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A SAS2H/KENO-V METHODOLOGY FOR 3D FULL CORE DEPLETION ANALYSIS

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ABSTRACT

This paper describes the use of a SAS2H/KENO-V methodology for 3D full core depletion analysis and illustrates its capabilities by applying it to burnup analysis of the IRIS core benchmarks. This new SAS2H/KENO-V sequence combines a 3D Monte Carlo full core calculation of node power distribution and a 1D Wigner-Seitz equivalent cell transport method for independent depletion calculation of each of the nodes. This approach reduces by more than an order of magnitude the time required for getting comparable results using the MOCUP code system. The SAS2H/KENO-V results for the asymmetric IRIS core benchmark are in good agreement with the results of the ALPHA/PHOENIX/ANC code system.

Key Words: 3D full core depletion analysis, SAS2H/KENO-V sequence, IRIS

1. INTRODUCTION

The MOCUP code developed by INEEL [1] has been in use at UC Berkeley (UCB) for full core burnup analysis of fast reactors for some time. MOCUP consists of the MCNP-4C Monte Carlo code [2] and of ORIGEN2.1 depletion code [3] managed by the MOCUP driver [1]. This tool, although very accurate, was found not to be adequate for depletion analysis of relatively large thermal core designs due to limitations described below. The specific LWR core design we tried to apply MOCUP to is IRIS – a modular, small/medium size (335 MWe) PWR with integral vessel configuration [4,5]. The reference IRIS core is to operate for up to four years without fuel shuffling or partial reloading; it makes enhanced use of ^{10}B burnable absorbers in the form of IFBA, a thin layer of ZrB_2 coating fuel pellets. Using erbium as integral fuel absorber is also considered. The initial space-dependent depletion analysis of IRIS using MOCUP was proven to be unsatisfactory due to: (a) too long running time, (b) a limited number (50) of zones for depletion analysis, (c) a limited number (150) of nuclides for which one-group cross sections for depletion analysis could be determined, and (d) a possible problem with the convergence of the spatial distribution of fission neutron sources, unless a large number of generations with a large number (10^5) of neutron histories per generation is used, prohibitively extending the running time.

In order to remove the above-mentioned limitations and at the same time to retain accuracy, we developed a new methodology that involves several modules of the SCALE-4.4a code system [6]. The multigroup Monte Carlo code KENO V.a [7] is used to generate the power distribution per node, and the SAS2H code sequence [8] is used to perform space-independent depletion analysis for each of the core nodes in the model.

The new methodology was first verified by comparing its results for an IRIS 15x15 fuel assembly burnup analysis against results of two well-benchmarked code systems [18]. The first is MOCUP, which we previously verified for thermal reactor applications [9]. The second code system is KENO V.a/ORIGEN2 recently developed at UCB [10]; it is similar to MOCUP except that it uses the KENO V.a code instead of MCNP-4C. It enables including in the fuel depletion analysis all fission products for which cross section data exist in the SCALE multi-group cross section libraries. It has been found that the latter code system is as accurate as MOCUP and is faster than MOCUP by more than an order of magnitude. The newly developed SAS2H/KENO-V methodology was then applied to the 3-D burnup analysis of two configurations of the IRIS PWR reactor core with IFBA (Benchmark#44 core [11]) that features different burnable poison distributions; one configuration uses axially symmetric [18] and the other uses axially asymmetric IFBA distribution that will be described in this paper.

A short description of the IRIS benchmark core configuration is presented in Section 2. The new SAS2H/KENO-V methodology is described in Section 3. The results and the discussion of the 3-D burnup analysis of the IRIS core with non-uniform axial boron distribution are presented in Section 4. The comparison with the deterministic code system ALPHA/PHOENIX/ANC is presented in Section 5.

2. THE IRIS BENCHMARK CORE CONFIGURATION

The IRIS 1000 MWt PWR core with IFBA (Benchmark#44 core configuration [11]) consists of 89 fuel assemblies arranged as shown in Figure 1. The core is surrounded by a reflector consisting of 50% stainless steel and 50% water. An IRIS core fuel assembly is composed of 15x15 square unit cells 204 of which are fuel cells and the remaining 21 contain guide thimbles (GT) or an instrumentation tube (IT). The GT and IT unit cells are assumed identical for these calculations. The dimensions and other specifications of the fuel assemblies are defined in [5].

The reference fuel assemblies (FAs) contain UO_2 with uranium enrichment of 4.95 w/o ^{235}U. Peripheral fuel assemblies contain UO_2 with enrichment of 2.6 w/o ^{235}U. Axial blankets at the top and bottom are 30.48 cm (1 ft) long and made of UO_2 with enrichment of 4.95 w/o ^{235}U. Most FAs have symmetric IFBA coating (IFBA coating is expressed in mg ^{10}B per cm of fuel rod). IFBA coating covers the central 355.76 cm (12 ft) of the fuel, i.e., a fuel rod is axially composed of: 30.48 cm (1 ft) enriched (4.95 w/o ^{235}U) uncoated, 355.76 cm (12 ft) enriched (4.95 w/o ^{235}U) coated, and 30.48 cm (1 ft) enriched (4.95 w/o ^{235}U) uncoated UO_2. The fuel assembly is surrounded on each side by a 0.0508 cm (0.020") water gap, hence, the fuel assembly pitch is 22.66442 cm (8.923"). The pin cells are square; their outer fuel radius is 0.46482 cm, the outer gap radius is 0.47371 cm, the outer Zircaloy-4 clad radius is 0.53721 cm and the cell pitch is 1.50419 cm. Grid plates, assumed to occupy 1 volume % of the moderator, are homogenized with the moderator.

Several boron loading patterns were assumed; each pattern corresponds to a different benchmark problem. Figure 2 gives a vertical cut through a core with a non-uniform boron distribution.

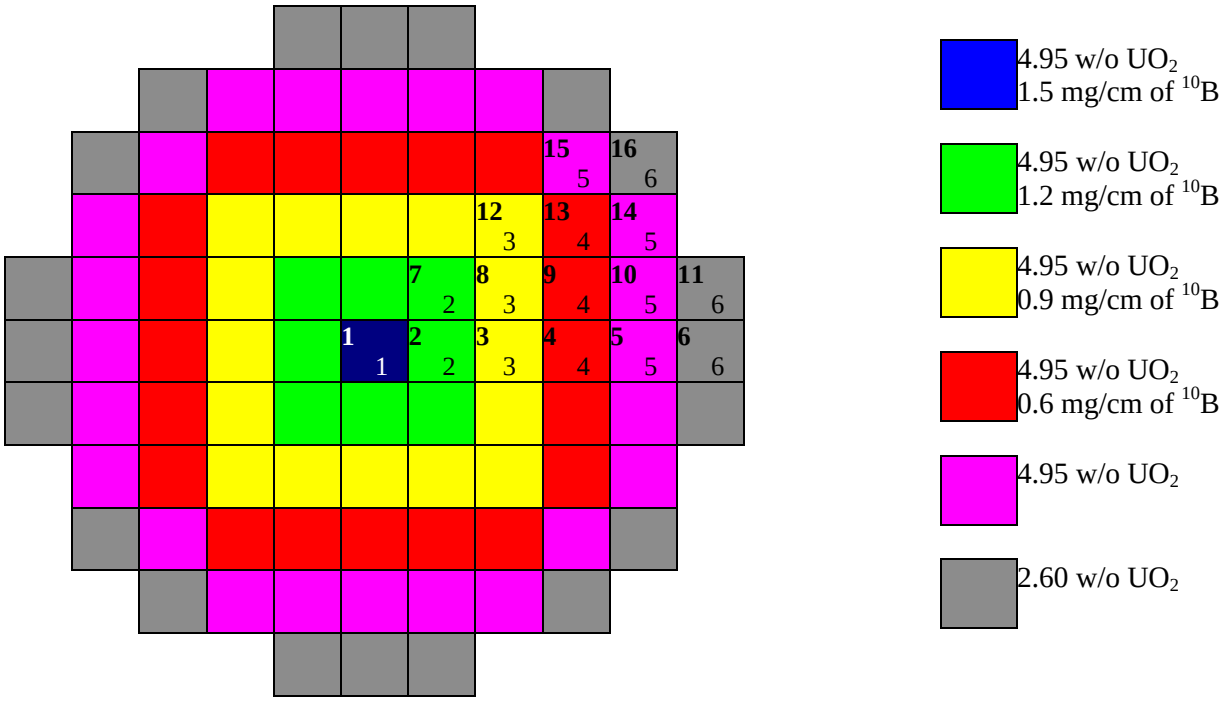


Figure 1. Horizontal cut through the IRIS Benchmark#44 core

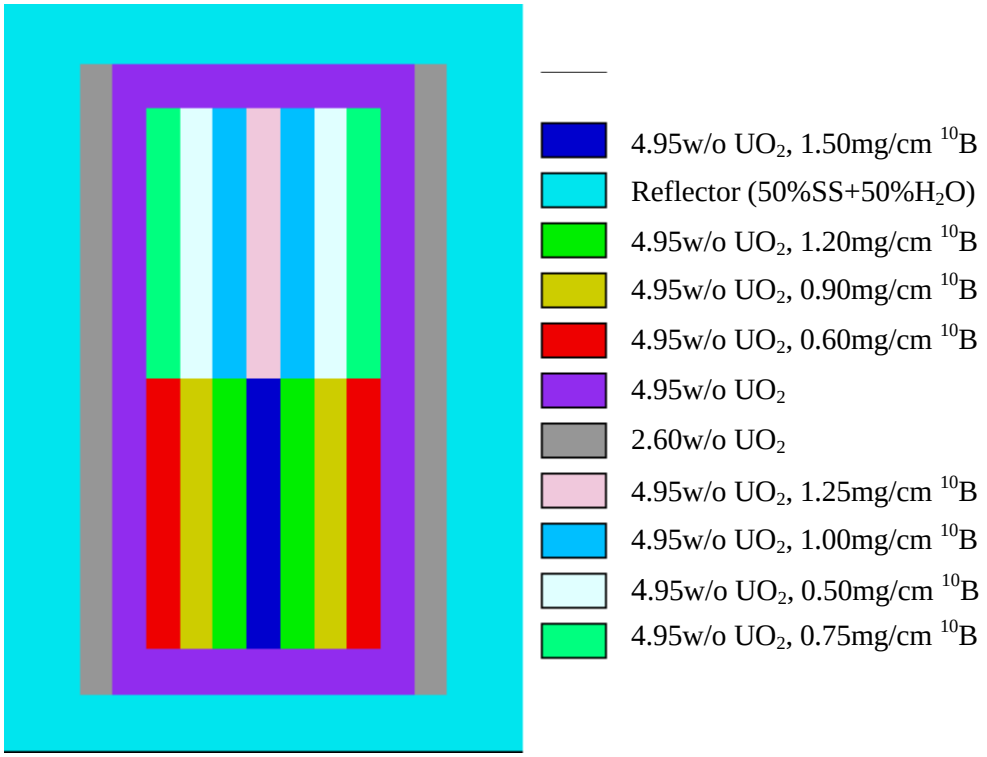


Figure 2. Vertical cut of Benchmark#44 with non-uniform axial boron distribution

The UO_2 is assumed to be at a uniform temperature of 810.96 K and have a density of 10.4164 g/cm^3. Zircalloy-4 is used for the cladding and structural material. It is assumed to be at a uniform temperature of 616.46 K, have a density of 6.56 g/cm^3 and composed of (in weight percent): 98.23% Zr, 1.45% Sn, 0.10% Cr; 0.21% Fe and 0.01% Hf. The moderator is composed of water (99 volume %; 0.703 g/cm^3) and Zircalloy-4 (1 volume %, 6.56 g/cm^3) at a uniform temperature of 584.16 K. All components outside of the active core are represented by a homogenized reflector having an effective radial and axial thickness of 40 cm. Included in the reflector are: the fission gap plenum, top and bottom core support plates, core baffle and so on. The homogenized reflector composition consists of 50% (by volume) H_2O and 50% stainless steel SS304. It is assumed that the average linear power is 129.07 W/cm corresponding to an average specific power of 20.7107 MW/tHM and to a total core power of 1000 MWt.

3. THE NEW SAS2H/KENO-V METHODOLOGY

The proposed methodology for a 3D full core fuel depletion analysis of LWR reactors makes a novel use of several SCALE-4.4a (or SCALE-4.3) modules and sequences. This methodology is based on three approximations/assumptions: First, it is assumed that modeling of the space-dependent fuel depletion in a full 3D core can be represented by two independent calculations. One calculation determines the power distribution in large volumes called nodes, and the other calculates the burnup-dependent node (fuel) composition. This assumption is commonly done in core depletion analysis. For the first calculation, the Criticality Safety Analysis Sequences (CSASI) [12] is used to provide a Monte Carlo-formatted cross-section library for each of the nodes, and the WAX module [13] is used to combine these libraries into one working library. The KENO-V.a code uses the processed multigroup cross sections and calculates the effective multiplication factor (k_{eff}), neutron flux and fission density of three-dimensional system models. The geometric modeling capabilities available in KENO-V.a coupled with the automated cross-section processing within the control sequences allow complex, 3D systems to be easily analyzed.

The second and third approximations are related to the modeling of fuel depletion. The second approximation assumes that the time-dependent fuel composition in the fuel assembly can be predicted accurately enough by using the SAS2H control sequence for one-dimensional (1D) fuel depletion analysis. In this sequence, 1D neutron transport analysis of the reactor fuel assembly is performed with the XSDRNPM code [14] using a simple lattice-cell model, which is an infinite Wigner-Seitz pin cell having radii that preserve the volumes of each material in the fuel assembly. The fuel neutron flux spectrum obtained from this lattice-cell calculation is used to generate the effective one group cross sections for the fuel isotopes that are needed for burnup analysis. These cross sections are processed by the COUPLE module [15] and used by the ORIGEN-S code [16] to perform point-depletion computation. The outcome is burnup-dependent fuel composition that is to be used in the next KENO-V.a calculation. For reasons of calculation accuracy, SAS2H, CSASI and KENO-V.a use a 44-group library [17] that is based on ENDF/B-V. The third approximation, which is related to the IRIS core design, involves smearing the boron layer with the clad (using volume homogenization) for the fuel depletion analysis. This approximation is used only for the SAS2H calculation in order to overcome numerical difficulties associated with modeling the effect of a very thin layer of a strong absorber in a discrete ordinates neutron transport code XDRNPM. In the full core modeling using KENO-V.a, each fuel cell is accounted for explicitly and the boron layers are described explicitly. The reflector regions are also treated rigorously in the KENO-V.a calculations.

The presence of ^{10}B depletion and xenon build-up and redistribution in the IRIS Benchmark#44 cores dictates use of small burnup steps between the flux recalculation during the fuel depletion analysis. For the case with non-uniform axial boron distribution, we used even smaller time steps then in the uniform case analyzed in [18]. The following predictor-corrector procedure was found to be effective for reducing the flux calculation frequency and adequately accounting for the coupling between the spatial power distribution and the depletion of fuel and burnable absorber. For each time step, the predictor procedure determines the distribution of fission power in each volumetric node for the initial burnup step B1 (the first KENO-V.a run). In order to reduce the statistical uncertainties and possible problems in the convergence of the spatial distribution of fission neutrons in each neutron generation, we use arithmetic average of this power distribution and the power distribution calculated during the corrector step in the previous burnup step. This averaged power distribution is used for burnup calculation (using SAS2H) to burnup level B2. The corrector step is then used to determine the final distribution of fission power for depletion step B2 (the second KENO-V.a run), and this new power distribution is used to do the repeated depletion analysis from the initial (B1) to the final (B2) step (via SAS2H). The final fuel isotopics and burnable absorber concentrations in each node at the final (B2) step are determined by averaging the concentrations obtained in the predictor and corrector steps. The above procedure was verified [18] for a 2D X-Y model of the IRIS 15x15 fuel assembly (infinitely long in the Z-direction with reflective boundary condition) against two well benchmarked code systems: MOCUP [3] and the KENO-V.a/ORIGEN2.1 code system recently developed at UCB [10].

4. FULL CORE ANALYSIS

Criticality calculations using MCNP-4C and KENO-V.a can suffer from two potential problems. The first is the failure to converge the spatial distribution of the fission source from its initial guess to a distribution fluctuating around the fundamental eigenfunction solution. The second potential problem arises from the fact that the criticality algorithm produces a very small negative bias in the estimated eigenvalue. The bias depends upon 1/N, where N is the number of source neutrons per generation. Thus, it is desirable to make N as large as possible. In the current 3D calculation of Benchmark#44 the following strategy for KENO-V.a calculations was chosen: (a) at least 50000-100000 histories per generation are used, (b) at least 150 of the initial generations are discarded (since the problem is loosely-coupled), (c) at least 600 active generations are run, and (d) 95% confidence intervals (2σ), not 1σ, are used. After all, it is expected that the true answer lies outside of $\pm 1\sigma$ 33% of the time, but only 5% of the time for 2σ confidence intervals.

The fuel depletion analysis for Benchmark#44 cores is performed assuming a core average linear power of 129.07 W/cm (20.7107 MW/tHM); it corresponds to a total power of 1000 MWt. The initial core composition is shown in Figure 1. The 1/8th of the core is divided into 16 radial nodes (Fig. 1), and 14 equal-height axial nodes. KENO calculations were done using the following burnup increments in MWd/tHM: 10, 15, 25, 100, and 500 for the first, second, third, fourth, and fifth burnup step, respectively; 1000 from 650 MWd/tHM to 38650 MWd/tHM (EOC). Typical CPU time for chosen parameters and for Benchmark#44 core with 16(radial)x14(axial) nodes on a 2 GHz Pentium IV PC with Windows 2000P is 30 days for 100,000 source neutrons per generation. In order to reduce the time to about 15 days, we selected and used 95 fission products instead of 186 FPs that we used up to now. Our analysis has shown that it did not affect the accuracy, but it did reduce the time considerably.

Figures 3 through 11 illustrate results from a full core simulation of IRIS Benchmark#44 having 89 fuel assemblies and non-uniform axial distribution of ^{10}B, obtained by SCALE-4.4a code system using 16(radial)x14(axial) burnup nodes. The initial composition of this core is defined in Figure 1. Figure 3 shows the evolution of k_{eff} whereas Figure 4 shows the evolution of the ratio of peak-to-average core power. Illustration of the variation in the core average radial and axial power distribution in nodes is given in Figures 5 to 11. It is observed that the power density distribution varies widely both radially and axially.

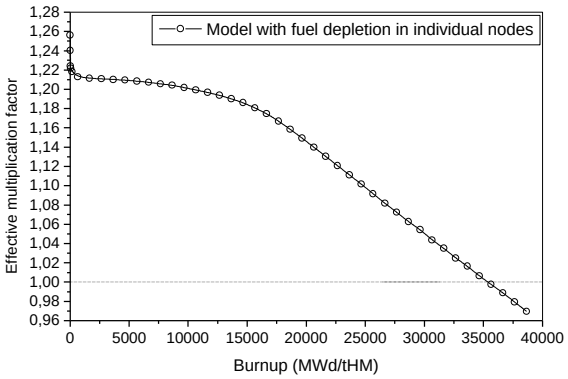


Figure 3. k_{eff} evolution with burnup

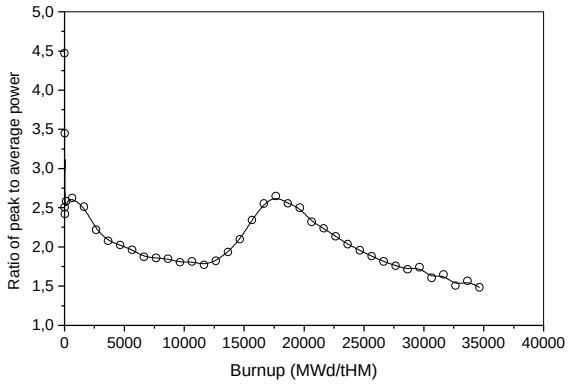
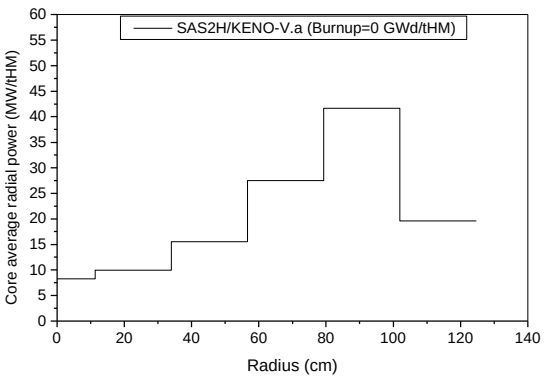


Figure 4. Evolution, with burnup, of core peak to average power density ratio



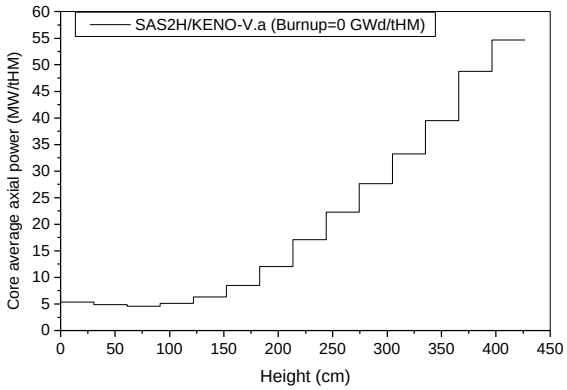


Figure 5. Radial and axial power distribution at 0 GWd/tHM

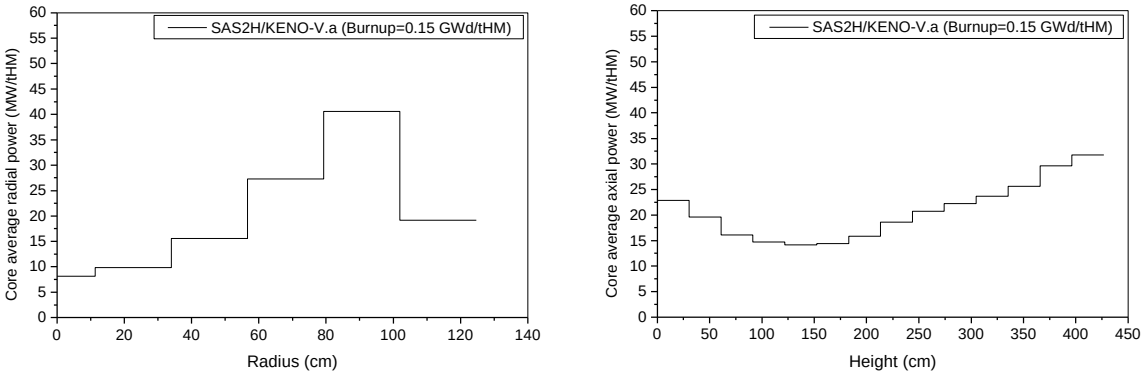


Figure 6. Radial and axial power distribution at 0.15 GWd/tHM

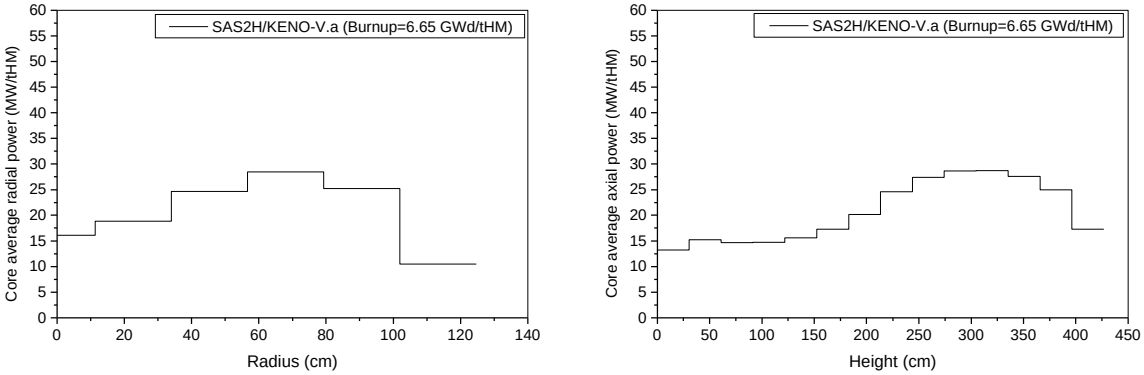


Figure 7. Radial and axial power distribution at 6.65 GWd/tHM

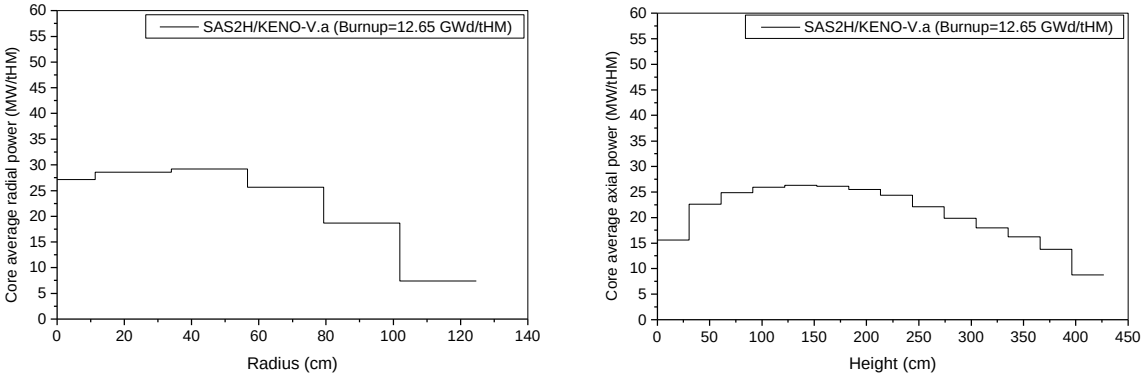


Figure 8. Radial and axial power distribution at 12.65 GWd/tHM

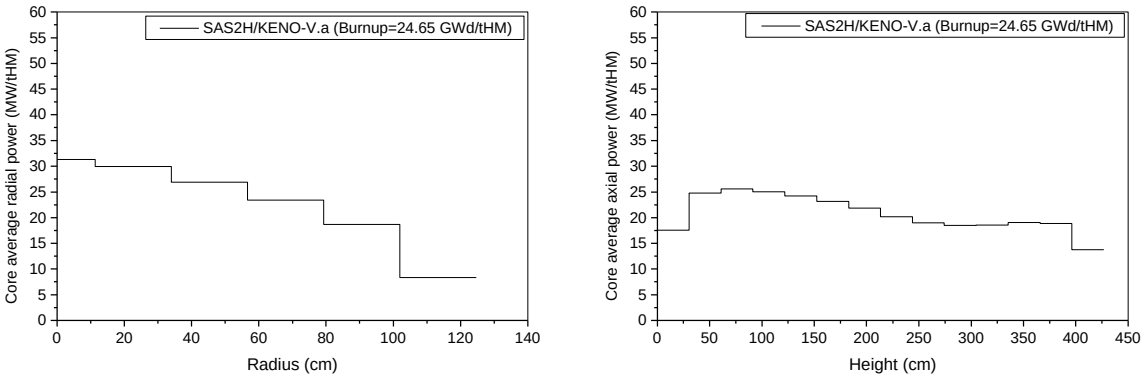


Figure 9. Radial and axial power distribution at 24.65 GWd/tHM

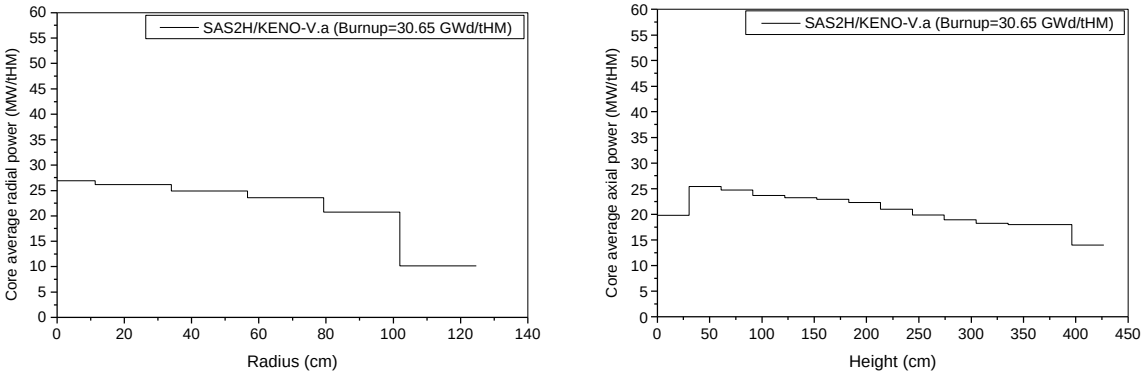


Figure 10. Radial and axial power distribution at 30.65 GWd/tHM

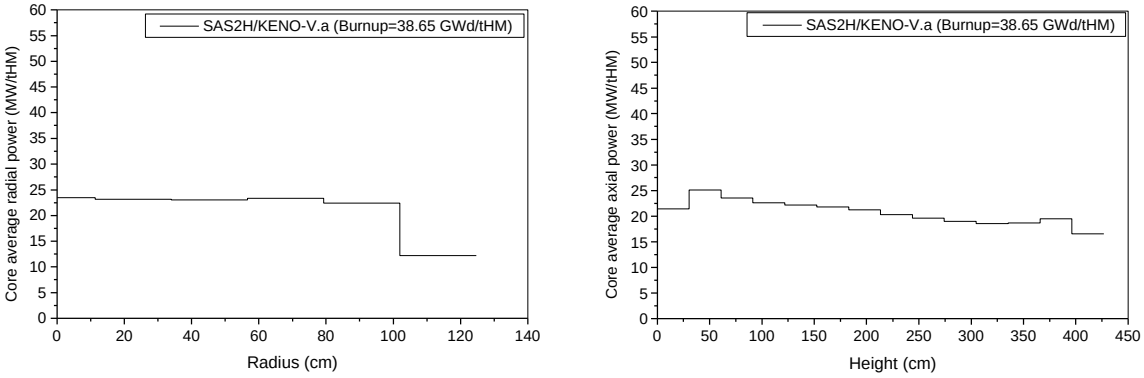


Figure 11. Radial and axial power distribution at 38.65 GWd/tHM

5. COMPARISON WITH PHOENIX/ANC

In order to assess the feasibility/practicality of the SAS2H/KENO-Va methodology for full core modeling, we compared it with the well-established Westinghouse ALPHA/PHOENIX/ANC deterministic code system [19]. In analyzing the IRIS core benchmark, it was not always possible/practical to use exactly identical assumptions. Therefore, some differences in the results are expected. We should emphasize that the objective of this comparison was not to investigate and resolve these small differences, but to establish that with reasonably similar assumptions we can obtain reasonably close results. The reasons for the differences include the following: (1) non-standard use of the Westinghouse codes, (2) slightly different reflector representation, (3) different treatment of the Doppler feedback (the fuel temperature feedback has been completely removed in SAS2H/KENO-Va, but not completely eliminated in ANC), (4) somewhat different treatment of fission products close to BOC, (5) burnup and change in fuel isotopics is followed on a different axial mesh, (6) statistical noise in MC results, combined with a possibility that the solution is still not completely converged to fundamental mode, (7) different cross section libraries, including B-V and B-VI based data, (8) possibly some differences in material compositions, densities (different modeling of expansion), grid volume, and temperatures. It should also be noted that the Benchmark#44 core configuration does not necessarily represent an actual (acceptable) core configuration and that some non-physical conditions are imposed (e.g., some feedback effects are intentionally removed).

Figure 12 compares k_{eff}. The solid line represents PHOENIX/ANC, the dashed line is SAS2H/KENO-Va. The difference is acceptable, almost constant over the whole range, and primarily due to different assumptions. Figure 13 compares power peaking factors as a function of burnup; the behavior is very similar. SAS2H/KENO-Va results are always slightly higher, typically about 6 percent, with additional several percent variability. This average difference is consistent with the difference in applied Doppler feedback, while the variability is partly due to MC statistics. The 2σ uncertainty in SAS2H/KENO-Va results at the maximum power density location is typically ~2%, and higher elsewhere, contributing some of the variability in the observed differences. Overall, a very good agreement in power peaking factors is observed.

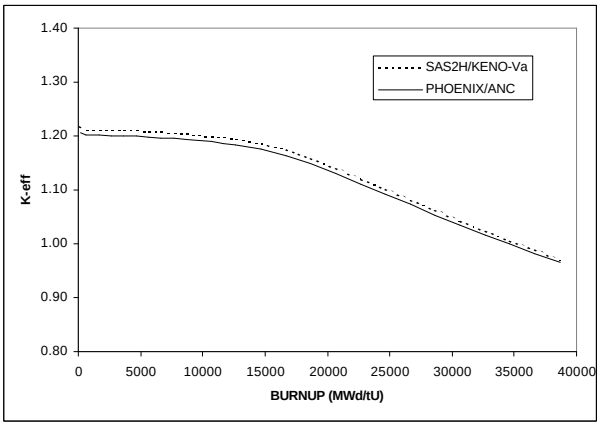


Figure 12. k_{eff} evolution with burnup

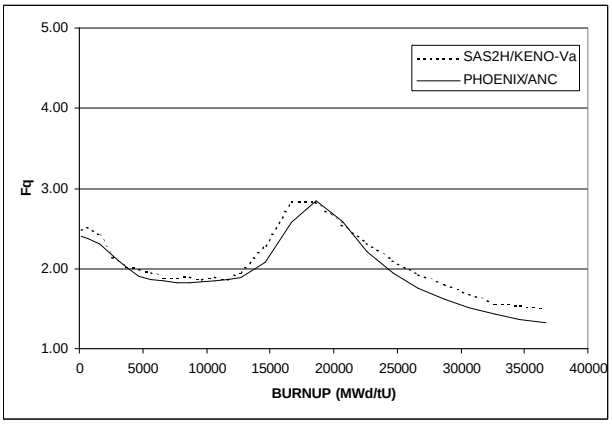
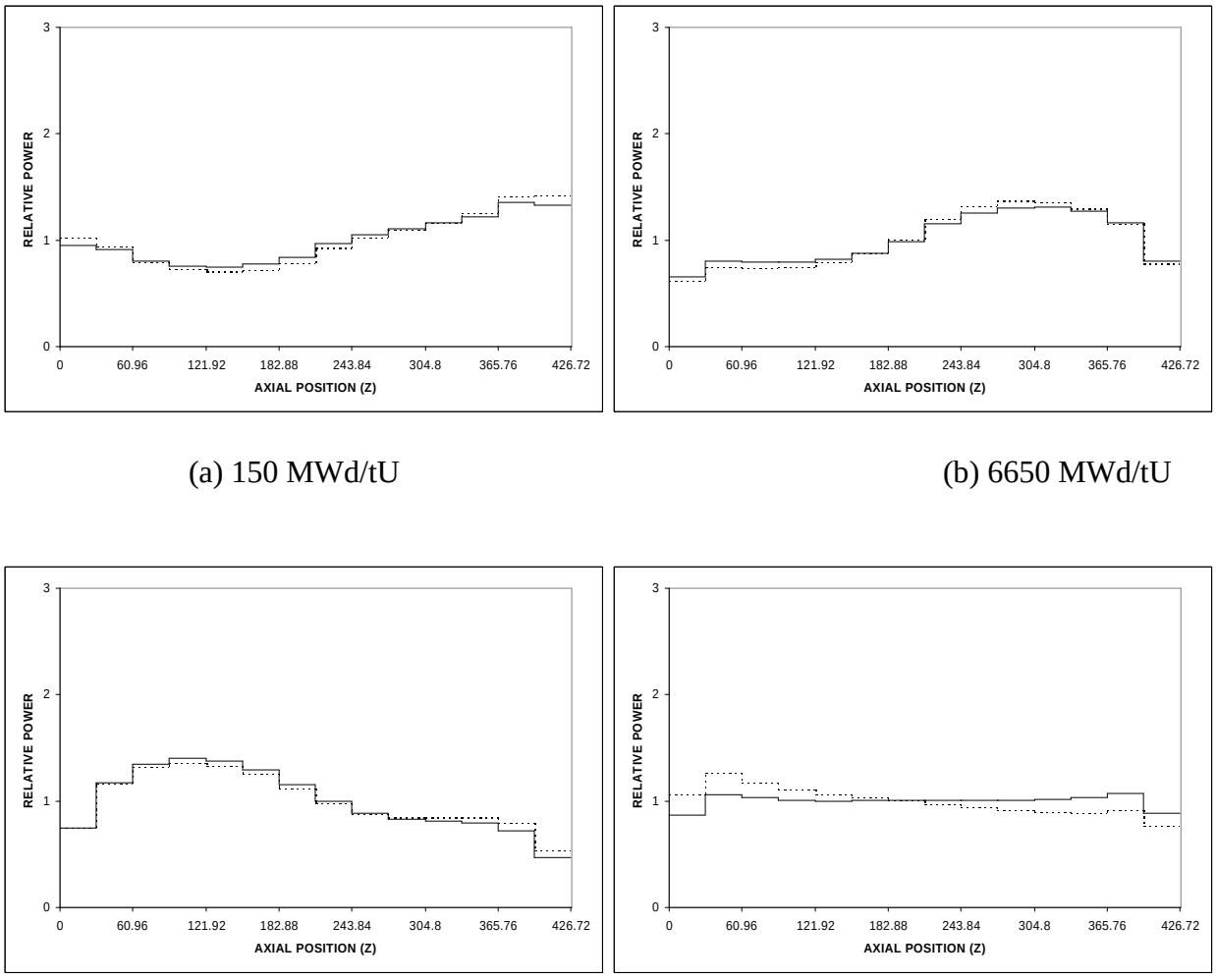


Figure 13. Evolution, with burnup, of core peak to average power density ratio

Figure 14 compares axial power distributions at 150, 6650, 18650, and 36650 MWd/tHM, where the PHOENIX/ANC results have been projected to the SAS2H/KENO-Va axial mesh. The observed agreement is very good for the main part of the cycle. The difference is somewhat higher at EOC; this is tentatively explained by the differences in modeling, i.e., different axial mesh used by the two code systems. Consequently, fuel isotopics is followed for axial zones of different length, this difference builds up with fuel depletion, and for the very long cycle considered here it eventually impacts the axial power shape.



(a) 150 MWd/tU (b) 6650 MWd/tU

(c) 18650 MWd/tU (d) 36650 MWd/tU

Figure 14 Axial power profile at representative core burnups

Figure 15 presents percent differences in node-wise radial power distribution at BOC and EOC. An excellent agreement is observed, i.e., all differences are within 1.3% at BOC and within 2.5% at EOC.

-0.5%	0.0%	0.4%	-0.1%	-0.7%	0.2%
0.0%	0.4%	0.7%	0.2%	-0.5%	-1.3%
0.4%	0.7%	1.3%	1.3%	-1.2%	
-0.1%	0.2%	1.3%	1.2%	-0.4%	
-0.7%	-0.5%	-1.2%	-0.4%		
0.2%	-1.3%				

(a) 150 MWd/tU

1.8%	2.1%	1.5%	0.4%	-1.3%	0.0%
2.1%	1.9%	1.4%	0.3%	-1.2%	-1.4%
1.5%	1.4%	1.0%	0.0%	-2.5%	
0.4%	0.3%	0.0%	-0.7%	-1.3%	
-1.3%	-1.2%	-2.5%	-1.3%		
0.0%	-1.4%				

(b) 36650 MWd/tU

Figure 15 Percent difference in radial node-wise power distribution at BOC and EOC

In summary, a very good agreement of the two code system results has been observed. Moreover, the existing differences may be attributed in large part to different modeling assumptions.

5. CONCLUSIONS

The newly developed SAS2H/KENO-V.a methodology was found adequate for 3D burnup analysis of full LWR cores. The new methodology can predict with good accuracy not only k_{eff}, but also the space dependent nuclide composition, burnup and power distribution. The prediction of accurate burnup and power distribution in relatively large thermal reactor cores like the IRIS core requires use of a predictor-corrector algorithm and a large number of source neutrons and is therefore time consuming – on the order of one to two weeks on a single 2 GHz Pentium IV processor. However, this is more than an order of magnitude shorter than the time required for getting comparable results using the MOCUP code system. A very good agreement was obtained when compared with the ALPHA/PHOENIX/ANC code system.

ACKNOWLEDGMENTS

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