U.S. NUCLEAR REGULATORY COMMISSION

SAFETY EVALUATION FOR

STUDSVIK TOPICAL REPORT SSP-14-P01/028-TR-S1 REVISION 0, SUPPLEMENT 1,

"GENERIC APPLICATION OF THE STUDSVIK SCANDPOWER CORE MANAGEMENT

SYSTEM TO PRESSURIZED WATER REACTORS:

SUPPLEMENT FOR EXTENDED ENRICHMENT, BURNUP, AND SMRS"

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1.0 INTRODUCTION

By letter dated January 28, 2025 (Ref. 1, Agencywide Documents Access and Management System (ADAMS) Accession No. ML25028A254), Studsvik Scandpower, Inc. (Studsvik) submitted Topical Report (TR) SSP-14-P01/028-TR Supplement 1, "Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors: Supplement for Extended Enrichment, Burnup, and SMRs" (Ref. 2, ML25028A256), to the U.S. Nuclear Regulatory Commission (NRC) for review and approval for reference in licensing applications. SSP-14-P01/028-TR, Supplement 1, extends the range of applicability of SSP-14-P01/028-TR, "Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors" (Ref. 3) to include increased enrichment (IE), higher burnup (HBU), and generic application to light-water, pressurized small modular reactors (SMRs). Specifically, Studsvik requested an extension of the uranium enrichment and maximum rod-average burnup range of applicability from the current limits to 10 weight percent (wt%) uranium-235 (U-235 (U-235) and 80 gigawatt-day per metric ton of uranium (GWd/MTU), respectively, and for generic applicability to light-water, pressurized SMRs. As discussed below, however, Studsvik later requested that the NRC staff limit its review to 8 wt% U-235 enrichment.

2.0 REGULATORY EVALUATION

The Core Management System 5 (CMS5) code system is a modern production core analysis tool that can be applied generically to modeling and analysis of pressurized water reactor (PWR) cores. The verification and validation for the application of the CMS5 code system to IE and HBU conditions is described in SSP-14-P01/028-TR, Supplement 1 (Supplement 1). The NRC staff reviewed the TR to evaluate the applicability of CMS5 to IE and HBU, by verifying that the results of the analyses show that CMS5 is an acceptable tool for a licensee to use to determine that its core design meets the General Design Criteria (GDC) specified in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50. The NRC staff review is based on guidance contained in Chapter 4.3, Rev. 3, "Nuclear Design," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (Ref. 4). Based on NUREG-0800, Chapter 4.3, the NRC staff determined that the following GDC are applicable:

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- GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.

- GDC 11 requires that, in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

- GDC 20 requires automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety occurs under accident conditions. There are usually primary and secondary independent reactivity control systems.

- GDC 26 requires that two independent reactivity control systems of different designs be provided, and that each have the capability to control the rate of reactivity changes resulting from planned, normal power changes. One of the systems must be capable of reliably controlling anticipated operational occurrences. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.

The above-referenced GDCs are generally hardware requirements that pertain to the core design, and the CMS5 code system is used to demonstrate that the core design satisfies these criteria.

The NRC staff reviewed the areas concerning analytical methods. These areas are:

- Descriptions in SSP-14-P01/028-TR-P-A of the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, and burnup.

- The database and/or nuclear data libraries used in Supplement 1 to SSP-14-P01/028-TR-P-A for neutron cross-section data and other nuclear parameters, including delayed neutron and photoneutron data and other relevant data.

- Verification and validation of the analytical methods used in Supplement 1 to SSP-14-P01/028-TR-P-A for comparison with measured or calculated data.

NUREG-0800, Chapter 4.3, Section II, states, in part, that there are no specific criteria that must be met by the analytical methods or data that are used by an applicant or reactor vendor. In general, the analytical methods and databases should be representative of the state of the art, and the experiments used to validate the analytical methods should be adequate representations of fuel designs in the reactor and encompass a sufficient range of variables and operating conditions such that the full range of applicability is covered or justified. Analytical methods or code predictions should be consistent with the typical measurement uncertainties of the associated parameters to ensure the code remains within acceptable performance bounds. This ensures that the uncertainties and other assumptions applied within the safety analysis, operating setpoints, and technical specifications limits will restrict allowable reactor core operation to conditions that have been demonstrated to be in compliance with applicable regulatory requirements. Additionally, ongoing use of analytical methods provides continued confirmation that code predictions are reasonably consistent with system measurements. Therefore, the NRC staff focused its review on the applicant's approach to justifying acceptable performance of the code system to the requested range of applicability.

3.0 TECHNICAL EVALUATION

3.1 Introduction

The CMS5 code system is a neutronics code system applicable to PWR steady state core modeling. It consists of the lattice physics code CASMO5, cross-section functionalization code CMSLINK5, and nodal core simulator SIMULATE5. The CMS5 suite also includes a methodology for developing Nuclear Uncertainty Factors (NUFs), and a set of generic Nuclear Reliability Factors (NRFs) from applicable operating data that any plant within the range of applicability may use without further justification. CMS5 has been previously reviewed and approved by the NRC in SSP-14-P01/028-TR-P-A (Ref. 3) for fuel enrichments up to 5 wt% U-235 and maximum rod-average burnups of 62 GWd/MTU. Supplement 1 to SSP-14-P01/028-TR-P-A seeks to extend the range of applicability of CMS5 from its current limits to 10 wt% U-235 and 80 GWd/MTU maximum rod-average burnup. It also seeks generic applicability to light water pressurized SMRs. As discussed below, however, Studsvik later requested that the NRC staff limit its review to 8 wt% U-235 enrichment.

Studsvik indicates that no changes to the CMS5 methodology were made since the approval of SSP-14-P01/028-TR-P-A (Ref. 2). However, in the review of Supplement 1, the NRC staff took into consideration the base methodology, its applicability to the requested range of approval based on consistent performance and quantified uncertainties, and any potential limitations that could restrict the range of applicability. The NRC staff focused its review on the validation and verification presented in Supplement 1 and examined the results for any trends or biases that would indicate degrading predictive accuracy at conditions beyond the current licensed range of applicability.

A discussion of the application of CASMO5, CMSLINK5, and SIMULATE5 to IE and HBU is given in Sections 3.2, 3.3, and 3.4 of this SE, respectively. A discussion of the generic application of CMS5 to light water pressurized SMRs is in Section 3.5 of this SE. To extend CMS5 to IE and HBU conditions, Studsvik performed additional verification and validation using critical benchmarks and code-to-code comparisons, including comparisons to higher order methods such as SERPENT2 and MCNP6 (Monte Carlo N-Particle). SERPENT2 and MCNP6 are Monte Carlo transport codes that are higher order methods in comparison to the CMS5 code system because of their continuous energy or highly detailed X-group energy spectra and their explicit method of tracking particles across a wide variety of geometries without requiring simplifications or assumptions. The NRC staff notes that the depletion algorithms in Monte Carlo transport codes are numerical methods, and in this regard, are not considered a higher order method in comparison to the CMS5 code system.

3.2 CASMO5

CASMO5 is a two-dimensional (2D) multigroup transport theory lattice physics code for PWRs and boiling water reactors (BWRs). Nuclear data are collected in a library containing microscopic cross sections in 586 energy groups. CASMO5 can model cylindrical fuel rods in a square pitch array, asymmetric and symmetric fuel bundles, reflectors and baffles, and absorber or water rods with different pin cell positions. CASMO5 models thermal expansion of dimensions and densities, individually calculates effective resonance cross sections for each fuel pin, and depletes each fuel and burnable absorber pin with a predictor-corrector method that couples the burnup calculations to the neutron transport solution.

3.2.1 Application to Increased Enrichment

Studsvik states that no methodological changes to CASMO5 are necessary to support its application to higher enrichments. As technical justification for the enrichment range extension, Studsvik provided comparisons to relevant critical benchmarks and higher order methods.

3.2.1.1 Benchmarks

Studsvik chose five critical experiments with increased fuel enrichment from the International Criticality Safety Benchmark Evaluation Project (ICSBEP) to benchmark CASMO5. Four were conducted at the Kurchatov Institute in the 1960s, and the fifth is from the Otto-Hahn program. The NRC staff reviewed the range of enrichments, pin pitches, and pin geometries covered by the experiments and concludes that they sufficiently cover the requested range of applicability. Studsvik used all experiments for reactivity comparisons. Studsvik also used the Otto-Hahn experiment for fission rate comparisons. Studsvik used MCNP6 to calculate the axial buckling to enable suitable 3D eigenvalue comparisons between CASMO5 and MCNP6 because CASMO5 is a 2D code and lacks the ability to calculate axial buckling. Studsvik notes that both CASMO5 and MCNP6 use the same evaluated nuclear data, and as such, the differences in the reactivity would represent the difference in numerical approximations and treatments of axial leakage between the codes.

The NRC staff reviewed Studsvik’s results for the Otto-Hahn experiments. CASMO5 exhibits good agreement with MCNP6 for both the eigenvalue and 2D fission rate comparisons. When assessing the CASMO5-to-experiment results for the Kurchatov Institute critical benchmarks provided in Supplement 1, the NRC staff considered the previous benchmark results from SSP-14-P01/028-TR-P-A (Ref. 3). The NRC staff noted that the eigenvalue comparisons provided in Supplement 1 are larger in magnitude and have a wider variance than what was previously observed in SSP-14-P01/028-TR-P-A, suggesting a bias and increased variance with respect to IE. The bias and increased variance could affect quantified NRFs, which Studsvik did not change or revalidate on the basis that CMS5 does not exhibit any significant biases or increased uncertainty in the extended range of applicability. The NRC staff asked Studsvik in a Request for Additional Information (RAI) (Ref. 5) for justification that the NRFs remain bounding. Studsvik explained in its response (Ref. 6) that MCNP6 exhibited significant differences from some of the experiments, suggesting that the benchmark models may not fully capture the experimental conditions. Studsvik also noted that using P_3 scattering in CASMO5 improved the accuracy of the predictions, further suggesting that the discrepancies in eigenvalue results are attributed to the geometrical configuration of the critical experiment rather than a result of IE of the fuel. Additionally, Studsvik states that there are no pronounced trends within a specific experimental series.

After reviewing the critical benchmark documentation for the Kurchatov Institute experiments (Ref. 7), the NRC staff concluded that it is likely there are characteristics (e.g., operating conditions, geometric configurations) of both the benchmark model and the experiments themselves that were not thoroughly documented and are therefore difficult to reproduce in CASMO5. For example, experimental uncertainties are reported, but not all components of the experimental uncertainty were addressed by the authors. The generally good agreement between CASMO5 P_3 scattering and MCNP6, which has been well benchmarked and validated, suggests that CASMO5 P_3 predicts well. The improved agreement from CASMO5 Transport-Corrected P_0 to CASMO5 P_3 supports the argument that the apparent increases in bias and variance can be attributed to geometry rather than higher enrichment. Additionally, confirmatory calculations performed by Oak Ridge National Laboratory (ORNL) for the NRC (Ref. 8) using

MCNP6 and SCALE further suggest that the critical experiments have higher-than-reported experimental uncertainties. However, the NRC staff notes that there remains a bias between CASMO5 and the experiments. This bias is sufficiently large such that the quantified uncertainties established in SSP-14-P01/028-TR-P-A may no longer be bounding. The benchmarks in SSP-14-P01/028-TR-P-A are more prototypical reactor configurations, and the benchmarks in Supplement 1 are more unconventional in their geometries, which may explain in part the increased discrepancies. While Studsvik's code-to-code comparisons to MCNP6 and the confirmatory calculations provided by ORNL provide assurance that a portion of the computed differences are likely caused by geometry considerations within the benchmark model itself, the NRC staff was unable to determine with reasonable assurance whether the full magnitude of the bias can be attributed to geometry differences alone, and not fuel enrichment.

While the code-to-code comparisons discussed in Section 3.2.1.2 of this SE show good agreement, it does not preclude the possibility of a bias between both codes and "real world" data. To further investigate, the NRC staff reviewed the critical benchmark datasets available in SSP-14-P01/028-TR-P-A and Supplement 1 and the operating reactor dataset used to develop the generic NRFs in SSP-14-P01/028-TR-P-A. Upon visual inspection of the critical benchmark data provided in SSP-14-P01/028-TR-P-A and SSP-14-P01/028-TR-S1, the NRC staff noted that the bias and variance of the critical experiments below 5 wt% and between 5 wt% and 8 wt% are comparable, and it is possible that the differences between the two datasets could be attributed to measurement uncertainty in the benchmark models themselves, and not inherently borne from potential enrichment dependencies in CASMO5.

The NRC staff conducted a Welch's t-test between the critical benchmark dataset from SSP-14-P01/028-TR-P-A and the critical benchmark data below 8 wt% enrichment in Supplement 1 and concluded that the means, or biases, between the two sets are comparable. The NRC staff acknowledges that the critical benchmark dataset from SSP-14-P01/028-TR-P-A is non-normal, but the Welch's t-test remains applicable because the dataset is minimally skewed and the sample size is sufficient. A Welch's t-test between the operating plant dataset used to generate generic NRFs and the critical benchmark data below 8 wt% in Supplement 1 also confirms that the biases between the two sets are comparable. From these results, the NRC staff is confident that the datasets are comparable enough that the generic NRFs remain bounding, and that the increased spread in the critical benchmark data below 8 wt% in Supplement 1 is due to measurement uncertainty in the benchmarks and not from inherent dependencies on enrichment. With the consideration of the good agreement in code-to-code comparisons discussed in Section 3.2.1.2 of this SE, and the lack of apparent limitations in the CASMO5 methodology, the NRC staff concludes that the generic NRFs remain applicable up to 8 wt%.

When analyzing the critical benchmark data for enrichments above 8 wt%, the NRC staff could not draw the same conclusions. A Welch's t-test between the critical benchmarks above 8 wt% and all other datasets, including the critical benchmarks from Supplement 1 with enrichments below 8 wt%, confirms that there is a statistically significant bias in the eigenvalue differences. This bias may invalidate the quantified uncertainties established in SSP-14-P01/028-TR-P-A. between the data for enrichment above 8 wt% and below 8 wt%. The NRC staff recognizes that these results may be due to high experimental uncertainties, but absent further data with which to assess the performance of CASMO5 for applications above 8 wt%, the NRC staff could not conclude with reasonable assurance that the generic NRFs will remain applicable or bounding above 8 wt%.

For these reasons, the NRC staff limits the application of the generic NRFs to 8 wt% U-235 enrichment. This is captured in Limitations and Condition (L&C) #1 in Section 4.0 of this SE. Additionally, the NRC staff noted that the bias discussed above is observed in the standalone executions of CASMO5. Therefore, users exercising standalone executions of CASMO5 between 8 wt% and 10 wt% must note and account for potential biases in the results. This is captured in L&C #2 in Section 4.0 of this SE.

3.2.1.1.1 Discussion on Range of Applicability

SIMULATE5 depends on nuclear data generated by CASMO5. Thus, any adverse biases or trends observed in the performance of CASMO5 could potentially impact the performance of the CMS5 code system overall. In assessing the observed bias in CASMO5 results between 8 wt% and 10 wt% and the requested continued applicability of the generic NRFs, the NRC staff considered the development of a potential limitation and condition and associated penalty factors for the generic NRFs. These factors would help ensure the NRFs remain representative of validation results for enrichments between 8 wt% and 10 wt%.

During the review, the NRC staff informed Studsvik of the potential limitation on the application of the generic NRFs in a closed public meeting that occurred on December 29, 2025 (Ref. 9). Following the meeting, Studsvik requested that the NRC staff limit its review to 8 wt% U-235 enrichment, and to forgo efforts of defining and justifying penalties that would allow application of the generic NRFs between 8 wt% and 10 wt% (Ref. 16). As discussed in Sections 3.2.1.3 and 3.4.1.3 of this SE, the NRC staff concluded that the CMS5 code system is acceptable for licensing applications up to 8 wt%. However, because the NRC staff evaluated the CMS5 code system up to 10 wt% (as documented in this SE), the NRC staff further concluded that an application seeking to use the CMS5 code system between 8 wt% and 10 wt% may do so if adequate justification addressing the impacts on the generic NRFs is provided. This justification will require NRC review and approval on a plant application specific basis. This is also captured in L&C #1 in Section 4.0 of this SE.

3.2.1.2 Code-to-Code Comparisons

A result of using higher enriched fuel in core designs is the increased burnable poison (BP) loading and soluble boron concentration in the coolant required to mitigate excess reactivity at the beginning of cycle (BOL). To determine the enrichment dependency of reactivity worth, Studsvik designed a variety of PWR lattices ranging over the following conditions:

- Lattice Designs: Westinghouse (WH) 14x14, WH 15x15, WH 17x17, Combustion Engineering (CE) 14x14
- Boron concentrations (ppm): 0 - 2500
- Fuel enrichment (wt% U-235): 6.0 - 10.0
- Moderator temperature (K): 300 - 600
- Fuel temperature (K): 293 - 1200
- Removable BP type: PYREX, WABA
- Integral BP Gadolinium (wt% Gd): 2.0 - 16.0
- Integral BP IFBA (ZrB2) (mg/cm B-10): 1.5 - 3.0
- Control rod absorber type: AIC, B_4C, HAF, W

Studsvik used these lattices in code-to-code comparisons to MCNP6 to determine the accuracy of CASMO5 and note any potential biases. Studsvik noted larger differences between the codes for cases with burnable absorbers or fuel temperature variations to assess the Doppler defect, but asserted the errors remain within acceptable ranges. The NRC staff reviewed Studsvik's results and concluded that there are no significant biases in reactivity worth predictions as a function of enrichment for varying boron concentrations, fuel temperatures, moderator temperatures, and absorber types , and as such, the results assure that reactivity worth predictions remain accurate at U-235 enrichments up to 10 wt%.

The supplement does not include discussion on the effects of IE and HBU on the generation of the reflector cross sections. The reflectors are expected to be in close proximity to highly burned fuel. Studsvik stated in a regulatory audit (Ref. 10) and confirmed in a response to a Request for Confirmatory Information (RCI) (Ref. 6), that the reflector cross sections are generated using a neutron source that is representative of driver fuel. In the RCI response, Studsvik also provided a comparison between two different SIMULATE5 models using different reflector cross sections generated by CASMO5. One uses 6.1% enriched fuel as the driver fuel, and the other uses 3.1%. Studsvik found the differences between end of cycle eigenvalue, relative assembly powers, and peak pin powers to be minimal. The NRC staff reviewed these results and while Studsvik has requested an enrichment extension to 8 wt% and the comparison only uses fuel up to 6.1 wt%, the NRC staff recognizes that it is improbable and impractical that fuel with an enrichment of 6.1 wt% or higher be placed in proximity to the reflectors. In the engineering judgement of the NRC staff, Studsvik's analysis results demonstrates that the reflector cross sections are minimally affected by higher enrichments and burnups, and the risk associated with placing fuel up to 10 wt% enrichment is minimal. Therefore, the NRC staff concluded that the reflector cross sections are minimally affected fuel enrichments up to 10 wt% and rod-average burnups up to 80 GWd/MTU, and that no further verification or validation of the reflector cross sections is necessary.

3.2.1.3 Conclusions

Given the validation discussions above, which assess the requested range of applicability, the NRC staff concluded that CASMO5 performs adequately at IEs in the ranges considered, with the limitations discussed above. Therefore, the NRC staff concludes that it is acceptable to extend the range of applicability of CASMO5 to include U-235 enrichments up to 8 wt% for use consistent with SSP-14-P01/028-TR-P-A and the range of applicability specified in this SE. For use of CASMO5 at enrichments between 8 wt% and 10 wt%, restrictions or justifications must be supplied consistent with L&C #1 in Section 4.0 of this SE.

3.2.2 Application to Higher Burnup

Similar to its application to IE, Studsvik states in Section 2.1 of Supplement 1 that no methodological changes to CASMO5 are necessary to support its application to higher burnups. As technical justification for the burnup range extension, Studsvik provided comparisons to relevant critical benchmarks and higher order methods.

3.2.2.1 Benchmarks

Benchmarking against data is incredibly important for higher burnup uses because, as noted in Section 3.1 of this SE, when benchmarking depletion algorithms, code-to-code comparisons with a Monte Carlo transport code are not considered comparisons to a higher order method. The depletion algorithms for Monte Carlo transport codes such as SERPENT2, and lattice physics codes such as CASMO5, are typically based on similar numerical methods. Additionally, both codes use the same nuclear data libraries, and a comparison between the codes will not account for any potential errors in these libraries. Supplement 1 does not contain any analyses involving benchmarks of CASMO5 to higher burnup data. The NRC staff discussed this topic with Studsvik in a regulatory audit (Ref. 10), and Studsvik referenced a conference paper (Ref. 11) detailing a third-party analysis of higher burnup fuel using CASMO5.

The third-party analysis compares reactivity worths calculated by CASMO5 to measured reactivity worths of highly burned PWR fuel (up to 120 GWd/MTU) in the Light Water Reactor (LWR)-PROTEUS Phase II program. For fuel assemblies containing UO_2 fuel in H_2O moderator, there is a slight bias between the calculated and measured reactivity worths. CASMO5 slightly underpredicts the reactivity worths. The underprediction is less prominent for cases with boron in the moderator. The NRC staff reviewed the paper and concludes that the small discrepancies in the results suggest that CASMO5 performs adequately at very high burnups. However, given the limited discussion within the paper, the NRC staff could not verify the rigor with which the code was executed. Therefore, the NRC staff used the document to inform its overall assessment of CASMO5 performance rather than utilize it as a significant basis for approval. The primary basis for approval of CASMO5 for maximum rod-average burnups is the results of the code-to-code comparisons discussed in the next section of this SE. The results of this paper provide assurance that the nuclear data libraries and numerical methods implemented in CASMO5 are performing adequately.

3.2.2.2 Code-to-Code Comparisons

Code-to-code comparisons with SERPENT2 were used to assess the accuracy of CASMO5 reactivity predictions as a function of increasing exposure. As noted above, SERPENT2 is not a higher order method in comparison to CASMO5 regarding depletion calculations. However, the comparisons are still useful in helping assess the adequacy of CASMO5 by detecting trends or potential biases, as the comparison provides an independent verification that the depletion algorithm is performing accurately.

Studsvik used a single pin cell and a 17x17 generic lattice containing IFBA fuel rods and gadolinium fuel rods for the depletion comparisons. The pin cell was depleted at the conditions below:

- Fuel enrichments: 5.0, 8.0, 10.0 wt.% U-235
- Soluble boron concentration: 0.0 ppm
- Moderator temperature: 600.0 K
- Cladding temperature: 600.0 K
- Fuel temperature: 900.0 K

The pin cell depletion results, given as a difference between CASMO5 and SERPENT2, indicate increasing error as a function of enrichment (further discussed below). As a function of burnup, all enrichment cases stabilize around 200 pcm difference at 80 GWd/MTU. The NRC staff notes that the increased error at higher burnups is likely to be nonphysical and an artifact of the code burning the pin at a constant specific power beyond the point at which the pin becomes subcritical, and the fuel has been depleted. It is physically impossible for the pin to sustain the hard coded specific power to 80 GWd/MTU, resulting in increased errors that are not indicative of degrading code performance.

The lattices were depleted at the following conditions:

- Fuel enrichments: 5.0, 6.30, 10.0 wt.% U-235

- Gadolinium enrichments: 6.0, 8.0, 12.0 wt.% Gd_2O_3
- Soluble boron concentration: 800.0 ppm
- Moderator temperature: 600.0 K
- Cladding temperature: 600.0 K
- Fuel temperature: 900.0 K

The lattice depletion eigenvalue results share a similar trend as the pin cell results, where higher enrichments have a higher error bias. After gadolinium burnout, errors decrease and the trend remains stable. Fission rate comparisons between CASMO5 and SERPENT2 captured at 0, 22, 60, and 90 GWd/MTU for 6.3 wt% U-235 indicate a growing discrepancy as a function of burnup. However, the trend is minimal, and the maximum error present in the requested range of applicability remains consistent with the error observed in the currently approved range of applicability. CASMO5 generally predicts the peak pin in the same location as SERPENT2, except for the 90 GWd/MTU case, where it predicts a nearby pin instead.

The nuclear data generated by CASMO5 that is provided to downstream codes and methods is heavily dependent upon the accurate prediction of isotopic concentrations. It is therefore important to ensure accurate tracking of the production and removal of major fission-related isotopes. The trends in these isotopes can also provide valuable insight into the how well the underlying models used to generate them perform in the intended application space. Therefore, the NRC staff asked Studsvik in a RAI (Ref. 5) to provide isotopic concentration comparisons between CASMO5 and SERPENT2 as a function of burnup for major actinides and poisons. Studsvik responded (Ref. 6) with comparisons of fuel-assembly average number densities for a PWR 17x17 fuel-assembly lattice with enrichment of10 wt% U-235 and 12 wt% Gd_2O_3 up to 90 GWd/MTU. The NRC staff reviewed the results and concluded that the behavior of CASMO5 is expected and consistent with SERPENT2. There are no major trends that indicate degrading accuracy. Based on these results, the NRC staff concludes that the burnup comparisons are acceptable because the results indicate that CASMO5 has comparable predictive capability to SERPENT2 at high burnups.

3.2.2.3 Conclusions

The NRC staff reviewed the code-to-code comparisons provided by Studsvik and notes that the results provide assurance that the CASMO5 depletion algorithm is implemented properly and is functioning without significant trends or biases. However, as discussed in Section 3.2.2.1 of this SE, code-to-code comparisons cannot account for potential errors in the nuclear data libraries or numerical methods themselves. While the NRC staff cannot verify the rigor or validity of the third-party benchmark analysis provided by Studsvik and discussed in Section 3.2.2.1 of this SE, the analysis results provide assurance that the depletion algorithms and nuclear data libraries are performing adequately. Additionally, as discussed in Section 2.0 of this SE, analytical methods such as CMS5 undergo consistent validation when applied to plant operations. Given these considerations and the validation detailed above, which covers the requested burnup range of applicability, the NRC staff concluded that CASMO5 performs adequately at higher burnups, and therefore concludes that it is acceptable to extend the range of applicability of CASMO5 to a maximum rod-average burnup of 80 GWd/MTU.

3.3 CMSLINK5

CMSLINK5 is a linking code that processes and functionalizes data from CASMO5 for use by SIMULATE5. Data passed from CASMO5 to SIMULATE5 via CMSLINK5 include cross sections, detector data, pin power reconstruction data, kinetics data, isotopics data, and spontaneous fission data. The methodology is independent of the requested extension to the range of applicability of CMS5. Any potential discrepancies in accuracy will be captured by code-to-code comparisons between CASMO5 and SIMULATE5, which as discussed in Section 3.4.1.1 of this SE, demonstrate acceptable performance. Therefore, the NRC staff concludes that a review of CMSLINK5 regarding the extended range of applicability requested in Supplement 1 is not necessary and that CMSLINK5 is applicable to enrichments up to 10 wt% U-235, maximum rod-average burnups of 80 GWd/MTU, and light water pressurized SMRs.

3.4 SIMULATE5

SIMULATE5 is a multigroup analytical nodal code for PWRs. It solves the multigroup diffusion or the simplified P_3 transport equations. SIMULATE5 receives cross-section data from CASMO5 through CMSLINK5. SIMULATE5 also includes a thermal hydraulic module that computes fuel temperatures.

3.4.1 Application to Increased Enrichment

Studsvik claims in Section 2.1 of Supplement 1 that no methodological changes to SIMULATE5 are necessary to support its application to higher enrichments. As technical justification for the enrichment range extension, Studsvik provided comparisons to relevant critical benchmarks and higher order methods. While SIMULATE5 receives enrichment-dependent neutronic data from CASMO5 via CMSLINK5, it contains separate neutronic models that necessitate separate validation.

3.4.1.1 Benchmarks

Studsvik used the Otto-Hahn critical experiment to numerically benchmark SIMULATE5 for application to higher enrichment. Otto-Hahn is a small, high leakage core with 3.5 wt% and 6.6 wt% enriched uranium. Two-dimensional and three-dimensional configurations were analyzed. The comparisons are not made to experimental data but to a 2D CASMO5 model and to 2D and 3D MCNP6 models of the experiment. CASMO5 is considered a higher order method in comparison to SIMULATE5 because SIMULATE5 is based on diffusion theory while CASMO5 is based on transport theory.

The 2D and 3D eigenvalue results show good agreement with MCNP6 and the experiment. Some fission rate comparisons show high differences, but, in the NRC staff's technical judgement, the overall root mean square error is reasonable. Most importantly, the peak pin fission rate differences are minimal. The results provide assurance that SIMULATE5 can adequately model high leakage cores with steep power gradients.

3.4.1.2 Code-to-Code Comparisons

Studsvik did not use code-to-code comparisons to validate SIMULATE5 for higher enrichments.

3.4.1.3 Conclusions

While the validation discussed above does not fully cover the requested range of applicability, the NRC staff notes that most of the SIMULATE5 enrichment dependence lies in the cross-section data provided by CASMO5. Because of this, confirmation that CASMO5 performs adequately at higher enrichments is sufficient for assessing the performance of SIMULATE5 beyond the ranges provided in the validation discussed above. Therefore, based on the conclusions discussed in Section 3.2.1.3 of this SE, the NRC staff is assured that the SIMULATE5 neutronic models perform adequately at higher enrichments. As discussed in Section 3.2.1.1 of this SE, the requested range of approval for CMS5 was adjusted to 8 wt% U-235 fuel enrichment. The NRC staff has assessed the performance of SIMULATE5 up to 8 wt% and concluded SIMULATE5 is acceptable for use for U-235 enrichments up to 8 wt% for the reasons discussed above. Application of SIMULATE5 between 8 wt% and 10 wt% must include justification for doing so.

3.4.2 Application to Higher Burnup

Studsvik states that no methodological changes to SIMULATE5 are necessary to support its application to higher burnups up to a maximum rod-average burnup of 80 GWd/MTU. As technical justification for the burnup range extension, Studsvik provided comparisons to relevant critical benchmarks and higher order methods. While SIMULATE5 receives burnup-dependent neutronic data from CASMO5 via CMSLINK5, it contains separate neutronic and fuel performance models that necessitate separate validation. Thermal hydraulics are largely independent of burnup, and these models do not require validation. As such, the NRC staff focused its review on the validation of the neutronic and fuel performance models.

3.4.2.1 Benchmarks

To validate SIMULATE5 at higher burnups, Studsvik referenced an analysis performed by Dominion Energy (Ref. 12) with data from a lead test assembly (LTA) from North Anna Power Station. The LTA was burned to a maximum rod-average burnup of 72.6 GWd/MTU over four cycles. The axial burnup distribution was measured through radioassay and calculated by CMS5 at corresponding axial positions. Several cases were analyzed, using different standard CMS5 core design processes, including different axial nodalization and submesh options due to the uncertainty associated with the nonuniform thermal expansion of the LTA. Studsvik noted that the analysis results show that CMS5 adequately predicts burnup for all rods except one, which was located next to guide thimbles and is less neutronically coupled and more difficult to predict than the other three rods. The discrepancy is an overprediction, which is conservative. Therefore, the NRC staff concluded that the results of the analysis are acceptable. Additionally, while the whole CMS5 is executed for this comparison, this benchmark validates models and features of SIMULATE5 that are not dependent on CASMO5. Therefore, the NRC staff concludes that the results are appropriate for assessing the performance of SIMULATE5 at higher burnups.

The thermomechanical properties of the fuel change as fuel burnup increases. These properties affect the fuel thermal conductivity, which is used to calculate fuel temperatures. Accurate fuel temperature predictions are important for determining reactivity feedback due to the Doppler effect. Supplement 1 does not contain fuel temperature calculation validation at higher burnups. The validation in the original TR, the Halden benchmark, uses fuel rods up to 50 GWd/MTU, which is insufficient for covering the requested range of applicability. Studsvik stated in a regulatory audit (Ref. 10) that data for fuel at higher burnups is unavailable, but confirmed in an RCI (Ref. 6) that the fuel thermal conductivity degradation model used in SIMULATE5 uses the same model implemented in FAST (Fuel Analysis under Steady State and Transients), specifically the version that is based on the correlation proposed by Nuclear Fuel Industries and is modified to alter the temperature-dependent portion of the burnup and include a dependency on gadolinia content (Ref. 13). While FAST is not an NRC approved code, this model has been validated up to and beyond Studsvik’s requested range of applicability (Ref. 14).

The NRC staff notes that the way a model is implemented can affect the output of the model and, therefore, the output of the code. Consequently, an assertion that the models are identical does not necessarily ensure that the codes will output identical, or adequately similar, results. While there are several minor factors, the most significant contributor to potential differences are the sources for the inputs to the model. Because of this, the NRC staff reviewed the inputs to the FAST model and the sources of those inputs, and notes that it is a data-driven correlation (i.e., empirical) rather than a first principles, physics-based model, and it must be used within its range of applicability. It is noted that the range of applicability for this model is 0 to 90 GWd/MTU for UO_2, and 0 to 50 GWd/MTU for UO_2 fuel with Gd_2O_3. The model inherently accounts for any differences between fuel that contained Gd_2O_3 before it was burned and fuel that did not. If gadolinium is still present in a rod beyond 50 GWd/MTU, the model is no longer valid. In its RAI response (Ref. 6), Studsvik provided plots of gadolinium concentration as a function of burnup for a generic 17x17 PWR lattice with 10 wt% U-235 fuel and 12 wt% gadolinium burnable absorber. The gadolinium concentration is near zero and negligible at 50 GWd/MTU, and as such, the NRC staff expects that all gadolinium initially present in a rod will be depleted by the time the rod reaches 50 GWd/MTU. In the engineering judgement of the NRC staff, it is highly unlikely that a core design may be implemented where a significant concentration of gadolinium exists within a rod that is at or above burnups of 50 GWd/MTU if the initial concentration of gadolinium is 12 wt% or lower. For this reason, the NRC staff retains the original maximum gadolinium concentration limit of 12 wt% from SSP-14-P01/028-TR-P-A. This will assure that negligible concentrations of gadolinium are present in the fuel beyond 50 GWd/MTU, and that the SIMULATE5 thermal conductivity degradation model will be used within its range of applicability.

3.4.2.2 Code-to-Code Comparisons

Studsvik did not use code-to-code comparisons to validate SIMULATE5 for higher burnups.

3.4.2.3 Conclusions

While the validation discussed above does not fully cover the requested range of applicability, the LTA data up to 72 GWd/MTU and the validation of the thermal conductivity degradation model assure that the SIMULATE5 neutronic models are performing adequately at higher burnups. These models have the highest likelihood of exhibiting degrading accuracy at high burnups, and there were no specific issues identified by the NRC staff that would result in degraded accuracy between 72 GWd/MTU and 80 GWd/MTU. Additionally, as discussed in Section 3.3 of this SE, the results discussed in Sections 3.4.1.1 of this SE have shown that the CASMO5 neutronic inputs are being adequately processed by CMSLINK5 and properly implemented in SIMULATE5, as expected. For these reasons, the NRC staff concludes SIMULATE5 is acceptable for use up to a maximum rod-average burnup of 80 GWd/MTU.

3.5 Application to Light Water Small Modular Reactors

In Section 4.1 of Supplement 1 (Ref. 2), Studsvik states that the CMS5 methodology can be generically referenced in licensing applications for SMR concepts with designs meeting the methodology application range limits. If approved for generic applicability to light-water, pressurized SMRs, the CMS5 methods, verification and validation basis, and NUF and NRF methodology will be applicable to light-water, pressurized SMRs that meet the limitations and conditions in Section 4.0 of this SE.

Studsvik states in Section 4.2 of the supplement that CMS5 has been approved for reference in the design certification of an SMR by the NRC (Ref. 15), and the LCT-81 CMS5 benchmarks discussed in Section 3.4.1.1 of this SE demonstrate adequacy of the CMS5 code system in capturing the nuclear performance of varied PWR configurations. The NRC staff notes that, at the time of this writing, a limited scope of light-water, pressurized SMRs has been reviewed and approved by the NRC. Future light-water SMR designs may differ greatly from these approved designs such that the application of CMS5 to the broader category of light-water SMRs may not be valid, and the regulatory findings associated with CMS5 in reference (Ref. 15) may not apply. As such, the NRC staff is not using the referenced TR (Ref. 15) as a basis for generic approval for application of CMS5 to light water pressurized SMRs, and the L&Cs and its associated SE (Ref. 15) do not apply to this SE.

The NRC staff notes that the most significant neutronic difference between a PWR and a light water pressurized SMR that meets the range of applicability as defined by L&C #1 of this SE is core size, and subsequently, light water pressurized SMRs typically have higher axial leakage and larger flux gradients than PWRs. The NRC staff notes that as demonstrated in Section 3.4.1.1 of this SE, CMS5 is capable of accurately modeling small cores with steep local fluxes and power gradients, which are expected characteristics of light-water, pressurized SMRs. The CMS5 validation database includes light water pressurized SMRs without comparatively high axial leakage or large flux gradients, as these are characteristics of operating PWRs with modern core designs. Other significant neutronic differences that may arise between PWRs and light water pressurized SMRs are the materials within the core and fuel and the geometry of the fuel. These conditions are maintained within the CMS5 validation database by L&C #1 of this SE. Thus, there is assurance that future light-water, pressurized SMR designs will be accurately modeled by CMS5 if they meet the range of applicability as defined by L&C #1 of this SE. Therefore, the NRC staff concludes that the CMS5 methodology detailed in the TR (Ref. 3) is acceptable for generic application to light-water, pressurized SMRs, if the licensing applications referencing the CMS5 methodology meet the L&Cs in both SSP-14-P01/028-TR-P-A and this SE, except for those in SSP-14-P01/028-TR-P-A that are superseded by this SE. Light-water, pressurized SMR licensing applications must be within the application range limits of the methodology, as defined by L&C #1 of this SE. As required by L&C #3, a light-water, pressurized SMR licensee using the CMS5 methodology must determine the NUFs or NRFs specific to its designs or justify the use of the approved generic NRFs in SSP-14-P01/028-TR-P-A in licensing applications.

3.6 Range of Applicability

The approved range of applicability of the CMS5 code system methodology is restricted to the range of applicability detailed in L&C #1 of this SE based on the technical discussion in sections 3.2 through 3.5 of this SE. The specified range of applicability applies to the CMS5 code system, unless stated otherwise. To extend the range of applicability of the CMS5 code system to fuel designs not meeting L&C #1 (e.g., larger array sizes, non-square lattice arrangement geometries, new non-zircaloy assembly/fuel rod materials, non-UO2 fuel matrix, etc.), additional analysis and benchmarking for the new fuel or reactor design and NRC review and approval would be required.

4.0 LIMITATIONS AND CONDITIONS

Based on the technical evaluation above, the NRC staff concludes that the CMS5 code system methodology for LWRs is acceptable for reference by licensing applications, subject to the following L&Cs. L&Cs marked by an asterisk (*) are either newly introduced by this SE or supersede L&Cs from SSP-14-P01/028-TR-P-A for the purpose of Supplement 1 to SSP-14-P01/028-TR-P-A. L&Cs without an asterisk are reproduced from SSP-14-P01/028-TR-P-A for convenience, and the NRC staff have determined that they remain applicable. For these L&Cs the basis for the L&C is discussed in the NRC staff's original SE in SSP-14-P01/028-TR-P-A.

1. The range of applicability of the Studsvik Scandpower Core Management System for PWRs and light water pressurized SMRs shall be defined by the following materials or conditions of fuel, cladding, poison, and other core materials, as specified below:

    a. Fuel and Poison

        i. *The CMS5 code system has been assessed to 10 wt% U-235 and is approved for enrichments up to 8 wt% U-235. Application of the CMS5 code system for enrichments between 8 wt% and 10 wt% requires:

            1. Justification for the continued applicability of the generic NRFs, or

            2. Proposition of and justification for an acceptable penalty to the generic NRFs

            The basis for this L&C is discussed in Section 3.2.1.1.1 of this SE.

        ii. Pin lattice geometries ranging from 14x14 to 17x17, including both large and small water hole designs

        iii. The typical range for nominal density between 10.3 and 10.9 g/cc (or between 0.94 and 0.985 as a fraction of theoretical density)

        iv. Integral burnable absorber Gd_2O_3 with gadolinium concentration up to 12 percent as a mass fraction of the total fuel weight

        v. Integral burnable absorber (IFBA) ZrB_2

        vi. Discrete absorber types WABA, B_4C-AlO_3, boron silicate glass, and hafnium suppressor rods

    b. Cladding

        i. Zircaloy-2, Zircaloy-4
        ii. ZIRLO and Optimized ZIRLO
        iii. M5
        iv. TVEL cladding materials (E110 and E635)
        v. Japanese cladding material (NDA and MDA)
        vi. Zr based cladding materials with trace elements of other materials (such as Nb, Sn, Fe, Cr, Ni, and O)

c. Structural Materials (including, but are not limited to)[1]

i. Stainless Steel, Inconel-718, Inconel-750

d. Fuel Burnup

i. *Maximum rod-average burnup of 80 GWd/MTU

2. *Standalone executions of CASMO5 (e.g., lattice depletions for downstream use in other analyses, spent fuel pool criticality analyses, etc.) between 8 wt% and 10 wt% U-235 fuel enrichment must include a demonstration or justification that no significant biases exist. The basis for this L&C is discussed in Section 3.2.1.1 of this SE.

3. *The CMS5 code system is applicable to small modular, light-water, pressurized reactors, provided that the designs fall under the range of applicability of CMS5 defined by L&C #1 of this SE and that prospective applicants determine the NUFs or NRFs specific to their designs or justify the use of the approved generic NRFs in SSP-14-P01/028-TR-P-A. The basis for this L&C is discussed in Section 3.5 of this SE.

4. Any change management with respect to addition of new features, new functionality, correction of software errors, and/or usage of additional data from operating reactors, test reactors, or experiments must ensure that the NRFs generated by exercising the methodology reported in the CMS5 TR remain conservative.

   a. * New features and new functionalities shall not include new models, new correlations, and new numerical methods of solution. For implementation of new models, new correlations, and changes in the solution methods into the approved version of the TR shall be subject to NRC staff approval.

5. The resonance upscatter model (RUP) described in the CASMO5 methodology manual and further described in response to an RAI has not been implemented in the CMS5 application. The NRC staff has reviewed the RUP model and determined that Studsvik may use the RUP factors and apply to the resonance integral (RI) in CASMO5.

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[1] The NRC staff notes this L&C is inherited from SSP-14-P01/028-TR-P-A, the basis for which originates from Studsvik's response to RAI 1.b in SSP-14-P01/028-TR-P-A. In its response, Studsvik states that CASMO5 is capable of modeling any cladding material. CASMO5 has a library of default compositions for common cladding and structural materials, but the user may input specific compositions that are provided by a fuel vendor. While the CMS5 validation database contains many common compositions and multiple vendor-specific compositions, it is possible that a user may specify a composition outside of the CMS5 validation database. Small differences in cladding and material composition generally do not make a significant neutronic difference because the components (i.e. Fe, Cr, Zr, etc.) are relatively neutronically transparent. A significant effect in neutronic calculations would only occur if the composition was changed drastically, such that it would no longer be the same material. These materials generally have thermomechanical and chemical effects on the system, which are out of scope of this review. For the purposes of Supplement 1 to SSP-14-P01/028-TR-P-A, the NRC staff interprets this L&C to mean that the CMS5 code system is approved for modeling cladding and structural materials that are within the validation database, or compositionally similar such that there is no significant neutronic difference.

5.0 CONCLUSIONS

The technical review of benchmarking results of the CMS5 code system for application to IE, higher burnup, and light-water, pressurized SMRs has been completed. The documentation provided in the supplement and the responses to the RAIs demonstrate that the methodologies are correctly implemented and appropriate to the requested application range. The validation presented demonstrates that CMS5 is an acceptable tool for a licensee to use to determine that its core design meets the requirements of the GDCs discussed in Section 2.0 of this SE. Thus, based on the technical evaluation above, the NRC staff finds the CMS5 code system methodology for LWR core design acceptable for licensing applications subject to the L&Cs specified in Section 4.0 this SE.

6.0 REFERENCES

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