Studsvik

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Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors: Supplement for Extended Enrichment, Burnup, and SMRs

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Abstract

Studsvik Scandpower is extending the range of applicability of the CMS5 Generic PWR Topical Report SSP-14-P-01/028-TR-P-A with the technical bases presented in this Topical Report Supplement 1 for higher enrichment, higher burnup, and noted applicability to light water pressurized Small Modular Reactors (SMRs). The CMS5 methodology, without modifications or adjustments, is thoroughly exercised and compared to multiple relevant higher-order method analyses, select criticality benchmark experiments, and meaningful operational data. The technical bases and comparison results presented herein justify the following ranges and applicability statements regarding use of the CMS5 code system to: PWR fuel with U-235 enrichment in UO_2 up to and including 10 wt%; PWR fuel designs with maximum rod-average fuel burnup up to and including 80 GWd/MTU; and light water pressurized Small Modular Reactors. Extension of the CMS5 methodology range of applicability in this manner would contribute to improved power generation flexibility (e.g. longer cycle length) while retaining efficient regulatory oversight as plants safely extend operational domains.

Revision History

Supplement Revision Number	Description	Date
0	Initial Issuance.	January 2025

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1. Introduction

1.1 Background

Studsvik Scandpower has attained acceptance from the United States Nuclear Regulatory Commission (NRC) staff for generic reference in licensing applications of the CMS5 Core Management System for Pressurized Water Reactors [1.1], so applied within the limitations and conditions delineated in the associated topical report and final safety evaluation [1.2]. For clarity and brevity, the Reference [1.1] Topical Report and the associated Reference [1.2] Final Safety Evaluation are hereafter in combination referred to as the CMS5 Generic PWR TR.

Application of the CMS5 Generic PWR TR is implemented, for example, by a US utility in each of its current in-service PWRs (see [1.3]). Other CMS5 customers could similarly apply GL 83-11 [1.4] to fully utilize the methods and/or conservative generic nuclear reliability factors of the CMS5 Generic PWR TR, or possibly utilize the topical report and references therein for non-generic, distinct licensing submittals (e.g. spent fuel pool criticality analysis) for NRC review. Furthermore, NRC has provided clarification [1.5] regarding changes to the underlying CMS5 methodology such that implementation of new models, new correlations, and changes in the solution methods of the CMS5 Generic PWR TR shall be subject to NRC review and approval. Otherwise, an applicable software change management process must ensure the generic nuclear reliability factors remain conservative for the addition of new features, new functionality, correction of software errors, and/or usage of additional operating reactor data.

1.2 Purpose

This topical report aims to provide sufficient technical bases and background information to supplement the CMS5 Generic PWR TR SSP-14-P01/028-TR-P-A [1.1] for an extended application range without changes to the underlying CMS5 methodology. Namely, this TR Supplement 1 for CMS5 Generic PWR application intends to achieve these items:

- Extend the application range for U-235 enrichment up to and including 10 wt%
- Extend the application range for maximum rod-average burnup up to and including 80 GWd/MTU
- Provide clarification on applicability to light water pressurized Small Modular Reactors (SMRs) that demonstrate compliance with code system range and validity requirements

The information presented in this TR Supplement 1 will demonstrate how and why the CMS5 methodology is acceptable for performing the same steady-state core physics analyses covered in the CMS5 Generic PWR TR [1.1] when exercised in the extended application range and reactor type listed above. Note that the CMS5 methodology is comprised of the CASMO5 lattice physics and the SIMULATE5 nodal physics codes as described in [1.1], each code referred to independently as appropriate to the purpose of the TR Supplement 1.

1.3 Organization

This TR Supplement 1 is arranged in a manner addressing each extended application and clarification items listed in Section 1.2. Specifically, Section 2 provides information justifying the application of the CMS5 methodology for fuel enrichments greater than the 5 wt% application limit of the CMS5 Generic PWR TR [1.1] up to and including 10 wt% enrichment for U-235, by way of comparison to higher-order reference solutions, evaluation of relevant critical experiment benchmarks with the CASMO5 lattice code, and benchmark of the SIMULATE5 nodal code with a reactor experiment. Section 3 presents the basis for application of the existing CMS5 methodology from a maximum rod-average burnup of 62 GWd/MTU to an extended value up to and including a maximum rod-average burnup of 80 GWd/MTU, including comparisons of pin-cell and lattice depletion calculations to higher-order solutions, along with examination of comparative results to data for an operating reactor's extended burnup lead test assembly. Finally, Section 4 offers an explanation clarifying direct applicability of the CMS5 methodology to light water SMRs which comply with the approved CMS5 methodology application range and limitations.

Each section or sub-section of this TR Supplement 1 has a self-contained references section.

1.4 Adoption and Use of this TR Supplement 1

Upon review and approval as determined by a final safety evaluation from NRC, this TR Supplement 1 could be referenced in licensing submittals to NRC by new or existing CMS5 customers having commercially procured access to the CMS5 Generic PWR TR. Reference by licensees to this TR Supplement 1 would only occur in conjunction with reference to the CMS5 Generic PWR TR, i.e. stand-alone usage of TR Supplement 1 would not be expected nor meaningful for licensees seeking to exercise the CMS5 Generic PWR TR. Consistent with the process described in [1.1], the primary adoption mechanism by a given licensee remains the GL 83-11 [1.4] process implementing the CMS5 Generic PWR TR along with this TR Supplement 1 for use in a specified licensing application, including notification to NRC of the nuclear reliability factors for use in said licensing application. Additionally, a licensee may reference the CMS5 Generic PWR TR and TR Supplement 1 in separate and distinct licensing actions which the licensee may have submitted to NRC for review and approval. In this fashion, review could be expedited or enhanced for the relevant and applicable CMS5 methodology elements invoked in such licensing actions.

1.5 Section 1 References

1.1. SSP-14/P01-028-TR-P-A, Revision 0, "Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors," September 2017. **NRC Accession Number: ML17279A986** (non-proprietary version).

1.2. Letter, D. Morey (NRC) to B. Haugh (Studsvik Scandpower), FINAL SAFETY EVALUATION FOR STUDSVIK SCANDPOWER INC. TOPICAL REPORT SSP-14-P01/028-TR, "GENERIC APPLICATION OF THE STUDSVIK SCANDPOWER CORE MANAGEMENT SYSTEM TO PRESSURIZED WATER REACTORS," September 15, 2017. **NRC Accession Number: ML17236A419** (non-proprietary version).

1.3. a. Letter, M.D. Sartain (Dominion Energy) to NRC Document Control Desk, VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION ENERGY VIRGINIA) NORTH ANNA POWER STATION UNITS 1 AND 2 NOTIFICATION: IMPLEMENTATION OF GENERIC LETTER 83-11, SUPPLEMENT 1 GUIDELINES TO ALLOW USE OF TOPICAL REPORT SSP-14-P01/028-TR-P-A METHODOLOGY, November 15, 2017. **NRC Accession Number: ML17325A991**.

b. Letter, G. Bischof (Dominion Energy) to NRC Document Control Desk, VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION ENERGY VIRGINIA) SURRY POWER STATION UNITS 1 AND 2 NOTIFICATION: IMPLEMENTATION OF GENERIC LETTER 83-11, SUPPLEMENT 1 GUIDELINES TO ALLOW USE OF TOPICAL REPORT SSP-14-P01/028-TR-P-A METHODOLOGY, December 18, 2017. **NRC Accession Number: ML18002A359**.

c. Letter, M.D. Sartain (Dominion Energy) to NRC Document Control Desk, DOMINION ENERGY NUCLEAR CONNECTICUT, INC., MILLSTONE POWER STATION UNIT 3 NOTIFICATION: IMPLEMENTATION OF GENERIC LETTER 83-11, SUPPLEMENT 1 GUIDELINES TO ALLOW USE OF TOPICAL REPORT SSP-14-P01/028-TR-P-A METHODOLOGY, January 29, 2019. **NRC Accession Number: ML19042A078**.

d. Letter, M.D. Sartain (Dominion Energy) to NRC Document Control Desk, DOMINION ENERGY NUCLEAR CONNECTICUT, INC., MILLSTONE POWER STATION UNIT 2 NOTIFICATION: IMPLEMENTATION OF GENERIC LETTER 83-11, SUPPLEMENT 1 GUIDELINES TO ALLOW USE OF TOPICAL REPORT SSP-14-P01/028-TR-P-A METHODOLOGY, July 9, 2018. **NRC Accession Number: ML18194A581**.

1.4. United States Nuclear Regulatory Commission, Generic Letter 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses", June 24, 1999.

1.5. Letter, D. Morey (NRC) to A. Wharton (Studsvik Scandpower), CLARIFICATION OF SAFETY EVALUATION FOR STUDSVIK SCANDPOWER INC., TOPICAL REPORT SSP-14-P01/028-TR, "GENERIC APPLICATION OF THE STUDSVIK SCANDPOWER CORE MANAGEMENT SYSTEM TO PRESSURIZED WATER REACTORS," August 31, 2018. **NRC Accession Number: ML18227A813**.

2. CMS5 Application for U-235 Enrichment to 10 wt.%

2.1 Motivation

Recent consideration by the US nuclear generation industry of increasing U-235 enrichment is motivated by, among other things, the desire to lengthen fuel cycle operation in PWRs from 18 to 24 months. Fuel vendors are responding by offering fuel design options that include enrichments greater than 5 wt%. Extension of the approved CMS5 methodology in response to this evolution positions CMS5 to remain a top-tier tool for fuel-vendor independent core design and analysis.

There are no methodological changes needed in CMS5 to support the extension of the U-235 enrichment range to 10 wt%. Suitability must be demonstrated, however, by executing comparisons for relevant critical experiments and higher-order reference solutions.

2.2 Comparisons to Higher-Order Reference Solutions

Higher-order Monte Carlo codes provide reference solutions for comparison to CMS5 results at representative PWR assembly geometries and conditions. Preparation of the CMS5 Generic PWR TR included over 500 lattice comparisons between CASMO5 and MCNP6 [2.1]. Extension of the MCNP6 runs to higher enrichments allows for the assessment of CMS5 performance at higher enrichments [2.2].

Four assembly types were chosen to cover the general commercial examples of PWR fuel: Westinghouse (WH) 14x14 fuel assemblies, WH 15x15 fuel assemblies, WH 17x17 fuel assemblies and Combustion Engineering (CE) 14x14 fuel assemblies. The PWR lattice and operating conditions modeled in this analysis are summarized in the list below:

- Lattice Designs: WH 14x14, WH 15x15, WH 17x17, 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 (ZrB_2) (mg/cm B10): 1.5 - 3.0
- Control rod absorber type: AIC, B_4C, HAF, W

This analysis is performed to complement comparisons made between CASMO5 and measured data. Given that experimental data is not available over the range of conditions listed above, the use of Monte Carlo with continuous-energy cross sections (MCNP6 [2.3]) is a generally accepted next best reference.

Figure 2–1 shows the enrichment dependency of the reactivity worth relative differences between CASMO5 and MCNP6, where the boron reactivity worth is defined as $\rho = \left(\frac{k-k_0}{k \cdot k_0}\right)/(x - x_0)$, k is the eigenvalue, and x is the boron concentration. The case included here is enrichment/boron simulation data for the hot full power (HFP) operating condition, as well as the results for enrichments less than 5% following [2.1]. The reactivity worth differences are also shown for the cases with < 5% enrichment when re-run with a larger

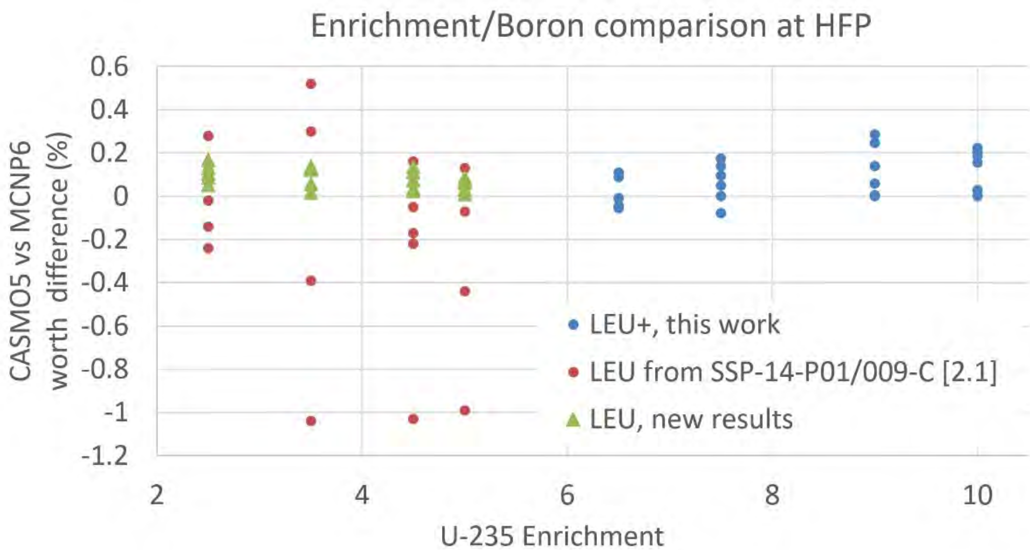


Figure 2–1. WH17x17 enrichment dependence of boron reactivity worth differences at HFP.

number of histories used in this work (marked "LEU, new results"). This allows a more direct comparison between the two enrichment ranges; indeed, the data spread is much narrower than the original results from [2.1] and is similar to the spread observed for the higher enrichments studied in this work.

Code-to-code comparison with consistent nuclear data is performed to provide insight into any significant modeling difference between CASMO5 and the higher order code, MCNP6. The resulting comparisons, detailed in [2.2], demonstrate excellent agreement (< 2% difference) between CASMO5 and MCNP6 for most parameters, with some larger outliers for certain control rods (3-4%) and the Doppler defect (up to 5.5%).

Table 2–1 summarizes the reactivity benchmark portion of the analysis [2.2] performed at discrete U-235 enrichments of 6.5 wt%, 7.5 wt%, 9.0 wt%, and 10.0 wt%. The table presents relative differences, where the reactivity worths are computed as $\rho = \left(\frac{k-k_0}{k \cdot k_0}\right)$ (except for the boron worths which use the previously defined equation). All mean CASMO5 calculated soluble boron worths, integral absorber worths, discrete absorber worths, IFBA worths and Gd worths were within 3.5% of the calculated values from MCNP6. A small negative bias in the reactivity worth of most control rod absorber types is observed, which indicates the MCNP6 worth is slightly higher at the BOL fresh fuel conditions of the analysis.

These lattice reactivity comparisons with MCNP6 show there are no large systematic biases that need to be addressed before providing the CASMO5 data to SIMULATE5 for application of U-235 enrichments ≤ 10 wt%.

Table 2–1. CASMO5 Beginning of Life (BOL) Reactivity Benchmarking vs. MCNP6.

Reactivity Worth Comparison	Lattice Type	Mean (% Difference)	Std. Deviation (% Difference)
Soluble Boron Worth	17x17	0.08	0.09
	15x15	0.14	0.13
	14x14 (W)	0.18	0.13
	14x14 (CE)	0.19	0.20
AIC Control Rods	17x17	-1.55	0.24
	15x15	-1.68	0.31
	14x14 (W)	-1.35	0.31
	14x14 (CE)	-1.55	0.46
B4C Control Rods	17x17	-2.88	0.21
	14x14 (CE)	-3.41	0.40
Hafnium Rods	17x17	-0.23	0.11
Tungsten Control Rods	17x17	-2.93	0.34
IFBA @ 1.5 mg	17x17 (16 Rods)	-0.23	0.09
	17x17 (264 Rods)	0.16	0.15
	15x15 (16 Rods)	-0.95	0.22
	15x15 (148 Rods)	-0.53	0.12
	14x14 (16 Rods)	-0.18	0.15
	14x14 (120 Rods)	0.04	0.17
IFBA @ 3.0 mg	17x17 (16 Rods)	-0.34	0.15
	17x17 (264 Rods)	0.19	0.16
	15x15 (16 Rods)	-1.02	0.22
	15x15 (148 Rods)	-0.53	0.09
	14x14 (16 Rods)	-0.25	0.10
	14x14 (120 Rods)	0.11	0.14
IFBA @ 1.5 mg, 8.0 w/o Gadolinium	17x17 (144 IFBA, 16 Gd)	-0.20	0.14
IFBA @ 1.5 mg, 16.0 w/o Gadolinium	17x17 (144 IFBA, 16 Gd)	-0.34	0.17
Removable Discrete BP Rods (Pyrex)	17x17 (4 Rods)	-1.15	0.18
	17x17 (24 Rods)	-1.12	0.14
	15x15 (4 Rods)	-1.06	0.32
	15x15 (20 Rods)	-1.12	0.22
	14x14 (4 Rods)	-0.85	0.17
	14x14 (16 Rods)	-1.01	0.20
Removable Discrete BP Rods (Wet Annular Burnable Absorber)	17x17 (4 Rods)	-0.79	0.17
	17x17 (24 Rods)	-0.80	0.10
	15x15 (4 Rods)	-0.78	0.17

Reactivity Worth Comparison	Lattice Type	Mean (% Difference)	Std. Deviation (% Difference)
	15x15 (20 Rods)	-0.85	0.05
	14x14 (4 Rods)	n/a	n/a
	14x14 (16 Rods)	n/a	n/a
Fixed Burnable Absorber (B4C)	14x14 (CE) (4 Rods)	-0.69	0.15
	14x14 (CE) (16 Rods)	-0.66	0.11
2.0 w/o Gadolinium	17x17 (4 Rods)	-1.12	0.12
	17x17 (16 Rods)	-0.51	0.06
	15x15 (4 Rods)	-0.07	0.05
	15x15 (16 Rods)	-0.45	0.10
	14x14 (4 Rods)	-1.10	0.15
	14x14 (12 Rods)	-0.36	0.13
	14x14 (CE) (4 Rods)	0.26	0.19
	14x14 (CE) (16 Rods)	0.41	0.08
8.0 w/o Gadolinium	17x17 (4 Rods)	-1.27	0.13
	17x17 (16 Rods)	-0.63	0.08
	15x15 (4 Rods)	-0.03	0.16
	15x15 (16 Rods)	-0.53	0.13
	14x14 (4 Rods)	-1.27	0.05
	14x14 (12 Rods)	-0.46	0.06
	14x14 (CE) (4 Rods)	0.31	0.08
	14x14 (CE) (16 Rods)	0.51	0.12
16.0 w/o Gadolinium	17x17 (4 Rods)	-1.51	0.18
	17x17 (16 Rods)	-0.92	0.19
Moderator Temp Reactivity Defect	17x17	0.08	1.90
	15x15	0.14	2.17
	14x14 (W)	0.66	2.85
	14x14 (CE)	-0.65	0.91
Fuel Temp Reactivity Defect	17x17	2.93	1.96
	15x15	2.38	1.91
	14x14 (W)	2.36	1.91
	14x14 (CE)	2.55	1.93

2.3 Benchmarks of Critical Experiments

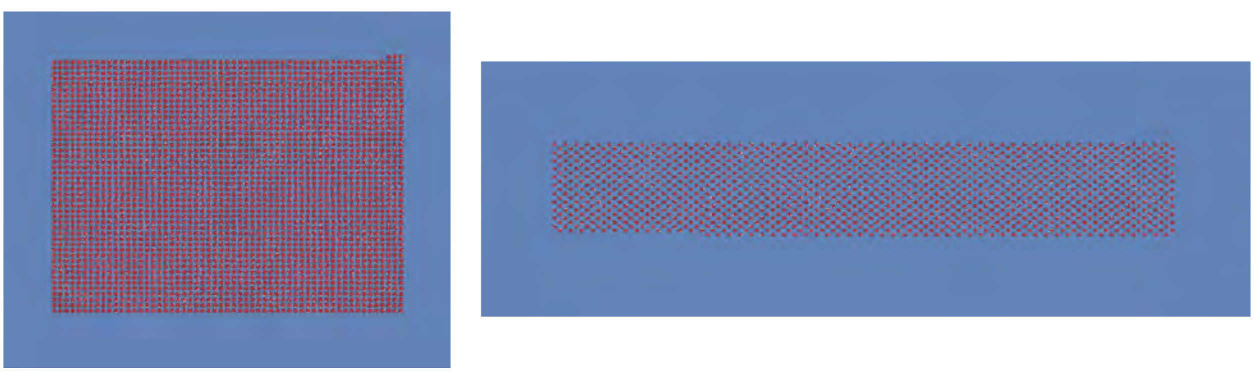
The following critical experiments have been identified for benchmark comparison within the International Criticality Safety Benchmark Evaluation Project (ICSBEP):

- LEU-COMP-THERM-022 (LCT-22): Hexagonal Pitch 10 wt.% U-235 Critical Experiments
- LCT-23: Hexagonal Pitch 10 wt.% U-235 Partially Flooded Critical Experiments
- LCT-24: Square & Hexagonal Pitch 10 wt. % U-235 Critical Experiments
- LCT-25: Hexagonal Pitch 7.5 wt. % U-235 Critical Experiments
- LCT-81: Otto Hahn 3.5-6.6 wt. % U-235 Critical Experiment

These benchmarks with enrichments > 5% complement prior benchmarks for enrichments < 5% with thermal-spectrum, water-moderated critical configurations involving arrays of cylindrical pin cells that have been previously modeled in CASMO5 [2.4].

For LCT-22, -23, -24, and -25 [2.5], the fuel is UO_2 with U-235 enriched to 7.5% or 10%, the moderator is water, and the cladding is stainless steel. Both hexagonal and square pitch lattice configurations are included.

The hexagonal fuel pin pitches are 0.7, 0.8, 1.0, 1.22, 1.4, 1.83, and 1.852 cm for LCT-22. The early configurations have more pins that are tightly packed, whereas the later configurations have fewer pins that are spread out. The primary reactivity control was adding fuel pins one-at-a-time to the core. The hexagonal pin pitch was 1.4 cm for LCT-23. The unique aspect of LCT-23 was that the core was partially flooded (that is, in contrast to the other criticals studied here, the water level was below the top of the fuel). LCT-24 used a square pitch of 0.62 cm. The second configuration used an unusually long, thin aspect ratio, rather than the normal circular or square layout, as shown in Figure 2–2. In contrast to the other three experiments at 10% enrichment, LCT-25 employed fuel enriched to 7.5%.



(a) Array 1 (b) Array 2

Figure 2–2. Fuel configurations of LCT-24. [2.5]

The axial buckling for each configuration is computed using MCNP6, based on the input from the corresponding ICSBEP handbook descriptions. The eigenvalue (*k*) results for the four series of criticals are presented in Table 2–2 for LCT-22, Table 2–3 for LCT-23, Table 2–4 for LCT-24, and Table 2–5 for LCT-25, along with the computed axial buckling and the MCNP6 eigenvalue standard deviation (σ_k). Two CASMO5 eigenvalues are given – those with the code defaults, which assume Transport-Corrected P_0 (TCP0), and those with the defaults plus P_3 scattering.

This set of experiment benchmarks demonstrates that CASMO5 satisfactorily predicts the reactivity for cores with fuel of 7.5% to 10% enrichment composed of various lattice array types. The MCNP6 predicted reactivities show a larger variation than for most other experiments typically analyzed by Studsvik (see [1.1] and [2.4]), up to 1% Δk/k discrepancies. Delta-k values, defined as the differences between CASMO5 and MCNP6 eigenvalues, are plotted in Figure 2–3 for each of these 16 experimental benchmark arrays.

Since both deterministic and Monte Carlo calculations are based on the same evaluated nuclear data, the delta-k values reflect differences in numerical approximations, as well as treatment of the axial leakage. As shown in Figure 2–2, the second configuration of LCT-24 features an unusually long, thin aspect ratio which creates a strong anisotropic effect in the scattering sources. Hence, the use of P_3 scattering in CASMO5 significantly improved the accuracy of the predicted eigenvalue relative to the default TCP0. LCT-23 array 4 uses an unusually large axial buckling approximation value in CASMO5 due to the large axial leakage in the core, which is explicitly captured in the three-dimensional MCNP6 calculation. The rest of the configurations exhibit improvement in the eigenvalue agreement relative to MCNP6 for P_3 scattering in CASMO5, although delta-k values using the default TCP0 remain within 500 pcm from the MCNP6 solution, which is deemed to be acceptable.

Overall, the goal of modeling these critical experiments is to demonstrate the validity of the CASMO5 methods with respect to higher enrichments. The use of P_3 scattering captures the effects of the unusual geometry of some configurations and is unrelated to the use of higher enrichment fuel. Likewise, configurations involving large axial leakage may challenge the axial buckling approximation but do not arise from the use of higher enrichment fuel in these configurations.

Table 2–2. LCT-22 MCNP6 and CASMO5 eigenvalue results.

Array	B^2_z (cm^{-2})	MCNP6 *k*	MCNP6 σ_k	CASMO5 *k*	CASMO5 P_3 *k*
1	1.02E-03	1.00264	0.00011	1.00618	1.00380
2	1.04E-03	1.00686	0.00012	1.01045	1.00676
3	1.06E-03	1.00735	0.00012	1.01123	1.00663
4	1.06E-03	1.00783	0.00012	1.01171	1.00730
5	1.07E-03	1.00341	0.00011	1.00695	1.00302
6	1.06E-03	1.00147	9.00E-05	1.00611	1.00387
7	1.06E-03	1.00387	9.00E-05	1.00878	1.00666

Table 2–3. LCT-23 MCNP6 and CASMO5 eigenvalue results.

Array	B^2_z (cm^{-2})	MCNP6 *k*	MCNP6 σ_k	CASMO5 *k*	CASMO5 P_3 *k*
4	4.59E-03	1.00148	0.000164	0.99873	0.99658
5	3.24E-03	1.00193	0.000161	1.00256	0.99981
6	1.85E-03	1.00258	0.000154	1.00494	1.00146

Table 2–4. LCT-24 MCNP6 and CASMO5 eigenvalue results.

Array	B^2_z (cm^{-2})	MCNP6 k	MCNP6 σ_k	CASMO5 k	CASMO5 P_3 k
1	1.02E-03	1.00130	0.00011	1.00400	1.00241
2	1.05E-03	1.00880	0.00012	1.01504	1.01072

Table 2–5. LCT-25 MCNP6 and CASMO5 eigenvalue results.

Array	B^2_z (cm^{-2})	MCNP6 k	MCNP6 σ_k	CASMO5 k	CASMO5 P_3 k
1	1.01E-03	0.98862	0.000167	0.99105	0.98912
2	1.03E-03	0.99597	0.000162	0.99864	0.99565
3	1.03E-03	1.00034	0.000164	1.00348	0.99999
4	1.04E-03	1.00174	0.000154	1.00512	1.00205

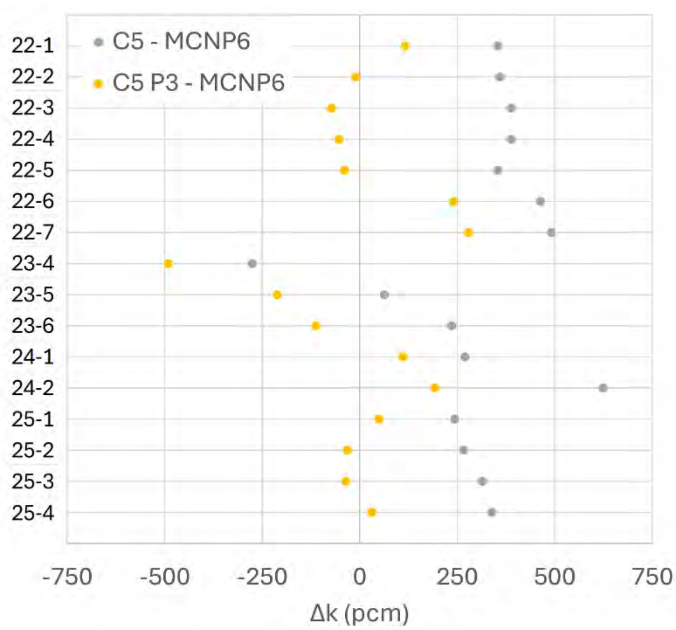


Figure 2–3. Eigenvalue differences between CASMO5 (C5) and MCNP6 for the various experiment configurations of LCT-22, LCT-23, LCT-24, and LCT-25.

Another suitable benchmark has been identified in the ICSBEP. The Otto Hahn nuclear ship program created small, high-leakage pressurized light-water reactors for propulsion of large ships. As part of the program, a second core was constructed for the ship, and tested for subcriticality at cold conditions, in a series of tests which occurred in 1972. This second core is evaluated in LEU-COMP-THERM-081 (LCT-81).

The core, pictured in Figure 2–4, has fuel with 3.5 wt% and 6.6 wt% U-235 enrichment. Burnable poison pins are present and are composed of ZrO_2 and ZrB_2. In full symmetry, twelve 17x17 assemblies are arranged with a small number of fuel pins in the corners. Each corner assembly contains four oversized 'G' rods, which consist of fuel and boron absorber. The square pins in the assembly corners are structural supports.

An axial buckling value is computed using MCNP6, based on the input from the description in the ICSBEP handbook. The cladding outer diameter for the large 'G' fuel rod with a B_4C core is approximated in order to fit inside a regular lattice pitch location.

Excellent agreement is found between the CASMO5 eigenvalue (k) and the three-dimensional MCNP6 estimated k [2.6], as shown in Table 2–6. A comparison of the two-dimensional fission rate statistics between CASMO5 and MCNP6 is shown below in Table 2–7, also indicating very good agreement.

Table 2–6. LCT-81 MCNP and CASMO5 eigenvalue results.

B^2_z (cm^{-2})	MCNP6 k	MCNP6 σ_k	CASMO5 k
1.0917E-03	0.99816	0.000004	0.99715

Table 2–7. LCT-81 MCNP6 and CASMO5 two-dimensional fission rate differences.

Quantity	Absolute	Relative
Maximum	0.5%	1.2%
Minimum	-0.6%	-0.5%
Average	0.0%	0.1%
Root Mean Square (RMS)	0.2%	0.3%

2.4 Benchmarking of SIMULATE5

In addition to the CASMO5 comparisons, SIMULATE5 is also benchmarked against the small, high-leakage PWR Otto Hahn critical experiment (LCT-81) [2.6] that contains UO_2 fuel enriched to 3.5 wt% and 6.6 wt.%. A direct comparison is made in both two- and three-dimensional configurations. The model devised for the Otto Hahn critical experiment in LCT-81 is well-suited for such comparisons. Figure 2–4 shows the quarter-core loading configuration used for LCT-81, indicating orientation of the assemblies for this analysis (relative orientation with cardinal North being at the top of the image).

Excellent agreement is found between the two-dimensional SIMULATE5 eigenvalue (k-eff) and both the MCNP6 and CASMO5 estimated k-eff, as shown in Table 2–8. Both CASMO5 and SIMULATE5 use the same axial buckling value for this k-eff comparison.

The predicted two-dimensional assembly relative power factors (RPF) and fission rate distribution from SIMULATE5 and MCNP6, assuming zero buckling, were also computed. The reference relative assembly powers from MCNP6 and the differences between SIMULATE5 and MCNP6 are given in Figure 2–5. A comparison of the fission rate statistics between SIMULATE5 and MCNP6 is also shown below in Table 2–9. Note that the assemblies at SW and NE core locations are symmetric and have equivalent results within the roundoff. Accordingly, subsequent tables and figures present only SW core location data.

Excellent agreement is found between the three-dimensional SIMULATE5 eigenvalue (k-eff) and the MCNP6 k-eff, as shown in Table 2–10. The MCNP6 and SIMULATE5 axial plane (enumerated bottom node 1 to top node 12) relative nodal powers, and the difference between SIMULATE5 and MCNP6, are plotted in Figure 2–6 and provided in Table 2–11 and Table 2–12, respectively. The SIMULATE5 predicted assembly-wise axial power distributions match those of the reference MCNP6 predictions very well with a maximum error of less than 1%, excluding the bottom and the top axial planes, which have slightly larger but less significant errors being at very low relative nodal powers.

A summary comparison of pin-wise fission rate statistics between SIMULATE5 and MCNP6 for each axial plane is shown below in Table 2–13 and Table 2–14. There is a slight increase in absolute error on mid-core planes with higher peaking factors. The SIMULATE5 predicted planar peak pin fission rates are within 2.8% absolute difference and 1.5% relative difference, excluding the bottom and top axial planes.

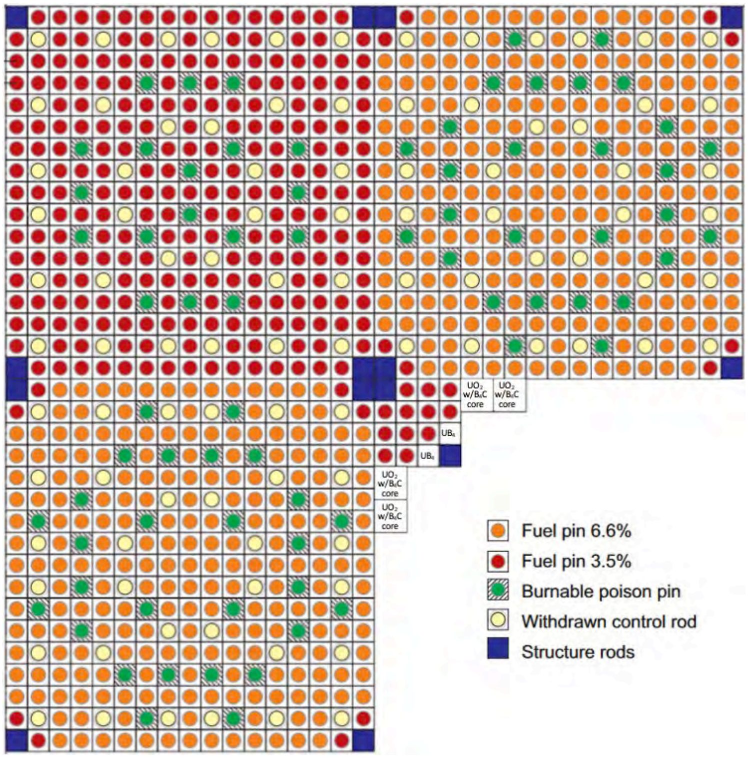


Figure 2–4. Otto-Hahn quarter-core loading configuration in LCT-81 [2.6].

RPF (MCNP6)

1.63	1.17
1.17	0.04

(SIMULATE5-MCNP6)*100

-0.3	0.1
0.1	0.1

Figure 2–5. Absolute difference in relative assembly powers between SIMULATE5 and MCNP6 LCT-81 two-dimensional calculations.

Table 2–8. SIMULATE5 eigenvalue comparisons for the LCT-81 two-dimensional model.

Code	Eigenvalue	Delta-k (pcm) vs MCNP6	Delta-k (pcm) vs CASMO5
MCNP6	0.99816 (±0.000004)		
CASMO5	0.99715	-101	
SIMULATE5	0.99891	75	176

Table 2–9. SIMULATE5 v. MCNP6 LCT-81 two-dimensional fission rate statistics.

	Quantity	Assembly NW	Assembly SW	Assembly SE	Core
Absolute % Difference	Max	2.25	9.13	2.98	9.13
	Min	-1.91	-4.59	-2.78	-4.59
	Ave	-0.23	0.05	1.35	0.00
	RMS	0.90	1.93	2.18	1.66
	Peak Pin	-0.29	-1.53	2.66	-1.53
Relative % Difference	Max	2.54	14.88	5.84	14.88
	Min	-1.67	-7.37	-5.94	-7.37
	Ave	-0.13	0.60	3.21	0.0
	RMS	0.80	2.69	4.71	2.32
	Peak Pin	-0.19	-0.86	4.27	-0.86

Table 2–10. SIMULATE5 eigenvalue comparisons for the LCT-81 three-dimensional model.

Code	Eigenvalue	Delta-k (pcm) vs MCNP6
MCNP6	0.998158 (±0.000013)	
SIMULATE5	0.99702	-113

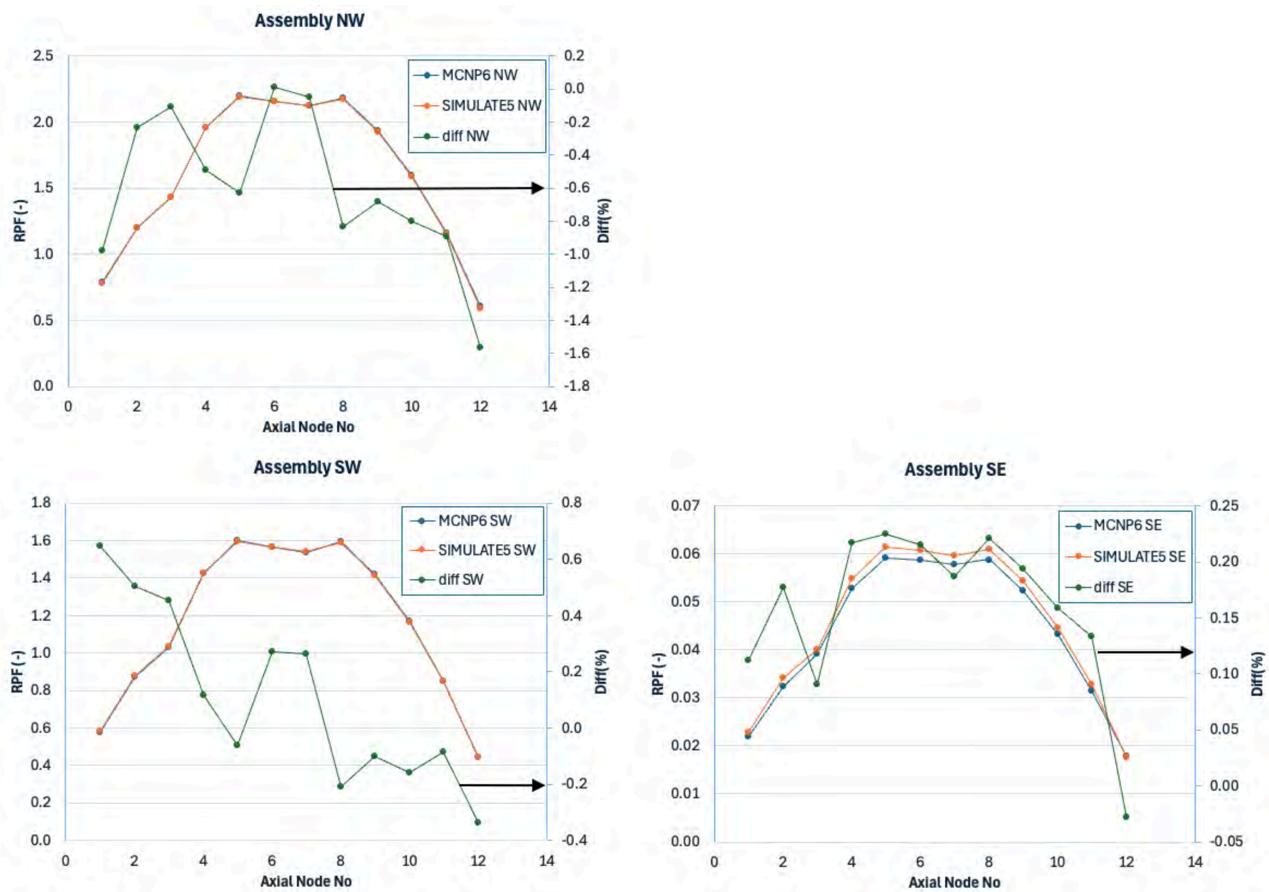


Figure 2–6. Absolute difference in axial relative powers between the LCT-81 three-dimensional SIMULATE5 and MCNP6 calculations.

Table 2–11. Absolute difference in axial powers between the three-dimensional SIMULATE5 and MCNP6 LCT-81 calculations.

Axial Plane	MCNP6 - Relative Power			Absolute % Difference (SIMULATE5-MCNP6) x100		
	Assm NW	Assm SW	Assm SE	Assm NW	Assm SW	Assm SE
12	0.605	0.444	0.018	-1.57	-0.34	-0.03
11	1.160	0.850	0.032	-0.89	-0.09	0.13
10	1.593	1.168	0.043	-0.80	-0.16	0.16
9	1.937	1.418	0.052	-0.68	-0.10	0.19
8	2.183	1.593	0.059	-0.83	-0.21	0.22
7	2.122	1.539	0.058	-0.05	0.26	0.19
6	2.159	1.566	0.059	0.01	0.27	0.21
5	2.197	1.598	0.059	-0.63	-0.06	0.22
4	1.961	1.427	0.053	-0.49	0.12	0.22
3	1.429	1.031	0.039	-0.11	0.45	0.09
2	1.204	0.873	0.032	-0.23	0.50	0.18
1	0.788	0.576	0.022	-0.98	0.65	0.11
Average	1.612	1.174	0.044	-0.60	0.11	0.16

Table 2–12. Relative difference in axial powers between the three-dimensional SIMULATE5 and MCNP6 LCT-81 calculations.

Axial Plane	MCNP6 - Relative Power			Relative % Difference (SIMULATE5-MCNP6)/MCNP6 x100		
	Assm NW	Assm SW	Assm SE	Assm NW	Assm SW	Assm SE
12	0.605	0.444	0.018	-2.59	-0.76	-1.57
11	1.160	0.850	0.032	-0.77	-0.10	4.24
10	1.593	1.168	0.043	-0.50	-0.13	3.68
9	1.937	1.418	0.052	-0.35	-0.07	3.70
8	2.183	1.593	0.059	-0.38	-0.13	3.76
7	2.122	1.539	0.058	-0.02	0.17	3.23
6	2.159	1.566	0.059	0.00	0.17	3.67
5	2.197	1.598	0.059	-0.29	-0.04	3.80
4	1.961	1.427	0.053	-0.25	0.08	4.11
3	1.429	1.031	0.039	-0.08	0.44	2.30
2	1.204	0.873	0.032	-0.19	0.58	5.47
1	0.788	0.576	0.022	-1.24	1.12	5.13
Average	1.612	1.174	0.044	-0.37	0.09	3.62

Table 2–13. Absolute differences in pin fission rates between the three-dimensional SIMULATE5 and MCNP6 for LCT-81.

Axial Plane	RMS	Max Diff	Min Diff	MCNP6 Peak Pin	Peak Pin Absolute % Diff (SIMULATE5-MCNP6) x100
12	1.06	3.35	-2.21	0.67	1.66
11	1.36	6.83	-3.22	1.28	1.67
10	1.80	9.30	-4.38	1.76	2.17
9	2.13	11.42	-5.19	2.14	2.53
8	2.41	12.55	-5.97	2.41	2.76
7	2.20	12.22	-6.49	2.33	1.65
6	2.24	12.60	-6.45	2.37	2.15
5	2.39	12.68	-5.91	2.42	2.32
4	2.13	11.47	-5.25	2.15	1.92
3	1.52	8.56	-3.93	1.57	0.81
2	1.34	7.23	-3.20	1.32	1.42
1	1.20	5.13	-1.90	0.87	-0.45

Table 2–14. Relative differences pin fission rates between the three-dimensional SIMULATE5 and MCNP6 for LCT-81.

Axial Plane	RMS	Max diff	Min Diff	MCNP6 Peak Pin	Peak Pin Relative % Diff. (SIMULATE5-MCNP6)/MCNP6x100
12	3.00	13.85	-10.47	0.67	2.54
11	2.57	15.38	-7.17	1.28	1.32
10	2.51	15.33	-7.14	1.76	1.25
9	2.50	15.52	-6.95	2.14	1.19
8	2.48	15.13	-7.14	2.41	1.16
7	2.44	15.05	-7.75	2.33	0.72
6	2.46	15.31	-7.62	2.37	0.92
5	2.49	15.22	-7.04	2.42	0.97
4	2.53	15.43	-7.00	2.15	0.90
3	2.60	15.61	-6.88	1.57	0.52
2	2.72	15.82	-6.91	1.32	1.09
1	3.65	16.78	-6.56	0.87	-0.51

2.5 Conclusion

The results of this analysis show that CASMO5 and SIMULATE5 can accurately model critical experiments involving higher enrichment without methodology changes or adjustments. Comparisons of CASMO5 versus high-order MCNP6 Monte Carlo solutions for BOL lattices spanning various conditions indicate that there is no significant bias in eigenvalue for higher enrichments relative to previous work [1.1]. The Otto Hahn benchmark (LCT-81) results presented in this section confirm the ability of CASMO5 and SIMULATE5 to model reactors involving higher enrichments and high-leakage small PWR cores. In conclusion, higher enrichments pose no challenge to the methods implemented in CASMO5 and SIMULATE5.

2.6 Section 2 References

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3. CMS5 Application for Maximum Rod-Average Burnup to 80 GWd/MTU

3.1 Motivation

For reasonable reload batch sizes, an extension to higher enrichment that enables longer fuel cycle length is expected to increase the burnup of discharged fuel. Therefore, extension of the CMS5 approved enrichment range targets an increase in the approved burnup limit, specifically for the peak (equivalently termed "maximum") rod-average burnup limit of 62 GWd/MTU provided in Ref. [1.2]. Several challenges arise for benchmarking against data at high exposures due to the data often being held as proprietary to a given fuel vendor. While such data are important to qualify the specific fuel thermo-mechanical performance at higher burnup, nuclear methods and neutronics are affected to a much lesser degree and retain a strong dependence only upon the isotopic characteristics as exposures increase.

To this point, an approach is pursued that includes comparison to higher-order reference solutions with detailed tracking of isotopic depletion [3.1]. Higher-order reference solutions are obtained from Serpent2 [3.2] depletion calculations and used to benchmark CASMO5 results. Detailed depletion calculations are performed for a typical PWR pin cell (using representative composition, temperature, and power densities of typical PWR fuel) but with higher enrichment. Furthermore, Serpent2 is used to benchmark CASMO5 by considering full 17x17 PWR assembly depletion calculations with higher enrichments.

Even with comparisons to such higher-order calculations, a benchmark to measured data is imperative to justify use of the CMS5 methodology for higher rod-average burnups. Thus, a comparison is made to meaningful burnup measurements obtained from fuel rods operated to relevantly high exposures, as presented in Section 3.4.

3.2 Pin-cell Depletion Comparison to Higher-Order Solutions

A single fuel pin cell, consisting of a uranium-fueled pellet, helium-filled pellet-cladding gap, a Zircaloy-2 cladding tube and a water-filled surrounding, was depleted with both CASMO5 and Serpent2 at the conditions specified below. Reflective boundary conditions were applied at the pin cell periphery. The resulting eigenvalues from CASMO5 are then compared to corresponding reference values computed by Serpent2.

Depletion conditions:

- 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 CASMO5 versus Serpent2 results tabulated in Table 3–1 show no significant bias versus enrichment. As shown in Figure 3–1, excellent agreement is found between the CASMO5 and Serpent2 predicted pin cell eigenvalues. (Note the tables and figures below may represent Serpent2 results as "S2" and CASMO5 results as "C5").

Table 3–1. Single Fuel Pin Cell Eigenvalue Comparisons (CASMO5-Serpent2) [pcm] for the Enrichments 5, 8 and 10 wt. % U-235.

Enrichment (wt%)	5.00	8.00	10.00
Maximum Burnup (GWd/MTU)	80.00	80.00	80.00
Sample Size	63	63	63
Average Difference (pcm)	67.3	107.2	128.2
Standard Deviation (pcm)	50.3	33.4	41.9
Maximum Difference (pcm)	202.0	192.0	190.0
Minimum Difference (pcm)	9.0	18.0	19.0

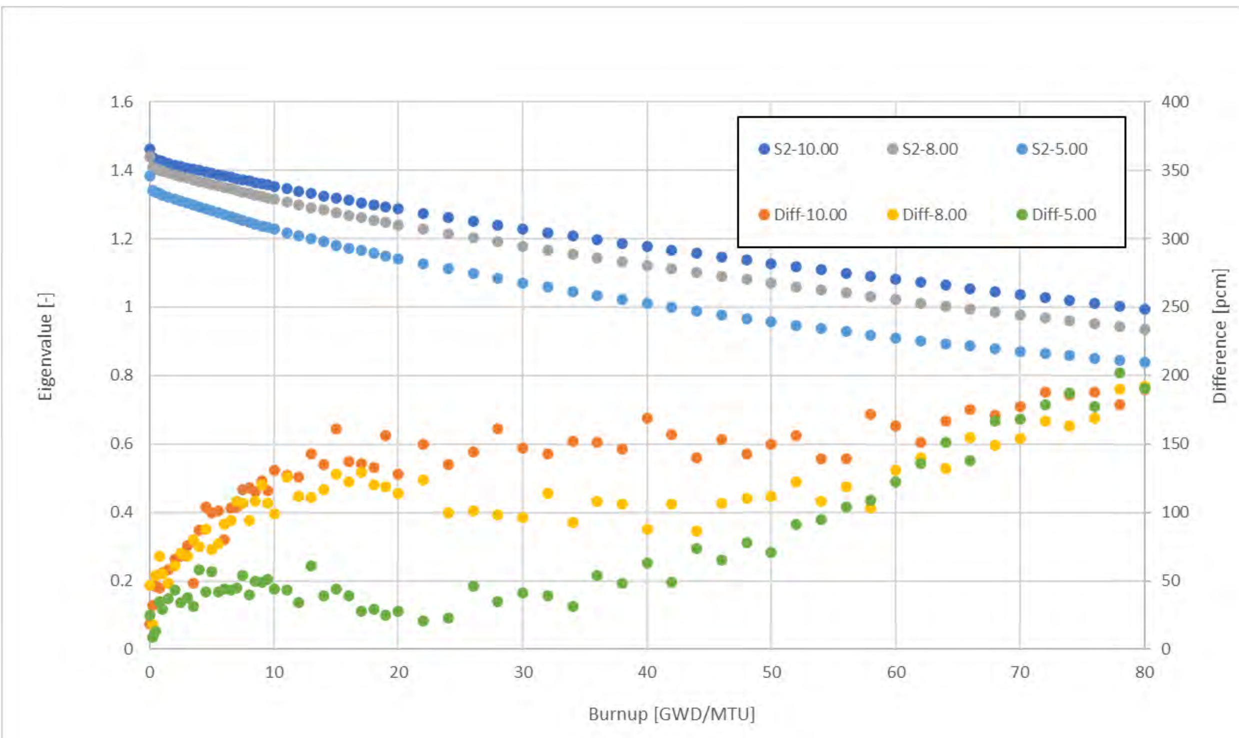

Figure 3–1. Single Fuel Pin Cell Eigenvalue Comparisons (CASMO5-Serpent2) [pcm] for Enrichments 5, 8, and 10 wt. % U-235.

3.3 Lattice Depletion Comparison to Higher-Order Solutions

A generic PWR fuel assembly lattice has been devised that contains both IFBA fuel rods (i.e., fuel pellets coated with a thin layer of zirconium diboride) and integral gadolinium fuel rods. Figure 3–2 displays the key features of this generic PWR fuel assembly lattice. The generic PWR fuel assembly configuration was depleted with CASMO5 and Serpent2 at nominal conditions. The resulting eigenvalues and fission rates from CASMO5 are then compared to corresponding reference values computed by Serpent2. Note this generic PWR lattice is evaluated at conditions similar to the pin cell calculations in Section 3.2, apart from the U-235 enrichment values and soluble boron concentration of 800 ppm. The lattice calculations were conducted at U-235 enrichment values of 5.00 wt%, 6.30 wt%, and 10.00 wt%, to evaluate an enrichment value (6.3 wt%) likely more appropriate for 24-month cycle lengths. Gadolinium content was selected as 6.0 wt%, 8.0 wt%, and 12.0 wt% Gd_2O_3 at the respective U-235 enrichments. Furthermore, the calculations were executed to 90 GWd/MTU exposure only to confirm reasonable behavior at exposures greater than 80 GWd/MTU.

The CASMO5 versus Serpent2 results tabulated in Table 3–2 show no significant bias versus pin cell depletions given by Table 3–1. As shown in Figure 3–3, excellent agreement is found between the CASMO5 and Serpent2 predicted lattice eigenvalues. Pin-wise fission rate distributions are also examined and displayed for the 6.3 wt% enrichment at a burnup of 0 GWd/MTU in Figure 3–4, 22 GWd/MTU in Figure 3–5, 60 GWd/MTU in Figure 3–6, and 90 GWd/MTU in Figure 3–7. Excellent agreement is also found between these CASMO5 and Serpent2 predicted fission rates with comparable level of accuracy for the other evaluated enrichments.

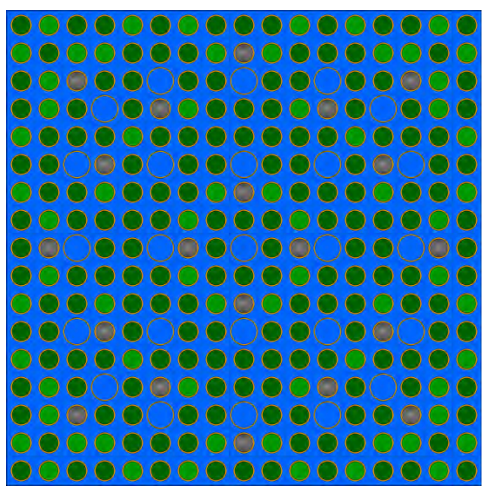

Figure 3–2. Generic PWR Fuel Assembly Lattice Configuration with 176 IFBA pins (dark green), 20 Gd_2O_3 pins (gray), and 68 ordinary fuel pins (light green).

Table 3–2. Generic PWR Assembly Lattice Eigenvalue Comparisons (CASMO5-Serpent2) [pcm] for the Enrichments 5, 6.3 and 10 wt. % U-235.

Enrichment (wt%)	5.00	8.00	10.00
Maximum Burnup (GWd/MTU)	90.00	90.00	90.00
Sample Size	71	86	132
Average Difference (pcm)	120.1	153.9	209.0
Standard Deviation (pcm)	44.8	44.2	56.9
Maximum Difference (pcm)	216.0	245.0	303.0
Minimum Difference (pcm)	57.0	77.0	107.0

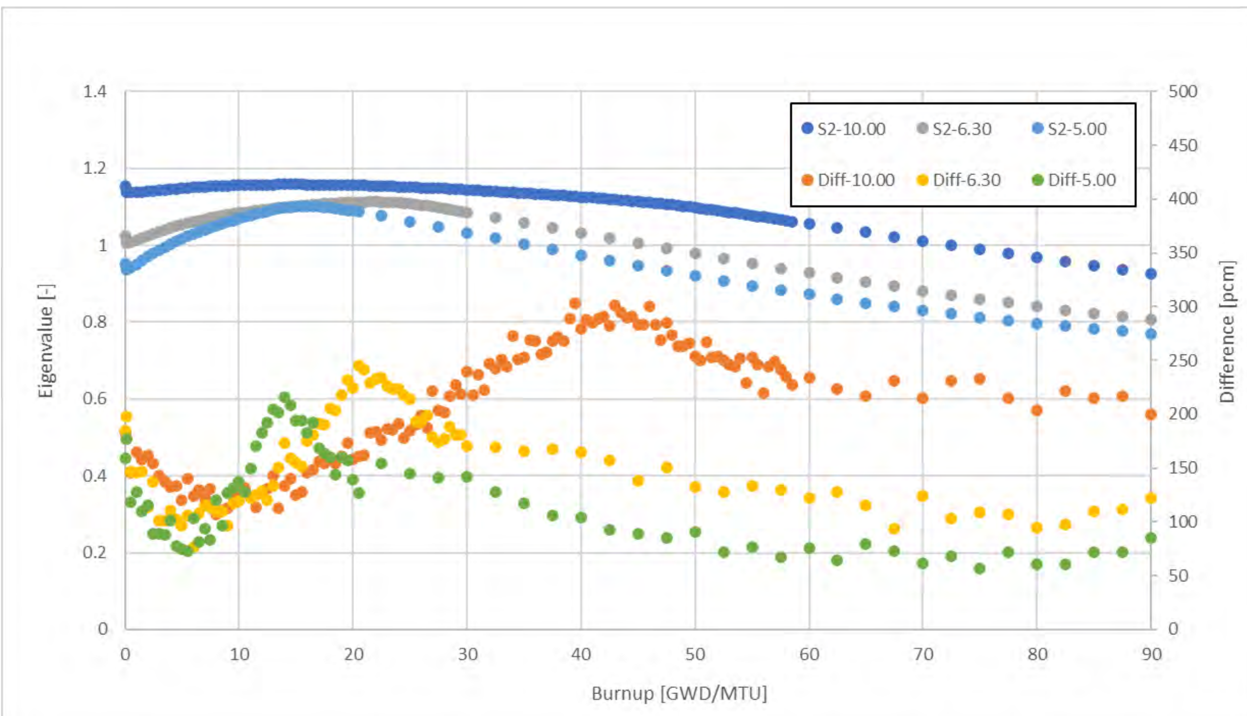

Figure 3–3. Generic PWR Assembly Lattice Eigenvalue Comparisons (CASMO5-Serpent2) [pcm] for Enrichments 5, 6.3, and 10 wt. % U-235.

	1	2	3	4	5	6	7	8	9
		S2 Ref.							
1		C5-S2			Burnup		0.000		
	1.008	0.999			Maximum		0.007		
2	-0.003	-0.003			Minimum		-0.010		
	0.385	1.045	1.038						
3	-0.003	-0.001	-0.006						
		1.067	1.074						
4		-0.010	-0.005						
	1.069	1.035	1.029	1.040	1.091				
5	-0.001	-0.001	-0.003	-0.002	0.001				
	1.071	1.035	1.054	0.389	1.064				
6	-0.005	0.001	0.001	-0.003	-0.001				
		1.064	1.064		1.075	1.042	0.383		
7		-0.008	-0.002		0.005	0.000	0.001		
	0.385	1.046	1.041	1.077	1.053	1.066	1.031	1.017	
8	-0.001	0.000	0.001	-0.001	-0.007	0.005	0.004	0.001	
	0.963	1.006	1.088	1.052	1.098	1.038	1.028	1.088	1.056
9	0.001	0.001	0.003	-0.004	0.004	-0.001	0.003	0.007	0.000

Figure 3–4. Generic PWR Assembly Lattice Pin Fission Rate Comparisons (CASMO5-Serpent2) for 6.3 wt. % U-235 at a Burnup of 0.0 GWd/MTU.

	1	2	3	4	5	6	7	8	9
		S2 Ref.							
1		C5-S2			Burnup		22.000		
	1.052	1.030			Maximum		0.008		
2	0.001	-0.004			Minimum		-0.006		
	0.872	1.026	1.026						
3	-0.001	0.003	-0.001						
		1.056	1.052						
4		-0.006	-0.002						
	1.049	1.022	1.030	1.066	1.054				
5	-0.002	0.000	-0.003	-0.004	0.003				
	1.046	1.018	1.024	0.884	1.077				
6	-0.003	0.001	0.005	-0.001	-0.003				
		1.038	1.043		1.055	1.033	0.786		
7		0.001	-0.001		0.001	-0.002	0.006		
	0.837	1.000	1.002	1.025	0.998	0.974	0.956	0.951	
8	0.006	0.006	0.000	0.001	-0.001	0.001	0.003	-0.005	
	0.987	0.983	0.979	0.977	0.969	0.963	0.956	0.955	0.959
9	-0.001	0.000	0.006	0.005	0.008	0.001	-0.001	-0.001	0.000

Figure 3–5. Generic PWR Assembly Lattice Pin Fission Rate Comparisons (CASMO5-Serpent2) for 6.3 wt. % U-235 at a Burnup of 22.0 GWd/MTU.

	1	2	3	4	5	6	7	8	9
		S2 Ref.							
1		**C5-S2**			Burnup		60.000		
	1.037	1.021			Maximum		0.009		
2	-0.006	0.001			Minimum		-0.008		
	0.915	1.017	1.013						
3	0.000	0.007	0.006						
		1.027	1.030						
4		-0.001	-0.005						
	1.026	1.017	1.019	1.035	1.026				
5	-0.002	0.000	0.000	-0.004	0.008				
	1.030	1.017	1.016	0.912	1.039				
6	-0.008	-0.002	0.006	0.003	-0.003				
		1.020	1.025		1.030	1.020	0.880		
7		-0.001	-0.006		-0.003	-0.005	0.007		
	0.902	1.007	1.003	1.018	1.002	0.990	0.978	0.970	
8	0.002	0.002	0.001	-0.008	-0.003	-0.002	0.003	-0.001	
	0.999	0.999	0.990	0.989	0.980	0.983	0.974	0.964	0.973
9	0.002	-0.003	0.004	0.001	0.009	-0.003	0.001	0.009	0.001

Figure 3–6. Generic PWR Assembly Lattice Pin Fission Rate Comparisons (CASMO5-Serpent2) for 6.3 wt. % U-235 at a Burnup of 60.0 GWd/MTU.

	1	2	3	4	5	6	7	8	9
		S2 Ref.							
1		**C5-S2**			Burnup		90.000		
	1.024	1.011			Maximum		0.009		
2	-0.008	0.002			Minimum		-0.009		
	0.916	1.011	1.009						
3	0.005	0.006	0.003						
		1.020	1.021						
4		-0.006	-0.008						
	1.018	1.011	1.009	1.024	1.020				
5	-0.004	0.000	0.003	-0.009	0.004				
	1.016	1.010	1.008	0.921	1.028				
6	-0.003	0.000	0.007	-0.002	-0.007				
		1.015	1.011		1.024	1.012	0.901		
7		-0.004	0.000		-0.006	-0.004	0.006		
	0.909	1.004	1.005	1.012	1.004	0.990	0.986	0.983	
8	0.007	0.006	0.000	-0.004	-0.001	0.007	0.008	0.002	
	1.003	1.004	0.995	0.995	0.998	0.990	0.990	0.985	0.995
9	0.000	-0.003	0.009	0.003	0.003	0.004	0.001	0.008	-0.004

Figure 3–7. Generic PWR Assembly Lattice Pin Fission Rate Comparisons (CASMO5-Serpent2) for 6.3 wt. % U-235 at a Burnup of 90.0 GWd/MTU.

3.4 Comparison to High-burnup Measurements

A further examination that bolsters the case for CMS5 application at higher burnup involves a lead test assembly used in the North Anna Power Station to very high rod-average exposures from which burnup measurements were obtained [3.3]. While the data and corresponding CMS5 core models are unavailable to Studsvik Scandpower, Dominion Energy – as a utility partner to Studsvik Scandpower and operator of the North Anna units – had evaluated performance of the CMS5 methodology applied to this high-burnup lead test assembly and contributed in justifying the burnup application range of the CMS5 Generic PWR TR, as submitted [1.1]. Being a licensee using the approved CMS5 methodology (see [1.3]), Dominion has performed an updated analysis for this lead test assembly at high rod-average burnup and reports the relative performance of CMS5 predictions in [3.4]. A summary of the results is provided in this section to clearly demonstrate suitability of CMS5 application to higher rod-average burnup.

North Anna Power Station participated in a fuel testing program that included a 17x17 PWR lead test assembly operated to a discharge burnup in excess of 60 GWd/MTU. This lead test assembly was installed in four cycles with the resulting (calculated) burnups shown in Table 3–3. Subsequent post-irradiation examination was performed on four fuel rods extracted from the lead test assembly [3.3] and sampled approximately at the fuel rod mid-plane. Each sample was radio-assayed for a suitable isotope marker to quantify axial burnup distribution measurements, thereby enabling comparison to burnup calculated at the corresponding axial position for each specified fuel rod. Nominal calculations for the comparisons are performed with the standard CMS5 core design process applied to the North Anna cores containing the lead test assembly.

To provide a more complete assessment of the CMS5 methodological capabilities for fuel at the representative high exposures, several cases were executed with variations to this standard CMS5 core design process. Variations applied to the model include differing axial nodalization and submesh options; other variations were applied to accommodate uncertainty in sample axial location associated with nonuniform thermal expansion of fuel rod components. In combination, a total of 12 cases provides reasonable coverage to assess performance of calculated CMS5 results to the burnup measurements. Additionally, the calculated burnup for each axial node is fitted with a spline function to approximate the internodal burnup at the actual sample axial location. Note the comparisons are defined as relative difference of spline-fit CMS5 calculated values to measured values and expressed as % difference.

Table 3–4 lists the resulting comparison of each variant relative to the four fuel rods sampled at the axial midplane. Results for Rods 1, 3, and 4 show excellent agreement with indicated relative difference less than the reported two-sigma measurement uncertainty [3.3]. Rod 2, however, consistently shows a conservative overprediction from the CMS5 methodology, albeit less than 10% in all cases and consistent with the reported 1-sigma measurement uncertainty. This rod was located nearby unfueled water holes (guide thimbles) as compared to the other measured rod locations being surrounded by similar fuel rods. Overall, these calculated-to-measured comparisons are quite reasonable.

Although limited in extent, these comparisons of calculated to measured burnup data at a single axial position in each fuel rod give greater confidence in the CMS5 methodology to adequately preform nuclear analysis of assemblies and rods operated to high burnup. The maximum rod-average burnup from the lead test assembly was 72.6 GWd/MTU, and no data for higher maximum rod-average burnup is presently available to Studsvik Scandpower nor Dominion Energy. Nonetheless, CMS5 performance would not degrade for maximum rod-average burnup of 80 GWd/MTU, as demonstrated by the higher order code comparisons in Section 3.3.

Table 3–3. High Burnup Lead Test Assembly Exposures from North Anna Power Station.

Unit, Cycle	Average Assembly Burnup* (GWd/MTU)	Maximum Rod-Average Burnup* (GWd/MTU)
1C13	22.1	23.9
1C14	46.8	48.5
1C15	53.2	56.9
2C16	68.6	72.6

* Reported results are from the licensee standard core design process.

Table 3–4. Relative Difference of Calculations to Measurements for Fuel Rods in the High Burnup Lead Test Assembly.

	Rod 1	Rod 2	Rod 3	Rod 4
Case 1	-2.9%	6.7%	0.7%	1.0%
Case 2	-3.0%	6.6%	0.6%	0.9%
Case 3	-3.1%	6.5%	0.5%	0.8%
Case 4	-0.6%	8.9%	3.4%	3.8%
Case 5	-0.7%	8.8%	3.3%	3.7%
Case 6	-0.9%	8.7%	3.1%	3.5%
Case 7	-3.5%	6.8%	0.4%	0.2%
Case 8	-3.6%	6.7%	0.2%	0.1%
Case 9	-3.7%	6.6%	0.1%	0.0%
Case 10	-1.3%	9.1%	3.0%	2.9%
Case 11	-1.4%	9.0%	2.8%	2.8%
Case 12	-1.6%	8.8%	2.6%	2.6%

3.5 Conclusion

Comparisons between CASMO5 and high-order reference solutions, provided by the Serpent2 Monte Carlo code, indicate no significant bias in reactivity versus increased enrichment as a function of burnup. The differences in eigenvalue and fission rate distributions for pin cell and lattice depletion are well within the expected level of agreement between CASMO5 and Serpent2 Monte Carlo reference solutions. Additionally, CMS5 results compare quite well to measured burnup data of pins in a fuel assembly operated to relatively high exposure in an operating PWR. In conclusion, higher enrichment and high burnup poses no challenge to the methods implemented in CMS5.

3.6 Section 3 References

3.1. P. Forslund Guimarães, "CASMO5-Serpent2 Fuel Pin Cell and Lattice Reactivity, Fission Rate and Number Density Comparisons for Higher Burnups and Enrichments," SSP-24-P01/009-CN, Idaho Falls ID, 2024.

3.2. J. Leppänen, M. Pusa, T. Viitanen, V. Valtavirta, and T. Kaltiaisenaho. "The Serpent Monte Carlo code: Status, development and applications in 2013." Ann. Nucl. Energy, 82 (2015) 142-150.

3.3. EPRI-1019102, "Assessment of Hot Cell Examination of AREVA M5(R) Guide Tubes and Fuel Rods Irradiated in North Anna 1 and 2," Sep 2009. ([https://www.epri.com/research/products/000000000001019102])

3.4. E. Frishholtz, "Summary of CMS5 Burnup Comparison for North Anna High Burnup Fuel Assembly FM3," Dominion Energy Engineering Technical Evaluation ETE-NAF-2024-0113, Rev 0, Jan 2025.

4. CMS5 Application to LWR Small Modular Reactors

4.1 Motivation

The CMS5 methodology has seen review and approval by NRC not only in the CMS5 Generic PWR TR [1.2], but also in application to the NuScale SMR [4.1]. Building upon this independent review and approval, Studsvik Scandpower clarifies the CMS5 methodology of Ref. [1.1] could be generically referenced in licensing applications for existing western PWRs as well as SMR concepts for which the designs meet the methodology application range limits. This may include reactor vendors considering design certification for their SMR by the NRC. Specifically, the CMS5 methods, verification and validation basis, and nuclear uncertainty factor/nuclear reliability factor methodology approved in the Safety Evaluation Report [1.2] are applicable to PWR SMRs within the constraints of the methodology, including the extensions to enrichment and burnup covered in this TR Supplement 1.

4.2 Justification

For the purpose of CMS5 methodology application in PWRs, a distinction is hereby given between "large" PWRs exemplified by the types of PWR reactors currently licensed by NRC for operation in the United States, and "small" PWRs exemplified by reactors systems that may or may not have an NRC-certified reactor design or approval for operation. This distinction presumes a notable degree of independence on rated thermal power, method of manufacture and assembly ("modular" compared to "on-site"), and other common terms of trade in the commercial nuclear power industry. Rather, the distinction is based upon PWR core and fuel configurations demonstrably important to nuclear performance, concerning elements such as reactivity management, isotopics tracking, and fluence to core structures.

Key configuration factors in this distinction are thus core size (namely diameter and height), total number of fuel assemblies, fuel lattice design, mechanism for excess reactivity control, and coolant/moderator flow control (forced or natural convection). A "large" PWR may then have a sizeable vessel containing many fuel assemblies such that flux and power distributions may be quite flat with minimal axial offset, whereas a pressurized light water SMR might for example have a comparatively smaller vessel and fewer (volumetrically proportional) fuel assemblies.

A typical Westinghouse 4-loop reactor with 193 fuel assemblies is characterized as the former, whereas the (presently) certified standard design of the NuScale SMR [4.2] is characterized as the latter. The CMS5 methodology has been approved for use in both large PWRs and SMR PWRs, thus demonstrating adequacy of the methodology to capture the nuclear performance of these varied PWR configurations.

Furthermore, the LCT-81 CMS5 benchmark results presented in Section 2 for the Otto Hahn reactor clearly demonstrate methodological adequacy for a quite different SMR. The Otto Hahn core configuration is not only small in diameter and height (active fuel region less than 3 feet), but also utilizes "non-standard" PWR lattice designs, particularly with the corner assembly layout. The small core with steep local flux and power gradients represents a more extreme variation to those expected of other PWR SMR concepts. In fact, NRC staff have previously exercised CASMO5 to provide cross-sections for their own benchmark [4.3] of the Otto Hahn reactor, finding reasonable performance of the CASMO5 multi-assembly MxN methodology.

These observations provide justification for generic use of the CMS5 methodology for PWR SMRs should those SMRs show compliance with code system range and validity requirements, in the same manner for application to the "large" PWRs. A presumed SMR licensee invoking generic use of the CMS5 methodology, by way of reference to the CMS5 Generic PWR TR and subsequently to this TR Supplement 1, must still determine the nuclear uncertainty factors/nuclear reliability factors specific to their designs, or justify use of the approved generic nuclear reliability factors in the CMS5 Generic PWR TR (as elected by NuScale in

[4.1]). Adoption and use of the CMS5 methodology must still occur as outlined in Ref. [1.1] and Section 1.4 herein.

4.3 Section 4 References

4.1. NuScale Topical Report, "Nuclear Analysis Codes and Methods Qualification," TR-0616-48793-NP-A Rev 1, December 2018. **NRC Accession Number: ML18348B036**.

4.2. Letter, A.H. Bradford (NRC) to Z.W. Rad (NuScale), "STANDARD DESIGN APPROVAL FOR THE NUSCALE POWER PLANT BASED ON THE NUSCALE STANDARD PLANT DESIGN CERTIFICATION APPLICATION," September 11, 2020. **NRC Accession Number: ML20247J564**.

4.3. R. Skarda, et al., "Validation of Polaris and PARCS against Otto Hahn Second Core Zero Power Experiment Benchmark," Proc. Int. Meeting PHYSOR 2016, Sun Valley, ID May 1-5, 2016.

5. Summary and Conclusions

The analysis presented here demonstrates that there are no methodological changes needed in CMS5 to support an enrichment range of up to 10 wt% and higher burnup up to 80 GWd/MTU. This is demonstrated by executing comparisons for suitable critical experiments and higher-order reference solutions.

Comparisons of CASMO5 versus high-order MCNP6 Monte Carlo solutions for BOL lattices spanning various conditions indicate that there is no significant bias in eigenvalue for higher enrichments. The mean CASMO5-MCNP6 differences for soluble boron worths, integral absorber worths, discrete absorber worths, IFBA worths and Gd worths were within 3.5% of the calculated values from MCNP6.

Critical experiments LCT-22 through LCT-25 demonstrate that CASMO5 can satisfactorily predict the reactivity for cores with fuel of 7.5% to 10% enrichment. Furthermore, the results for Otto Hahn critical experiment (LCT-81) confirm the ability of CASMO5 and SIMULATE5 to model cores involving higher enrichments and high-leakage small PWR cores. The difference between the CASMO5 predicted eigenvalue and the MCNP6 prediction is approximately 100 pcm. The CASMO5 maximum absolute and relative differences in fission rate distribution relative to MCNP6 are 0.6% and 1.2%, respectively. The difference between the SIMULATE5 predicted eigenvalue and the MCNP6 prediction is approximately 114 pcm. The SIMULATE5 predicted planar peak pin fission rates are within 2.8% absolute difference and 1.5% relative difference relative to MCNP6 predictions. The SIMULATE5 predicted assembly-wise axial power distributions match those of the reference MCNP6 predictions very well with a maximum error of less than 1%. Peak fission rate and axial power distributions exclude the bottom and the top axial planes.

Higher-order reference solutions were obtained from Serpent2 depletion calculations and used to benchmark CASMO5 results. Detailed depletion calculations are performed for a typical PWR pin cell but with higher enrichment. Furthermore, Serpent2 is used to benchmark CASMO5 by considering full 17x17 PWR assembly depletion calculations with higher enrichments. Comparisons between CASMO5 and Serpent2 indicate a difference of approximately 107 pcm (depleting up to 80 GWd/MTU) for 8% enrichment involving a pin cell and 154 pcm (depleting up to 90 GWd/MTU) for 6.3% enrichment for lattice configuration. The fission rate maximum difference for the lattice depletion remained equal or below 0.01 at selected exposure points. In conclusion, higher enrichment and high burnup poses no challenge to the methods implemented in CASMO5.

A final examination that supports the case for CMS5 high burnup involves a lead test assembly operated at the North Anna Power Station to a maximum rod-average burnup of 72.6 GWd/MTU. Four rods were harvested to obtain burnup measurements at each rod's axial mid-plane. When compared to these measurements, CMS5 calculations examining the actual operating conditions of the lead test assembly show differences less than 10% - a very reasonable result demonstrating the suitability for CMS5 accurately modeling fuel rods achieving high burnup.

In conclusion, the information presented in this TR Supplement 1 to the CMS5 Generic PWR TR justifies the following ranges and applicability statements regarding use of the CMS5 code system to:

- PWR fuel with U-235 enrichment in UO_2 up to and including 10 wt%
- PWR fuel designs with maximum rod-average fuel burnup up to and including 80 GWd/MTU
- Light water pressurized Small Modular Reactors (SMRs)