BURNUP CREDIT VALIDATION OF SCALE-4 USING LIGHT-WATER-REACTOR CRITICALS*

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INTRODUCTION

The ANSI/ANS 8.1 criticality safety standard[1] recommends validation and benchmarking of the calculational methods used in evaluating criticality safety limits for away-from-reactor applications. The lack of critical experiments with burned light-water-reactor (LWR) fuel in racks or in casks necessitates the validation of burnup credit methods by comparison with LWR core criticals. These benchmarks are relevant because they test a methodology's ability to predict spent fuel isotopics and to evaluate the reactivity effects of heterogeneities and strong absorbers. Data are available to perform analyses at precise state points.

As part of the Burnup Credit Analysis Verification (BCAV) Task, the U.S. Department of Energy has sponsored analysis of selected reactor core critical configurations from commercial pressurized-water-reactors (PWRs).[2] The initial analysis methodology used the SCALE-4 code system[3] to analyze a set of reactor critical configurations from Virginia Power's Surry and North Anna reactors.[4,5] However, the analysis procedure was complex and included the calculation of lumped fission products. The methodology has since been revised to simplify both the data requirements and the calculational procedure for the criticality analyst. This revised methodology is validated here by a comparison with three reactor critical configurations from Tennessee Valley Authority's Sequoyah Unit 2 Cycle 3 and two from Virginia Power's Surry Unit 1 Cycle 2.

Sequoyah Unit 2 Cycle 3 was chosen because the unit had a significantly long downtime of 2.7 years during the middle-of-cycle (MOC) 3, and no fresh fuel was loaded into the reactor prior to restart. The first benchmark critical calculation was the MOC restart at hot-full-power (HFP) critical conditions. The other two benchmark critical calculations were the beginning-of-cycle (BOC) startup at both hot-zero-power (HZP) and HFP critical conditions. These BOC conditions were used to check for consistency in the calculated results for different burnup, downtime, temperature, and xenon conditions.

The Surry cases were originally selected[4,5] because of the availability of the operating data and the three-dimensional (3-D) calculated nuclide density distributions from the utility's reactor physics calculations. Two critical calculations at BOC startup at HZP and end-of-cycle (EOC) operation at HFP are reanalyzed here to provide additional validation for the revised SCALE-4 methodology.

SAS2H ISOTOPIC CALCULATIONS

The SAS2H/ORIGEN-S depletion sequence[6] within SCALE-4 was used to predict isotopic concentrations for each fuel enrichment and assembly type [i.e., fresh burnable poison rods (BPR), depleted BPR, or no BPR], referred to as a fuel batch. A total of seven SAS2H batches each were required for Sequoyah and Surry. It was assumed that each assembly in a batch had the same operating power. The average operating power history and the average burnup of each batch were computed from individual assembly burnup data. The uptimes and downtimes were determined directly from the burnup intervals, the average powers, and the operating history data. Isotopic inventories were computed by SAS2H at burnup intervals of 3 to 5 MWd/kgU. Each SAS2H batch was depleted to 1.2 times the maximum assembly burnup, thus allowing interpolation of isotopic data on assembly burnup.

KENO V.a MODEL SETUP

Sequoyah Unit 2 contains 193 Westinghouse 17 × 17 fuel assemblies and Surry Unit 1 contains 157 Westinghouse 15 × 15 fuel assemblies. These fuel assemblies are generally loaded in an eighth-core symmetric pattern. In order to simplify and reduce the volume of input in the KENO V.a model, eighth-core symmetry was assumed in the isotopic input. This reduces the number of unique fuel assemblies to 31 for Sequoyah and 26 for Surry. No axial dependence of the isotopics or cross sections is modeled. A detailed description of the models are given in ref. 7.

The spent fuel isotopics were derived from the SAS2H calculations described previously. Isotopics were calculated for each of the eighth-core fuel assembly locations using a newly developed module called SNIKR to interpolate on burnup. ORIGEN-S was then used to decay the interpolated nuclide concentrations for the appropriate downtime.

The eighth-core fuel assembly locations were grouped into cross-section sets for calculation of the microscopic cross sections. The cross-section sets were created by dividing the fuel assemblies in each fuel batch into subgroups based on their burnups. Cross sections were calculated for each set of fuel assemblies with similar burnups. Average nuclide concentrations for each cross-section set were input to the CSASN analytical sequence that contains BONAMI and NITAWL-II to perform resonance processing of the microscopic cross sections. The 27BURNUPLIB cross-section library in SCALE was used. This library contains predominantly ENDF/B-IV data but has pre-released ENDF/B-V data for the fission products and higher-order actinides.

KENO V.a RESULTS

KENO V.a was used to model each reactor in full-core geometry. The criticality calculations were performed with 1000 neutrons per generation for a minimum of 303 generations. The calculated k_{eff} values in Table I are consistent and are in good agreement with the measured critical condition (k_{eff} = 1.0). The mean k_{eff} of these calculations was 1.0029. Therefore, we conclude that the revised SCALE-4 methodology can accurately calculate reactivity for spent PWR fuel assemblies. The Surry results from the validation of the initial methodology are included for comparison because they were previously published as part of the BCAV task. Note that the result for the Surry BOC case is approximately 2% $\Delta k/k$ greater than that reported for the original Surry reactor critical calculation performed with the initial SCALE methodology.[5] Differences in the earlier analysis that have been identified and their probable order of importance are: use of a lumped fission product

Table I
KENO V.a calculated results

Reactor	Burnup	Power	Boron (ppm)	New methodology k_{eff}	Old methodology k_{eff} from ref. 5
Sequoyah	BOC	HZP	1685	1.0006 ± 0.0009	---
Sequoyah	BOC	HFP	1150	1.0026 ± 0.0009	---
Sequoyah	MOC	HFP	475	1.0001 ± 0.0010	---
Surry	BOC	HZP	1030	0.9939 ± 0.0008	0.9757 ± 0.0022
Surry	EOC	HFP	123	1.0034 ± 0.0007	0.9960 ± 0.0020

to account for all nuclides not explicitly modeled; fewer fuel batches and cross-section sets; fewer neutrons per generation and fewer total neutron histories; and use of an earlier version of SAS2H.

The consistency in results for the new methodology and its straightforward procedure for calculating isotopics provide a high level of confidence in these results. The complexity in the isotopic calculational procedure of the initial methodology, the use of lumped fission products, and the inability to reproduce results due to loss of data from the earlier work reduces our confidence in the old methodology.

The results for these reactor criticals with spent fuel that consist of low-enriched uranium and plutonium oxides also agree well with calculations that analyzed 12 critical experiments consisting of mixed-oxide fuel rods in square lattice geometries.[8] From the consistency of results for the new methodology in Table I and the agreement with the work in ref. 8, we conclude that the SCALE code system provides a valid analysis methodology for burnup credit applications.

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