conservative prediction of shutdown margin, the NRC staff finds Framatome's response acceptable.

Based on the discussion above, the NRC staff finds ARCADIA® can acceptably predict startup physics test measurements and core follow measurements for plants with ERU fuel. Therefore, extending the range of applicability of ARCADIA® to include ERU is acceptable.

3.5 Additional Parameter Benchmarks

3.5.1 Cold Critical Benchmarks

Framatome presented comparisons of ARCADIA® to measurements made at cold conditions to support validation of the code system at these conditions. A set of nine critical boron concentration measurements were performed at Plant G2 (see Table 3.2 in this SE) at the end of cycle 8. The measurements were performed for various temperatures between approximately **[** **]** and for different control rod patterns. A cycle calculation was performed to reach the initial conditions and burnup of **[** **]** at which the measurements program was started. Each calculation considered the xenon concentration at the time of measurement.

Critical boron concentrations were calculated at the reported control rod configuration, moderator temperature, and pressure. Before these calculated results were compared to measurements, **[** **]**. This is consistent with the adjustments made to the critical boron concentration measurements in the startup physics tests of both the original ARCADIA® validation suite and the Supplement 1 revalidation suite.

The calculated cold critical boron concentrations demonstrate good agreement with the measurements; the maximum absolute deviation is **[** **]**, and the average deviation is approximately **[** **]**. Although this demonstrates a slight tendency of ARCADIA® to **[** **]**, it is small **[** **]** and shows no discernable trend with temperature or pressure.

Boron worth and isothermal temperature coefficient measurements were also made as part of the cold critical benchmarks. Calculations were performed **[** **]**. The calculated boron worth and isothermal temperature coefficient are also in good agreement with the measured results; differences are, respectively, **[** **]** .

The NRC staff observes the differences between calculated and measured cold critical boron concentration and isothermal temperature coefficient are consistent with what is seen in the HZP results of the startup physics tests. Based on the accuracy of the results and their consistency with other tests, the NRC staff finds ARCADIA® can acceptably model configurations at cold conditions.

3.5.2 Pseudo-Ejected Rod Worth Benchmarks

Validation of ARCADIA® with respect to determining individual control rod worth was provided by Framatome through comparison of calculated pseudo-ejected rod worths to measurements. Four measurements were used from three of the plants listed in Table 3.2 of this SE, Plant D1, Plant T1, and Plant S1. The pseudo-ejected rod worths were measured at Plant D1 using boron swap measurements, from Plant S1 using rod drop measurements, and from Plant T1 using both boron swap and rod drop measurements. All measurements were made during HZP physics testing during Cycle 1 startup. Calculations of the pseudo-ejected rod worths were made by modeling each core at the initial conditions of the measurement and considered critical boron concentration, which was then held constant.

Comparisons of the calculated and measured pseudo-ejected rod worths demonstrate good agreement in all the cases; all results fall within approximately **[ ]**. For purposes of gauging the adequacy of these results, Framatome compared them to the **[ ]**. Although this acceptance criterion **[ ]**, it is not unreasonable to use in the present case because **[ ]**. Therefore, based on these results, the NRC staff finds ARCADIA® can acceptably predict individual rod worths.

3.5.3 Doppler Coefficient Comparisons

The inclusion of **[ ]** (discussed in Section 3.1.2 of this SE) will have an impact on reactivity prediction and calculation of the DTC. The DTC is a key parameter in safety analyses, but it cannot be directly measured in commercial power reactors because of the inability to measure the effective fuel temperature. Instead, commercial power reactors measure and make use of the DPC. DTC is defined as the change in core reactivity per degree change in fuel temperature ($\Delta\rho/^{\circ}F$), but DPC is defined as the change in core reactivity as the result of a change in power level ($\Delta\rho/\Delta$ percent power) without moderator temperature changes. The primary difference between the two is that the DPC contains the fuel thermal response functions to power and the DTC does not. As such, the DPC is reasonably representative of the DTC.

In addition to the DPC, plants may also measure the power coefficient (PC), which is also known as the total power coefficient. The DPC and PC are similar, except that the PC also contains the effects of moderator temperature changes with power.

To demonstrate ARCADIA® acceptably predicts DTC, and as partial validation of the adequacy of the **[ ]**, Framatome provided a total of three predicted versus measured Doppler data sets. Two of the datasets are of DPC startup test results compiled from four different plants, and the third dataset is of PC startup test results that were compiled from two different plants. To assess the adequacy of the ARCADIA® Doppler predictions, Framatome grouped all three datasets together, yielding a mean bias of **[ ]** with a prediction uncertainty of **[ ]** at 1σ. This corresponds to a **[ ]** under-prediction in the magnitude of ARCADIA® Doppler coefficient prediction with an uncertainty of **[ ]** at 1σ. These results indicate that ARCADIA® has a tendency to conservatively predict DPC and PC. Figure 6-1 in the TR plots the measured versus predicted results and further demonstrates this tendency; only **[ ]** of the data points are shown to be non-conservatively predicted. However, these data fall well within one standard deviation of the uncertainty.

As mentioned above, when assessing the adequacy of the ARCADIA® Doppler predictions, Framatome chose to group all three datasets together. Because the data in these three datasets were collected from a variety of different plants under different conditions, the assumption that the data is poolable may not be valid, and Framatome did not supply any justification in the TR that combining them is statistically viable. Therefore, the NRC staff performed a one-way analysis of variance (ANOVA) test on the datasets to determine the acceptability of combining them. The resulting F-factor from the ANOVA analysis was ~0.6, indicating the means of the three datasets are likely not statistically different and there is no definitive reason the datasets must not be combined (although the ANOVA result does not necessarily mean the datasets can be combined, either). Therefore, for purposes of the present analysis, the NRC staff finds the ANOVA analysis supports Framatome's choice to combine the datasets.

Because the DPC is reasonably representative of the DTC, the results provided in the TR indicate ARCADIA® will tend to under-predict the magnitude of the DTC and consistently yield conservative results when the prediction uncertainty is accounted for. Based on this and the above discussion, the NRC staff finds the prediction of DTC to be acceptable. Additionally, based on the results presented in Section 3.1.2 of this SE, the NRC staff finds the **[ ]** implemented in APOLLO2-A (discussed in Section 3.1.2) to be acceptable.

3.6 Validation of ARCADIA® to Transient Benchmarks

Validation of the ARTEMIS™ transient model was provided in the original ARCADIA® review by comparison of ARTEMIS™-calculated results to reference or measured results for the TWIGL-2D benchmarks and the Nuclear Energy Agency Committee on Reactor Physics (NEACRP) reference rod ejection benchmarks and the rod drop tests. Some of the modifications incorporated into ARTEMIS™ have a potential impact on these results. Additionally, the inclusion of the up-scatter treatment in the cross-sections of APOLLO2-A has the potential to impact power levels in the dropped rod predictions. Therefore, Framatome provided updated validation results of the NEACRP reference rod ejection benchmarks and the rod drop test suite in Supplement 1. Framatome also expanded the transient validation suite by including two rod ejection experiments from the Special Power Excursion Reactor Test (SPERT) III E-core REA Benchmark program.

3.6.1 NEACRP Benchmarks

The NEACRP benchmarks are a set of analytical control rod ejection benchmarks. The core modeled in PWR transient is a 157 fuel assembly core. Six cases are presented for the PWR transient representing control rod ejections from three different configurations (center rod, four periphery rods, and single periphery rod) at two different power conditions (zero power and full power). The zero power cases eject a fully inserted control rod and the full power cases eject a partially inserted control rod. The NEACRP cases provide a range of reactivity insertions and core conditions that test the coupled neutronics, fuel thermal-mechanical, and thermal-hydraulic transient responses.

Comparison of the updated NEARCRP steady-state results to the reference case demonstrates excellent agreement. For all cases, critical boron concentrations are within **[ ]**, full power fuel temperatures are within [ ], initial local (nodal) peaks are within [ ], and reactivity releases (worth of the ejected rod with feedbacks on, at the initial power level) are within [ ]. The maximum difference between the predicted and reference power distributions is [ ]. Comparison of the updated NEACRP results to that of the original ARCADIA® NEACRP results demonstrates the performance of the modified version of ARTEMIS™ is consistent with that of the prior version; the steady-state results for the prior version of ARTEMIS™ are, respectively, [ ]. Similarly, comparison of the updated NEARCRP transient results to the reference case demonstrates very good agreement. The time to peak power for the zero power cases is within [ ], and the time to peak power for the full power cases is within [ ]. For all cases, average fuel temperatures are within [ ] at the time of peak power and within [ ] throughout the transient. Core exit temperatures are within [ ] throughout the transients (i.e., 0 to 5 seconds), and the maximum difference between predicted and reference power distributions is [ ]. For comparison, these parameters for the prior version of ARTEMIS™ are, respectively, [ ]. The comparison of transient results from the modified and prior versions of ARTEMIS™ again demonstrates that the performance of the modified version of ARTEMIS™ is consistent with that of the prior version.

Based on the good agreement between the predicted and reference results and the consistency in performance, the NRC staff concludes the modified version of ARTEMIS™, with the time-dependent neutronics solutions coupled to the time-dependent thermal-hydraulic and thermal-mechanical solutions, appropriately models the time-dependent phenomena.

3.6.2 Rod Drop Tests

The rod drop tests provided in the Supplement 1 revalidation suite of transient benchmarks heavily exercise the neutron kinetic equations of ARTEMIS™ and provide an assessment of the code's ability to accurately model neutron kinetics via evolution of the change in neutron flux throughout the transients. Five rod drop tests were performed for three plants with 193 fuel assemblies. In each test, two interior, half-core symmetric control rods were dropped, but the time in-cycle and the core reactor power was changed from test to test. In Test 1, the control rods were dropped near BOC with the reactor at 50 percent power; in Tests 2-4, the control rods were dropped near middle of cycle at 50.7 percent power; and in Test 5, the control rods were dropped near EOC at 50.1 percent power. To generate the predicted results, the cross-sections were calculated with APOLLO2-A, and the core was depleted using the ARTEMIS™ code to the starting conditions for each test.

Results are provided for the change in flux (normalized to core power) with respect to time (dP/dt) in terms of percent NP/s for each test. The predicted results show excellent agreement with the measured data; plots of dP/dt versus time show all major and minor plot trends are captured, and differences between the predicted and measured data, which amount to less than [ ], only occur after reaching the point of minimum core power. The time to reach the minimum change in dP/dt (maximum absolute) is also well-predicted; the average difference between predicted and measured is [ ] and the largest difference is [ ]. The difference between predicted and measured minimum dP/dt is also quite small, [ ]. When considering [ ], the maximum deviation expected on the minimum dP/dt will be [ ].

Comparison of the revalidation results to the original demonstrates performance of the modified version ARTEMIS™ is consistent with that of the prior version. The time to minimum change in dP/dt for the prior version of ARTEMIS™ is **[ ]** and the largest difference is **[ ]**. The difference between predicted and measured minimum dP/dt is **[ ]**.

Based on the good agreement between the predicted and measurements results and the consistency in performance, the NRC staff concludes the modified version of ARTEMIS™ appropriately models the transient change in neutron flux.

3.6.3 SPERT III Benchmarks

Framatome included additional transient validation benchmarks in Supplement 1 from the SPERT III E-Core facility rod ejection experiments. Two tests were selected for evaluation, a HZP initial condition test with a control rod ejection from a core power of 50 watts and a hot full power initial condition test with a prompt critical control rod ejection from a core power of 19 MW. Both tests are simulated using ARTEMIS™, but with cross-section a delayed neutron precursor data provided by the NRC.

These tests provide a valuable assessment of the ability of ARTEMIS™ to accurately capture the Doppler reactivity feedback. The SPERT III reactor does not include moderator voiding and the power pulses are short enough to ensure that no significant heat transfer to the moderator occurs prior to the mitigation of the prompt power excursion due to Doppler reactivity feedback. As such, this assessment provides confidence that ARTEMIS™ will accurately model the Doppler reactivity feedback in the absence of other reactivity feedback mechanisms.

The test assessments show that ARTEMIS™ well-predicts the prompt power pulse from the two SPERT III experiments. For first test, the calculated peak power of **[ ]** and calculated integral power of **[ ]** compare well with the respective measured values of 410 ± 41 MW and 8.55 ± 1.1 MW-s. For the second test, the respective calculated peak and peak integral powers of **[ ]** also compare very well to the measured values of 610 ± 60 MW and 17 ± 2 MW-s. Additionally, the NRC staff observed ARTEMIS™ well predicts the core power level before and after the power pulse in each test, which is an indication that the worth of the ejected rod was adequately determined.

The enthalpy rise for the two tests were calculated as **[ ]** (the experimental values are not known). Framatome indicates these results are within the range of applicability to PWR reactors and therefore, in conjunction with the power results discussed above, qualify ARTEMIS™ for use in rod ejection analyses. The NRC staff agrees with this assessment insofar as the results demonstrate ARTEMIS™ adequately models the neutronics responses consistent with rod ejection events (e.g., Doppler-only feedback and power response). However, this does not constitute approval of ARTEMIS™ to perform rod ejection analyses in a stand-alone state for purposes of demonstrating compliance with the applicable acceptance criteria for such an event as discussed in SRP Chapter 4.2 Appendix B and SRP Chapter 15.4.8 (i.e., fuel cladding failure criteria, core coolability criteria, fission product inventory, and peak RCS pressure limits). Performing such analyses would require NRC review and approval of an associated evaluation methodology. This has been captured in Section 4.0 of this SE as Limitation and Condition No. 5 of the ARCADIA® code system.

3.7 Power Distribution Uncertainties

During reactor operation, local peaking factors are checked at certain intervals to operate under licensed limits. The two peaking factors checked during operation, enthalpy rise hot channel factor ($F_{\Delta h}$) and heat flux hot channel factor (F_Q), refer to relative maximum fuel rod linear power and relative maximum fuel rod heat flux. For practical purposes, $F_{\Delta h}$ and F_Q can be interpreted as the pin peaking factor and the local (pellet) peaking factor, respectively. These peaking factors are calculated by two methods during reactor operation: using detector responses and using simulator predictions.

When using detector responses, measured detector response powers are converted to nodal powers using the core simulator. Measured detector responses are available only at locations where a detector is present. For locations without a detector present, the detector responses are inferred (see Section 3.2.15.1 of this SE) using the core simulator and detector response reconstruction algorithms implemented in the core monitor. The nodal powers are converted to pin powers using the core simulator.

When using simulator predictions, pin powers are directly calculated from the predicted nodal powers using the core simulator.

For licensing applications, uncertainties need to be applied to assembly peaking factors, and each of the peaking factor-calculation methods discussed above has a different uncertainty associated with it. Within Supplement 1, the uncertainty associated with the detector response method is referred to as “the inferred uncertainty”, and the uncertainty associated with the simulator prediction method is referred to as “the calculated uncertainty.” Framatome also refers to the calculated uncertainty as “Nuclear Reliability Factors.”

Both the inferred uncertainty and the calculated uncertainty are comprised of two terms that Framatome refers to as “error terms”, a local error term and a global error term. The local and global error terms respectively attempt to capture the error associated with power predictions at the pin and assembly levels. These error terms technically capture uncertainties, but to help distinguish between the various sources of uncertainty, the NRC staff has chosen to adopt Framatome’s terminology for the present discussion.

The local error term is determined the same way for both the inferred and calculated uncertainties, and is broken down into two parts: accuracy of APOLLO2-A pin power predictions (determined via comparisons of predicted and measured pin fission rates from critical experiments) and accuracy of ARTEMIS™ pin power predictions (determined via comparisons of ARTEMIS™ to APOLLO2-A).

The global error term is determined differently between the inferred and calculated uncertainties. Because the detector response method calculates assembly powers in non-instrumented locations based on detector measurements, the inferred uncertainty will be different for each detector system. The global error term for the inferred uncertainty must therefore be determined for each detector system. In contrast, the global error term for the calculated uncertainty is determined based on comparisons of ARTEMIS™ to measured values.

The local and global error terms for the inferred uncertainty are discussed in the following subsections.

The estimated inferred and calculated uncertainties are 95/95 tolerance limits, and they are calculated in two ways. The first, referred to as “Normal Uncertainty”, assumes the populations for the error components are normally distributed and are combined using a square-root-sum-of-squares (SRSS) approach. The second, referred to as “Non-Parametric Uncertainty”, performs Monte-Carlo simulations on the actual distributions to obtain a 95/95 tolerance limit. The non-parametric approach does not require the error component populations be normally distributed. The more limiting of the two tolerance limits is used for the uncertainties.

The NRC staff has investigated the Non-Parametric Uncertainty analysis technique and found the results of this process are sensitive to the range over which the input uncertainty distributions are sampled. For example, sampling the input uncertainty distributions from a truncated range of ±2σ does not reasonably predict the 95/95 tolerance limit that is produced when sampling from the full range of the input uncertainty distributions. Therefore, the NRC staff requested information regarding the input uncertainty distribution sampling ranges Framatome uses when performing Non-Parametric Uncertainty analyses. In the response to RAI-10, Framatome stated **[ ]**. Because this range **[ ]**, the NRC staff finds the response acceptable.

3.7.1 Local Error Term

As discussed in Section 3.7 of this SE, the local error term is broken down into two parts: accuracy of APOLLO2-A pin power predictions and accuracy of ARTEMIS™ pin power predictions

APOLLO2-A pin power accuracy is determined via comparisons of predicted and measured pin fission rates from the B&W-1980s critical experiments (see Section 3.3.1.2 of this SE). The acceptability of these experiments in determining the local error term was discussed in the NRC staff’s SE for the original ARCADIA® review (Reference 3). Comparison of results from the modified version of APOLLO2-A to the prior version demonstrate a slight improvement in results, approximately **[ ]**, for all core configurations. This leads to total peaking uncertainty of **[ ]** versus the prior result of **[ ]**. The calculated uncertainty is representative of pin power uncertainty for lattices with 1.94 wt% to 4.02 wt% U-235 enrichment and up to 4.0 percent Gd enrichments.

ARTEMIS™ pin power accuracy is determined via comparisons of colorset (multi-assembly) calculations from ARTEMIS™ and APOLLO2-A. In these comparisons, pin powers in multi-bundle (2×2) configurations with different enrichments and poison contents are calculated using the lattice physics code APOLLO2-A and considered as the reference solution. The reference solution is then compared with pin powers calculated by the core simulator ARTEMIS™ using the homogenized lattice data and form factors provided by the lattice physics code. Because ARTEMIS™ pin powers are calculated using the DHO process, the colorset comparisons are representative of the error associated with the DHO process. The acceptability of these comparisons in determining the local error term was discussed in the NRC staff’s SE for the original ARCADIA® review (Reference 3). Comparison of results from the modified version of ARTEMIS™ to the prior version demonstrate a slight improvement; the maximum relative error in peak pin powers is **[ ]** and while the relative pin power RMS error remains within **[ ]**, the standard deviation **[ ]**. Framatome takes a conservative approach in calculation of the DHO errors by considering the statistics from only the peripheral pins, which tend to have higher differences. The mean and standard deviation from these peripheral pins is **[ ]**, respectively. This is a slight difference from the prior mean and standard deviation results of **[ ]**, respectively.

The dataset used to determine DHO accuracy for the modified versions of ARTEMIS™ is expanded over the original by inclusion of four ERU fuel colorsets. Because Framatome seeks to expand the range of applicability of ARCADIA® to include ERU fuel (see Section 3.4.3 of this SE), the NRC staff statistically analyzed the periphery pins for ERU colorset No. 24. The calculated mean and standard deviation of **[ ]**, respectively, are consistent with those of other colorsets and the overall colorset results.

The 26 colorset calculations cover a wide range of fuel enrichments (2.25 wt% to 4.95 wt% U-235) and burnable poison enrichments (2 percent to 10 percent Gd). At least 23 of the colorsets include gadolinia pins and four of the colorsets include ERU. Each colorset was depleted around 20 gigawatt-days per metric ton of uranium (GWd/mtU) to cover a range of depletions of 0 GWd/mtU to 60 Wd/mtU. The NRC staff finds the ranges acceptable and, based on them, the calculated DHO uncertainty of **[ ]** can be considered representative of typical PWR simulations.

3.7.2 Global Error Term and Inferred Uncertainty

As discussed in Section 3.7 of this SE, the global error term is determined differently between the inferred and calculated uncertainties. Because the detector response method calculates assembly powers in non-instrumented locations based on detector measurements, the global error term (and hence, the inferred uncertainty) will be different for each detector system, and they must therefore be determined for each detector system.

A non-instrumented location can be a fuel bundle without a detector or a fuel segment located between two fixed incore detector locations. In either case, the global error term is determined by first obtaining a power map (radial for $F_{\Delta h}$ or axial for F_Q) from all the available detector locations. Then, a detector reading at a radial or axial position is assumed not to exist, and the missing measurement is reconstructed using a reconstruction methodology that involves simulator calculated values (e.g., MEDIAN). This results in an inferred power at that location. The process is repeated for each detector location to generate an inferred power distribution. The difference between the inferred and measured distributions generates a relative difference distribution. Relative difference distributions are determined for multiple burnup points throughout a cycle, across multiple cycles and (where available) multiple plants. The aggregate of this data forms the basis for the global error term.

Once the global error term for a detector system is determined, it is combined with the local error term using both the Normal Uncertainty approach and the Non-Parametric Uncertainty approach (see Section 3.7 of this SE) to generate two inferred uncertainties at 95/95 limits. The maximum of the two results (the more restrictive and thus conservative) is taken to be the inferred uncertainty of the measurement system.

Combining local and global error terms captures uncertainties associated with predictions (e.g., APOLLO2-A methodology uncertainties) and measurements. Therefore, the inferred uncertainty is comparable to measurement system uncertainty, and the acceptance criteria Framatome uses are based on the measurement system uncertainties reported in previously licensed methodologies. Inferred uncertainties and associated acceptance criteria for the different detector systems supported by Framatome are discussed in the following subsections.

3.7.2.1 MEDIAN with Movable Fission Detector System

Framatome provided an assessment of the inferred uncertainties for a movable fission detector system. This assessment was provided in part as further validation for extending the range of applicability of the MEDIAN reconstruction methodology to moveable fission detector systems (see Section 3.2.15 of this SE). A total of 22 cycles from six plants are used for the uncertainty calculations. The mean and standard deviation of the global error term for $F_{\Delta h}$ are **[ ]**, respectively. For F_Q, the mean and standard deviation of the global error term are **[ ]**, respectively. The inferred uncertainties (the combination of local and global error terms) for $F_{\Delta h}$ and F_Q **[ ]**, respectively. The NRC staff performed an independent analysis and confirmed these results represent 95/95 1-sided upper tolerance limits. These inferred uncertainty values for $F_{\Delta h}$ and F_Q **[ ]** and will therefore be used to represent the uncertainty of the inferred power distribution using MEDIAN with moveable fission detector systems. The inferred uncertainties for $F_{\Delta h}$ and F_Q are less than the measurement system uncertainties of **[ ]**, respectively, and are therefore acceptable.

3.7.2.2 MEDIAN with Combustion Engineering Fixed Rhodium Detector System

Framatome applied the MEDIAN reconstruction methodology to CE plants that make use of fixed rhodium incore detectors. Due to the nature of this type of detector system (**[ ]**), additional components to the global error term need to be evaluated. For CE plants, the $F_{\Delta h}$ peaking factor is referred to as F_R. Considering the nature of the fixed rhodium detector system, F_R and F_Q are determined via the following equations:

[ ]

[ ]

where:

[ ]

[ ]

[ ]

[ ]

The uncertainty associated with each of the terms comprising the equations for F_R and F_Q must be determined and combined to form the global error term. The exception to this is **[ ]**.

The **[ ]** term is arguably the most representative of the nature of the rhodium fixed detector system. Therefore, Framatome uses Plant C (see Table 3.2 of this SE), the only CE plant from the suite of validation data, to determine the uncertainty associated with it. A total of 9 cycles of data are included for this plant. The standard approach of assuming a detector is absent and applying the reconstruction methodology is used. The mean and standard deviation for the term are **[ ]**, respectively.

For the **[ ]** terms, more information than what is available from **[ ]** Plant C was needed to estimate the associated uncertainties. Plants S1, S2, and A were chosen because they have moveable detector traces, and Framatome posits that these **[**

]. This approach was discussed in the NRC staff's SE for the original ARCADIA® review (Reference 3) and found to be acceptable **[** **]**. A total of 10 cycles were analyzed from these three plants. The mean and standard deviation for **[** **]**, respectively. Likewise, the mean and standard deviation for **[** **]**, respectively.

Framatome does not combine the uncertainties for these terms to generate a separate global error term. Instead, they are directly combined with the local error term when determining the inferred uncertainty. The inferred uncertainty (the combination of local and global error term components) for F_R **[** **]**, and the inferred uncertainty for F_Q **[** **]**. These results represent 95/95 1-sided upper tolerance limits. The respective inferred uncertainty values for F_R and F_Q are the larger of the two uncertainty approaches used in the analysis, and for purposes of comparing to acceptance criteria, they will be used to represent the uncertainty of the inferred power distribution using MEDIAN with the CE fixed rhodium incore detector system. The inferred uncertainties for F_R and F_Q are less than the measurement system uncertainties of **[** **]**, respectively, and are therefore acceptable.

Unlike AMS or movable fission detectors, rhodium fixed incore detectors remain in the core throughout the operating cycle. As such, these detectors experience the effects of depletion much more readily than others, and this has an impact on the detector signal. Therefore, for CE plants with rhodium fixed incore detectors, the MEDIAN reconstruction uncertainty needs to be combined with the detector signal conditioning uncertainties (i.e., detector depletion effects on various uncertainty components) to provide the uncertainty of the monitoring system. Framatome indicates in Supplement 1 that, because this uncertainty is plant-type dependent, this uncertainty will be generated during plant-specific implementation of ARCADIA®. The NRC staff finds this acceptable.

3.7.2.3 MEDIAN AMS Detector System

The MEDIAN AMS is the measurement system planned to be used in the USEPR plants. A total of 10 cycles from two plants are used for the uncertainty calculations. The mean and standard deviation of the global error term for $F_{\Delta h}$ are **[** **]** and **[** **]**, respectively. For F_Q, the mean and standard deviation of the global error term are **[** **]** and **[** **]**, respectively. The inferred uncertainties (the combination of local and global error terms) for $F_{\Delta h}$ and F_Q **[** **]**, respectively. The NRC staff performed an independent analysis and confirmed these results represent 95/95 1-sided upper tolerance limits. The inferred uncertainty values for $F_{\Delta h}$ and F_Q **[** **]** and will therefore be used to represent the uncertainty of the inferred power distribution using MEDIAN with the AMS detector system. The inferred uncertainties for $F_{\Delta h}$ and F_Q are less than the measurement system uncertainties of **[** **]**, respectively, and are therefore acceptable.

3.7.2.4 MEDIAN with B&W Fixed Rhodium Detector System

Framatome assessed the inferred uncertainty of the MEDIAN reconstruction methodology for B&W plants that use fixed rhodium incore detectors. As discussed in Section 3.7.2.2 of this SE, due to the nature of the fixed incore detector system, additional components of the global error term need to be evaluated. Evaluation of the additional components is performed the same as the CE plants discussed in Section 3.7.2.2 of this SE.

The **[ ]** term is arguably the most representative of the nature of the rhodium fixed detector system. Therefore, Framatome uses Plant T1 (see Table 3.2 of this SE), the only B&W plant from the suite of validation data, to determine the uncertainty associated with it. A total of 4 cycles of data are included for this plant. The standard approach of assuming a detector is absent and applying the reconstruction methodology is used. The mean and standard deviation for the term are **[ ]**, respectively.

For the **[ ]** terms, more information than what is available from **[ ]** Plant T1 was needed to estimate the associated uncertainties. Plants S1, S2, and A were chosen because they have moveable detector traces, and Framatome posits that these **[ ]**. This approach was discussed in the NRC staff's SE for the original ARCADIA® review (Reference 3) and found to be acceptable **[ ]**. In contrast, the CE plant discussed in Section 3.7.2.2 of this SE **[ ]**. A total of 10 cycles were analyzed. The mean and standard deviation for **[ ]**, respectively. Likewise, the mean and standard deviation for **[ ]**, respectively.

Framatome does not combine the uncertainties for these terms to generate a separate global error term. Instead, they are directly combined with the local error term when determining the inferred uncertainty. The inferred uncertainty (the combination of local and global error term components) for $F_{\Delta h}$ and F_Q **[ ]**, respectively. The NRC staff performed an independent analysis and confirmed these results represent 95/95 1-sided upper tolerance limits. The inferred uncertainty values for $F_{\Delta h}$ and F_Q **[ ]**, and for purposes of comparing to acceptance criteria, they will be used to represent the uncertainty of the inferred power distribution using MEDIAN with the B&W fixed rhodium incore detector system. The inferred uncertainties for $F_{\Delta h}$ and F_Q are less than the measurement system uncertainties of **[ ]**, respectively, and are therefore acceptable.

Unlike AMS or movable fission detectors, rhodium fixed incore detectors remain in the core throughout the operating cycle. As such, these detectors experience the effects of depletion much more readily than others, and this has an impact on the detector signal. Therefore, for B&W plants with rhodium fixed incore detectors, the MEDIAN reconstruction uncertainty needs to be combined with the detector signal conditioning uncertainties (i.e., detector depletion effects on various uncertainty components) to provide the uncertainty of the monitoring system. Framatome indicates in Supplement 1 that, because this uncertainty is plant-type dependent, this uncertainty will be generated during plant-specific implementation of ARCADIA®. The NRC staff finds this acceptable.

3.7.3 Global Error Term and Calculated Uncertainty (Nuclear Reliability Factors)

The calculated uncertainties, which Framatome refers to as the nuclear reliability factors (NRFs), for $F_{\Delta h}$ and F_Q are determined similar to the inferred uncertainties described in the previous section. The difference between the two uncertainties is that the global error term of the inferred uncertainty is a measure of how well the instrument readings at non-instrumented locations are predicted, whereas in the NRFs, it is a measure of how well the simulator can predict the assembly power and instrumented locations. The global error term for the NRFs are calculated by comparing the detector-measured powers versus the simulator-calculated powers. As with the inferred uncertainty, the NRF 95/95 one-sided tolerance limits are determined by combining the global and local error terms using the Normal Uncertainty approach and the Non-Parametric Uncertainty approach. The more restrictive of the two is selected as the bounding uncertainty.

All plants modeled in the revalidation suite (see Table 3.2 of this SE) if Supplement 1 are used to obtain the global error term for the NRFs. This consists of 6 different PWR reactor types across 10 different plants with 7 different fuel types for a total of 58 cycles of operation. The mean and standard deviation of the global error term for $F_{\Delta h}$ are **[ ]**, respectively. For F_Q, the mean and standard deviation of the global error term are **[ ]**, respectively. The NRFs (the combination of local and global error terms) for $F_{\Delta h}$ and F_Q **[ ]**, respectively. The NRC staff performed an independent analysis and confirmed these results represent 95/95 1-sided upper tolerance limits. The NRFs for $F_{\Delta h}$ and F_Q **[ ]** and will therefore be used to represent the uncertainty of the calculated power distribution. The NRFs for $F_{\Delta h}$ and F_Q are less than the current licensing values of approximately **[ ]**, respectively, and are therefore acceptable.

3.7.4 Summary of Power Distribution Uncertainty

The inferred and calculated (NRF) uncertainties presented in Supplement 1 and discussed in the preceding sections were determined for a 95/95 one-sided upper tolerance limit using both the Normal Uncertainty Approach and the Non-Parametric Uncertainty approach. The more restrictive of the two approaches was chosen as the uncertainty. During the calculations, the global and local error term distributions were conservatively combined (e.g., positive biases were ignored if greater than 0, and the magnitude of negative biases retained) such that the upper tolerance limit represents the percentage by which the power distributions are under-predicted (i.e., 95 percent confident that 95 percent of calculations under-predict the measurement by ‘X’ percent or less).

Because it is conservative to over-predict the power distribution, Framatome converted the upper tolerance limits for the inferred and calculated uncertainties into decimal multipliers (e.g., 2.8 percent to 1.028). These multipliers and the equivalent acceptance criteria are summarized in Table 3.5, below.

Table 3.5: Power Distribution Uncertainty Summary.

Type of Uncertainty	$F_{\Delta H}$ (F_R for CE)		F_Q	
	Estimated[1]	Criteria	Estimated[1]	Criteria
Inferred Movable Fission	[			]
Inferred CE Rhodium[1]	[			]
Inferred B&W Rhodium[1]	[			]
Inferred AMS	[			]
Calculational (NRF)	[			]

[1]Uncertainties have been rounded up

[2]The uncertainty for these detector systems requires the inclusion of detector signal conditioning uncertainties on a plant-specific basis (see Sections 3.7.2.2 and 3.7.2.4 of this SE)

As shown in Table 3.5, all the inferred and calculated uncertainties for ARCADIA® power distribution predictions are less than the current licensing values used for the acceptance criteria.

3.7.5 Requested Removal of Limitation 3 of the ARCADIA® Safety Evaluation

On page 28 of the NRC's SE for the original ARCADIA® TR (ANP-10297P-A, Reference 3), the NRC staff imposed the following limitation:

> The ARCADIA® code system is limited to fuel types with non-Inconel grids unless additional verification and validation of uncertainties is conducted to account for any peaking biases due to grid type or other plant effects. Verification of uncertainties must be quantified and accounted for in the uncertainties and/or peaking allowances in the licensing calculations on a plant-specific basis.

This is Limitation No. 3 within the NRC staff's SE. This limitation was introduced in response to a statement made on pages 12-12 and 12-13 of the original ARCADIA® TR, which stated that assemblies containing Alloy-718 (Inconel) grids were not included in the peak statistics (assemblies from Plant B and Plant G2 Cycles 1 and 2). At the time of the review, and at present, the NRC staff's position was that peak prediction uncertainties need to be addressed for each grid type and plant. A letter of clarification was later incorporated into the original ARCADIA® TR which notes Limitation No. 3 prohibits the application of the ARCADIA® code system to fuel types possessing grids made entirely of Inconel in the active region of the fuel, but allows application of the code system to fuel types possessing grids made with Inconel springs (e.g., Bi-Metallic grids) in the active region of the fuel and to fuels with grids made entirely of Inconel at the top and bottom of the fuel.

In Supplement 1, Framatome requested the removal of Limitation No. 3. In support of this request, the analyses presented in Supplement 1 for the inferred and calculated uncertainties were performed using data for assemblies with all Alloy-718 grids, bi-metallic grids, and all Zircaloy or M5 grids in the active region of the fuel. Thus, the impact of the presence of all Alloy-718 spacer grids is included in the peak statistics.

Table 3.5 of this SE summarizes these statistics for the detector systems supported by Framatome. The results show that uncertainties for the supported detector systems remain well within the acceptance criteria. Therefore, the NRC staff finds the basis for the introduction of Limitation No. 3 has been satisfied. Limitation No. 3 can therefore be removed.

3.8 Safety Parameter Uncertainties

Total rod worth and ITCs are typically measured and calculated as part of reactor startup during physics testing. Comparisons of these parameters were included in Supplement 1 as part of the validation of the ARTEMIS™ neutronics models. However, total rod worth and ITC are also used in safety analyses. Generally, uncertainties are applied to these parameters to assure that a conservative analysis is performed. Therefore, Supplement 1 includes a statistical analysis of total rod worth and ITC to generate uncertainty values for these parameters that can be applied when evaluating cores during safety analyses.

For the statistical analysis, the data provided for the startup physics test measurements were supplemented with an additional 14 cycles of data for a total of 72 cycles of operation across 6 different PWR reactor types and 7 different fuel types. The additional cycles come from Plant A (Cycles 15-18), Plant S1 (Cycles 15-20), and Plant S2 (Cycles 15-20). The NRC staff examined these additional cycles and observed the total control rod bank worths and ITCs meet the acceptance criteria discussed in Section 3.4.1 of this SE and the results are consistent with the existing cycle results. Therefore, the NRC staff finds the inclusion of the additional cycles in the present statistical analysis acceptable.

The 6 different PWRs comprising the data set contain a total of 10 different plants. Because Plant B contains only 1 cycle of data, it was combined with a plant of the same type and fuel, Plant A. Similarly, Plants S1 and S2 were combined to form one group because they also represent the same plant type and fuel type. The NRC staff finds the groupings of these plants acceptable on this basis. The resulting dataset therefore represents 8 different plant/fuel combinations.

To find the uncertainties associated with total rod worth and ITC, Framatome determines a mean, standard deviation, and weighting factor for each of the 8 plant/fuel combinations. The weighting factors are used when combining the individual distributions into a single set of data from which both the upper and lower 95/95 one-sided tolerance limits are found. The larger of the one-sided tolerances is taken to represent the uncertainty in the parameters. Using this methodology, the uncertainties associated with the total rod worth and ITC are **[** **]**, respectively.

Aggregating of data in the manner discussed above requires the data be normally distributed. Framatome performed a statistical test, known as the D' test, on the total rod worth data and on the ITC data to determine if the data sets are not normally distributed. In both instances, the assumption of normality was not rejected. The use of a weighting function captures the effects that each plant/fuel combination (and the amount of available data for each) will have on the total rod worth and ITC uncertainty. Additionally, the more restrictive of the upper and lower one-sided tolerance limits is used for the parameter uncertainty, which is a conservative approach. Therefore, for the range of plant/fuel combinations examined in the statistical analysis, the NRC staff finds the total rod worth and ITC uncertainties of **[** **]**, respectively, to be acceptable when evaluating cores for safety analyses using the ARCADIA® code system.

3.9 Summary and Qualification Method

3.9.1 Range of Applicability

Supplement 1 presents the intended application range for the ARCADIA® code system. A summary of the various plants, fuels, burnable absorbers, control rods, and detector systems included in the revalidation suite is provided to demonstrate the flexibility of the code system. As a concluding statement, Framatome states, "ARCADIA® is applicable to all PWR plant types, square lattice designs, burnable absorbers, and control rod types." This a broad statement that is not consistent with Limitation No. 1 from the NRC staff's SE of the original ARCADIA® review, which states:

> The range of applicability of the ARCADIA® code system Methodology is restricted to the fuel data provided in the TR, unless additional analysis and benchmarking is conducted to validate the ARCADIA® code system to a fuel type not mentioned in the TR.

The revalidation suite provided in Supplement 1 is representative of the vast majority of operating PWR plants, fuels, burnable absorbers (including IFBA and WABA), rod types, conditions, etc. in the U.S., but is not necessarily all-encompassing and is certainly not inclusive of future designs. Therefore, the NRC staff finds the range of applicability of the ARCADIA® code system remains restricted, but now to that of the expanded fuel data provided in Supplement 1. A summary of this range is provided in Table 10-1 of Supplement 1. The range of applicability also includes core designs of UO_2 enrichments up to 5.0 wt% U-235 and ERU enrichments up to 4.01 wt% U-235, gadolinia enrichments up to 10 wt%, and the following detector systems: moveable incore fission detectors, CE fixed rhodium detectors, B&W fixed rhodium detectors and Aeroball incore detectors (AMS). Additionally, the ARCADIA® code system, as presented in Supplement 1, may be considered applicable for analyses up to 71 GWd/MTU rod average burnup. The range of burnup is based on the NRC staff's assessment of the capability of APOLLO2-A to predict nuclear data. However, this should not be interpreted as allowance to operate fuel beyond currently licensed burnup limits as explicitly stated (e.g., in the latest approved version of COPERNIC, Reference 17) and implicitly indicated throughout Framtome's licensing methodology.

3.9.2 Plant Application

Framatome indicates the criteria presented in Table 10-2 of Supplement 1 will be used when applying the ARCADIA® code system. The criteria listed in Table 10-2 are consistent with the ANSI/ANS Reload Startup Physics standard with the exception of the total control rod worth and the ITC. The total control rod worth and ITC uncertainties are [±7 percent and ±0.6 pcm/ºF], respectively. The development of these two uncertainties is discussed in Section 3.8 of this SE. Framatome also indicates it will continue to monitor its methods with respect to current cycle designs for licensing applications, and will evaluate at least three full cycles of data against the Table 10-2 criteria of Supplement 1 prior to the first use of ARCADIA® at a plant. This includes verification of the power measurement uncertainties and/or calculational uncertainties by using the appropriate method as presented in Section 12 of the original ARCADIA® TR or Section 8 of Supplement 1. Additionally, Framatome will make comparisons to measured plant data as needed (e.g., application of ARCADIA® to a current cycle) to ensure there is no drift in the ARCADIA® models used in plant calculations. The NRC staff finds this acceptable.

3.9.3 Code Modification Change Process

In Section 10.4 of Supplement 1, Framatome presents a change process by which the ARCADIA® code system is intended to be modified. It is stated that there are many situations that might require a change to codes or libraries, some of which are related to maintaining modern and robust computer codes (e.g., through the introduction of first principle models rather than empirical models). Essentially, the purpose of the process is to maintain the ARCADIA® code system as state-of-the-art. However, many of the desired changes and modifications discussed in the change process are intended to be implemented without prior NRC review and approval. While the NRC staff understands the desire to maintain a code as state-of-the-art (and openly acknowledges the benefits this can reap), the NRC staff must balance this with sufficient regulatory oversight to ensure that core analysis codes provide a realistic or conservative result and that all applicable regulatory requirements are met. As such, the NRC staff reviewed Framatome's proposed change process for the ARCADIA® code system and assessed the impact such changes might have on the code system's regulatory compliance.

The change process presented in Supplement 1 indicates that, anytime a code modification is made, the following four steps are taken: 1) the change is documented, 2) test cases are performed (these include regression testing), 3) documentation for theory and user manuals is updated, and 4) a report will be generated annually summarizing all changes made within a year and provided to the NRC. The test cases performed in step 2 include a subset of the original (updated) validation suite presented in Supplement 1, the purpose of which is to capture the effects of the change and demonstrate adherence to the validation suite acceptance criteria (Table 10-2 of Supplement 1). The analysis is also intended to show the power distributions remain within the currently reviewed and approved uncertainties (presented in Table 8-24 of Supplement 1 and Table 3.5 of this SE). All changes are examined with regard to limitations specified in the NRC's SE. If the changes result in improvements outside the NRC limitations, a number of actions may be taken, such as a license amendment request for the changes on a plant-specific basis or the presentation of changes to the NRC for acceptance (e.g., additional ARCADIA® code system supplements).

The change process outlined in Supplement 1 demonstrates due-diligence in ensuring code functionality and applicability is retained whenever a change is made and an adherence to a quality assurance program. However, NRC staff assessment of similar change processes (e.g., AURORA-B, Reference 18) has demonstrated that many of the requested changes contain nuances that may not be consistent with the reviewed and approved methodology described in the TR and would therefore warrant further examination.

This SE documents the NRC staff's approval of the methodology described in ANP-10297P, as supplemented. The versions of the computer codes, the inputs, the nodalizations, etc. (i.e., the complete calculational model) used to generate the output presented in the TR are one implementation of the approved methodology described in the TR. The NRC staff recognizes that some changes to the complete calculational model associated with the approved methodology may be necessary during the application. Such changes are considered acceptable provided they are consistent with the methodology described in the TR and do not invalidate the staff's SE. Changes may be considered consistent with the methodology described in the TR even if they would result in a change to the description of the methodology provided in the TR. If the change results in a revision to the description of the methodology in the TR, then NRC concurrence will be necessary to conclude that the change is consistent with the methodology in the TR.

In some instances, Framatome may clearly determine a change is consistent with the approved methodology and does not invalidate the NRC staff's SE. However, in other instances, the determination may not be as clear. In such instances, Framatome may submit descriptions of a change to the NRC for confirmation that the change is within the scope of the approved methodology. Reference 19 and Reference 20 provide examples of this approach. The descriptions should include details on the change itself, justification that the change is consistent with the approved methodology, and demonstration that the results of the modified code are consistent with those of prior code versions. Based on previous submittals, the NRC staff anticipates a 3-month turnaround for confirmations.

Any changes must ensure that the validation suite acceptance criteria (Table 10-2 of Supplement 1) remain applicable. Should the NRC issue a report updating the NRC staff's position regarding change processes in general, such a report would supersede the above discussion.

4.0 LIMITATIONS AND CONDITIONS

As discussed previously in this report, limitations and conditions have been applied to the ARCADIA® code system as part of the review of the original TR. It is not unusual for limitations and conditions established in a prior NRC staff SE to be repealed or modified by the additional justification provided in a later supplement. Supplement 1 is such a case. For the purpose of clarity, the following table (Table 4.1) demonstrates the relationships that exist between previous limitations and conditions and the comprehensive list provided herein.

Table 4.1: ARCADIA® Code System Limitations and conditions Applied by NRC Staff SEs

ANP-10297P-A, Revision 0	Supplement 1
(1) The range of applicability of the ARCADIA® methodology is restricted to the fuel data provided in the TR, unless additional analysis and benchmarking is conducted to validate ARCADIA® to a fuel type not mentioned in the TR.	Applicability Retained
(2) The benchmarks provided in the ARCADIA® TR include uncertainty verification for plants that use moveable incore, rhodium fixed incore, and Aeroball incore detectors. Framatome will evaluate at least three cycles of data relative to these criteria prior to licensing the first cycle with Framatome fuel with ARCADIA®. Additionally, application of ARCADIA® to a new uncertainty measurement system(s) would require review and approval by the NRC staff prior to implementation.	Applicability Retained

ANP-10297P-A, Revision 0	Supplement 1
(3) ARCADIA® is limited to fuel types with non-Inconel grids unless additional verification of uncertainties is conducted to account for any peaking biases due to grid type or other plant effects.	Removed
(4) For any changes made to the stand-alone version of COBRA-FLX™ that is implemented in ARCADIA® (the COBRA-FLX™ module), Framatome will revalidate ARCADIA® output using measured data from multiple plants and cycles.	Applicability Retained
	(5) The NRC staff finds ARTEMIS™ acceptably models the best estimate neutronic time-dependent transient responses (e.g., power response to changes in Doppler, moderator, etc.), and that it is an acceptable tool for use in an evaluation model for non-LOCA SRP Chapter 15 events. However, use of ARTEMIS™ in an evaluation model for such events requires consideration of bounding conditions, inputs, limits, time-step sensitivities, etc. which are not included in Supplement 1. Therefore, as implied for ANP-10297P-A, Revision 0, February 2013, this SE does not constitute approval of ARTEMIS™ as a stand-alone evaluation model for non-LOCA SRP Chapter 15 events. NRC review and approval of an associated evaluation methodology using ARTEMIS™ is required prior to its use in non-LOCA SRP Chapter 15 event licensing analyses.

ANP-10297P-A, Revision 0	Supplement 1
	(6) Any changes made to the ARCADIA® code system must: a. ensure the validation suite acceptance criteria (Table 10-2 of Supplement 1) remain applicable, b. be consistent with the methodology described in ANP-10297P, as supplemented, and c. not invalidate the NRC staff's SE. In instances where it is unclear if a change is consistent with the approved methodology, Framatome may submit descriptions of a change to the NRC for confirmation that the change is within the scope of the approved methodology, as discussed in Section 3.9.3 of this SE.

4.1 Limitations and Conditions Applicable to Supplement 1

The summary of all limitations and conditions applicable to the ARCADIA® code system are restated below. Captions in italics indicate the source. The numbering scheme uses the ANP-10297P-A, Revision 0, limitations and conditions numbers (1-4) that are still applicable (1-2, and 4) supplemented by those that are added by this SE for Supplement 1 (5-8).

Limitations and Conditions Nos. 6 and 7 address the ARCADIA® code system change process presented in Section 10.4 of Supplement 1. Any changes to the calculational methodology, numerical methods, underlying principles, bases, assumptions, range of applicability, etc., are subject to these limitations and conditions.

1. The range of applicability of the ARCADIA® methodology is restricted to the fuel data provided in the TR, as supplemented, unless additional analysis and benchmarking is conducted to validate ARCADIA® to a fuel type not mentioned in the TR, as supplemented. *(This is Condition 1 of the SE for the original ARCADIA® TR (ANP-10297P-A, Revision 0). It remains applicable to Supplement 1, and it has been updated to include the expanded range of fuel data presented within Supplement 1).*

2. The benchmarks provided in the ARCADIA® TR, as supplemented, include uncertainty verification for plants that use moveable incore, rhodium fixed incore, and Aeroball incore detectors. Framatome will evaluate at least three cycles of data relative to these criteria prior to licensing the first cycle with Framatome fuel with ARCADIA®. Additionally, application of ARCADIA® to a new uncertainty measurement system(s) would require review and approval by the NRC staff prior to implementation. *(This is Condition 2 of the SE for the original ARCADIA® TR (ANP-10297P-A, Revision 0). It remains applicable to Supplement 1, and it has been updated to include the incore detector systems presented within Supplement 1).*

4. For any changes made to the stand-alone version of COBRA-FLX™ that is implemented in ARCADIA® (the COBRA-FLX™ module), Framatome will revalidate ARCADIA® output using measured data from multiple plants and cycles. *(This is Condition 4 of the SE for the original ARCADIA® TR (ANP-10297P-A, Revision 0). It remains applicable to Supplement 1).*

5. The NRC staff finds ARTEMIS™ acceptably models the best estimate neutronic time-dependent transient responses (e.g., power response to changes in Doppler, moderator, etc.), and that it is an acceptable tool for use in an evaluation model for non-LOCA SRP Chapter 15 events. However, use of ARTEMIS™ in an evaluation model for such events requires consideration of bounding conditions, inputs, limits, time-step sensitivities, etc., which are not included in Supplement 1. Therefore, as implied for ANP-10297P-A, Revision 0, this SE does not constitute approval of ARTEMIS™ as a stand-alone evaluation model for non-LOCA SRP Chapter 15 events. NRC review and approval of an associated evaluation methodology using ARTEMIS™ is required prior to its use in non-LOCA SRP Chapter 15 event licensing analyses. *(This is a new condition from this SE).*

6. Any changes made to the ARCADIA® code system must:

    a. ensure the validation suite acceptance criteria (Table 10-2 of Supplement 1) remain applicable,

    b. be consistent with the methodology described in ANP-10297P, as supplemented, and

    c. not invalidate the NRC staff's SE.

    In instances where it is unclear if a change is consistent with the approved methodology, Framatome may submit descriptions of a change to the NRC for confirmation that the change is within the scope of the approved methodology, as discussed in Section 3.9.3 of this SE. *(This is a new condition from this SE).*

5.0 CONCLUSIONS

In Supplement 1, Framatome presented new models and made modifications to existing models in the APOLLO2-A lattice physics code and the ARTEMIS™ core code in order to extend the range of applicability of the ARCADIA® code system. The NRC staff reviewed these new models and modifications. The documentation provided in the TR and the responses to the RAIs demonstrate that the analytical equation set and numerical method solution methods are correctly implemented. Though a number of modifications were made to the ARTEMIS™ core code for coupling to an approved system code (e.g., S-RELAP5), Framatome did not seek approval for the coupling of ARTEMIS™ to an approved system code within Supplement 1. As such, the NRC staff does not take issue with the modifications made to ARTEMIS™ to incorporate additional inputs, but neither does the NRC staff approve the coupling of ARTEMIS™ to a system code. The acceptability of coupling ARTEMIS™ to an approved system code and its application to transient analyses would need to be reviewed and approved in a separate submittal.

Based on the technical evaluation presented in this SE, the NRC staff finds the ARCADIA® code system methodology for PWRs as supplemented by ANP-10297P-A, Revision 0, Supplement 1, acceptable for licensing applications subject to the limitations and conditions specified in Section 4.0 of this SE. Specifically, the NRC staff found acceptable the expansion of the ARCADIA® code system's range of applicability to include ERU fuel; application of the code system to movable fission detector systems, CE and B&W fixed rhodium detector systems, and AMS detector systems through use of the MEDIAN reconstruction methodology; and the removal of Limitation No.3 that was established in the NRC staff's SE for ANP-1029P-A, Revision 0.

6.0 REFERENCES

1. Request for Review and Approval of ANP-10297P-A, Revision 0, Supplement 1, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results Topical Report," June 26, 2015, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15187A263 (Publically Available)).

2. ANP-10297P-A, Revision 0, Supplement 1, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," June 2015, (ADAMS Accession No. ML15187A268/ML15187A269 (Publically Available/Non-Publically Available)).

3. ANP-10297P-A, Revision 0, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," February 2013 (ADAMS Accession No. ML14195A145).

4. NUREG-800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor Edition" (ADAMS Accession No. ML070810350 (Publically Available)).

5. ANP-10297 R0, Supplement 1, R0 Q2P, Revision 0, "Additional Information – ANP-10297PA, Supplement 1, 'The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results,'" March 2018 (ADAMS Accession No. ML18072A097 (Publically Available)).

6. A. J. Koning and D. Rochman, "Modern Nuclear Data Evaluation with the TALYS Code System," Nuclear Data Sheets, Volume 113, Page 2841 (2012).

7. Audit Summary of the Audit Conducted January 18, 2018, "Audit to Close Outstanding Technical Issues for ANP-10297P(A), Revision 0, Supplement 1, 'The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results (TAC No. MF6469),'" January 2018 (ADAMS Accession No. ML18025A875 (Publically Available)).

8. A. Santamarina, D. Bernard, P. Leconte, and J-F. Vidal, "Improvement of ^{238}U Inelastic Scattering Cross-Section for an Accurate Calculation of Large Commercial Reactors," Nuclear Data Sheets, Volume 118, Pages 118-121 (2014).

9. M. Salvatores and R. Jacqmin, "Uncertainty and Target Accuracy Assessments for Innovative Systems Using Recent Covariance Data Evaluations," NEA/WPEC-26, ISBN 978-92-64-99053-1. OECD 2008.

10. A. Santamarina, D. Bernard, N. Dos Santos, O. Leray, C. Vaglio, and L. Leal, “Re-estimation of Nuclear Data and JEFF3.1.1 Uncertainty Calculations,” PHYSOR 2012, April 15-20 (2012).

11. ANP-10297Q1P, Revision 1, “Responses to Requests for Additional Information for the Review of The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results Topical Report,” April 2011 (ADAMS Accession No. ML11143A107/ML11143A115 (Publically Available/Non-Publically Available)).

12. Audit Summary of the Audit Conducted Between April 28, 2016 and April 29, 2016, “Audit for Understanding for ANP-10297P(A) Supplement 1 – The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results (TAC No. MF6469),” May 2016 (ADAMS Accession No. ML16131A404 (Non-Publically Available)).

13. ANP-10297 R0, Supplement 1, R0 Q1P, Revision 0, “Response to Request for Additional Information – ANP-10297PA, Supplement 1, ‘The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results,’” May 2017 (ADAMS Accession No. ML17153A046/ML17153A047 (Publically Available/Non-Publically Available)).

14. ANP-10287P, Revision 2, “Incore Trip Setpoint and Transient Methodology for U.S. EPR”, August 2014 (ADAMS Accession No. ML14220A193 (Non-Publically Available)).

15. Martinolli, E., Carter, T.C., Clement, F., Demy, P.M., Leclere, M., Magat, P., Marquis, A., Marotte, V., Schneider, M., Villatte, L., Marten, J., Misu, S., & Thareau, S., “APOLLO2-A – Framatome’s New Generation Lattice Physics Code: Methodology and Validation,” PHYSOR 2010, Pittsburgh, Pennsylvania, USA, May 9-14 (2010).

16. ANSI/ANS-19.6.1-2005, “Reload Startup Physics Tests for Pressurized Water Reactors,” American Nuclear Society, 2005.

17. BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code”, January 2004 (ADAMS Accession No. ML042930250 (Non-Publically Available)).

18. ANP-10300P, Revision 0, “AURORA-B: An Evaluation Mode for Boiling Water Reactors; Application to Transient and Accident Scenarios,” December 2009 (ADAMS Accession Nos. ML100040158/ML100040159 (Publically Available/Non-Publically Available)).

19. Hsueh, K., NRC, letter to Gary Peters, AREVA Inc., “Request for U.S. Nuclear Regulatory Commission Concurrence Regarding Applicability of Topical Report BWR-2149A, “Evaluation of Replacement Rods in BWFC Fuel Assemblies to Cores Containing Fresh Fuel Assemblies with Stainless Steel Rods”,” February 1, 2017 (ADAMS Accession No. ML16357A669 (Publically Available)).

20. Hsueh, K., NRC, letter to Gary Peters, “AREVA’s Request for U.S. Nuclear Regulatory Commission Concurrence Regarding Acceptability of the Standard Friction Factor Correlation within ANP-10311P-A, ‘Cobra-FLX: A Core Thermal-Hydraulic Analysis Code’,” October 18, 2017 (ADAMS Accession No. ML17249A888 (Publically Available)).

Attachment: Comment Resolution Table

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