

Validation of a Monte Carlo Based Depletion Methodology Using HFIR Post-Irradiation Measurements

David Chandler¹, R. T. Primm, III², and G. Ivan Maldonado¹

¹*Nuclear Engineering Department
The University of Tennessee
Knoxville, Tennessee 37996-2300*

²*Research Reactors Division, Oak Ridge National Laboratory
P. O. Box 2008; Oak Ridge, TN 37831-6399*

INTRODUCTION

Post-irradiation uranium isotopic atomic densities within the core of the High Flux Isotope Reactor (HFIR) were calculated and compared to uranium mass spectrographic data measured in the late 1960s and early 70s [1]. This study was performed in order to validate a Monte Carlo based depletion methodology for calculating the burn-up dependent nuclide inventory, specifically the post-irradiation uranium isotopic composition, in the core interior of HFIR.

The HFIR, located at the Oak Ridge National Laboratory, is a multipurpose research reactor that currently operates at 85 MW (rated at 100) and provides a peak thermal neutron flux of 2.6×10^{15} neutrons/cm²-sec. It is a beryllium-reflected, light water-cooled and – moderated, pressurized flux-trap type reactor that utilizes highly enriched uranium (~93 wt. % ²³⁵U) fuel in a mixture of U₃O₈-Al that is encapsulated within Al-6061 plates. The core consists of two concentric annular elements: an inner fuel element (IFE) and an outer fuel element (OFE), constructed of 171 and 369 involute fuel plates, respectively. Two concentric poison-bearing control elements and boron non-uniformly distributed along the arc of the IFE plates control reactivity.

The experiments analyzed were performed to examine spent fuel elements and to ensure no unexpected problems existed that would compromise continued operations. Three plates were studied: two from the OFE and one from the IFE. Fuel plates from elements 5-O (cycle 4, Sept. 1966), 21-O (cycle 16, Dec. 1967 - Jan. 1968), and 49-I (cycle 35, May - Jun. 1969) were irradiated for 2046, 2309, and 2319 MWd, respectively. Data from Ref. 1 along with a detailed description of the calculations performed were reported in Ref. 2.

METHODOLOGY

ALEPH [3], a Monte Carlo based depletion code developed at SCK-CEN in Belgium that links MCNP, NJOY 99.90, and ORIGEN 2.2 together, was utilized. MCNP-V [4], and continuous energy neutron ENDF/B-VI.8 [5] and JENDL 3.3 [6] cross section data were used. An ALEPH/MCNP model that replicates the reactor components, operational history, and conditions present

during the plate irradiations was developed by modifying an existing model of a recent core configuration [7].

The model of the flux trap was amended by inserting 30, 7.65 gram ²⁴²Pu targets in the 30 interior positions, aluminum capsules in the hydraulic tube (which was moved back to the center of the flux trap), and aluminum rods in the 6 peripheral target positions. Also, four removable beryllium facilities were removed from the model, horizontal beam tubes 2 and 4 were modified to replicate their original configurations, and engineering slant facilities 3 and 4 were reinserted in the reflector.

The MCNP representation of the fuel elements, homogenized concentric radial regions of varying fuel, clad, burnable poison, and moderator concentrations, was adjusted to incorporate the locations of the specimens that were submitted to mass spectrographic analysis. In the experiments, axially, two nominally minimum burn-up regions, two nominally intermediate burn-up regions, and the nominally maximum burn-up region were selected, and radially, the inner, outer, and central regions of the fuel plates were selected for the specimens' locations. Up to 24 axial zones were used to model the elements with the specimens being modeled as 0.5 cm in height.

Transformation cards were utilized to simulate the control element withdrawal as a function of burn-up. The withdrawal curves were obtained from operator control logs and were discretized such that the positions from one burn-up step to the next were kept under 1 cm. Tallies were used to calculate the spectra required by ORIGEN and 350 generations (50 being skipped) with 100,000 histories per generation were utilized in the MCNP kcode input.

The burn-up history was specified with multiple constant power (100 MW) irradiations. An ALEPH fractional absorption criterion of 0.9999 was utilized and 313 nuclides were specified for which reaction rates were to be calculated. The material temperatures were set to 300 K, which is slightly less than the fuel temperature range (340 – 380 K). The difference between the modeled and actual spatially dependent temperatures has been shown to be inconsequential. A total of 43,000 energy groups were used: 1,000 for every order of magnitude except in the resonance regions.

Following the completion of ALEPH, the ²³⁴U, ²³⁵U, ²³⁶U, and ²³⁸U densities were extracted for each specimen,

and the atomic percentages of the four uranium isotopes were calculated and compared to the mass spectrographic results.

RESULTS

The observed deviations between the experimental and calculated results were small. No experimental uncertainties were reported in Ref. 1 and ORIGEN doesn't calculate uncertainties associated with the nuclide densities. However, if a 5% uncertainty was applied to the experimental results (a value noted in Ref. 8 as applicable to local power density determinations based on fission product gamma activity), 93% of the calculated ^{234}U atomic densities, 56% of the ^{236}U atomic densities, and 100% of the ^{235}U and ^{238}U atomic densities would fall within the experimental uncertainty range.

The C/E ratios (calculated results divided by experimental results) for all three cycles for ^{234}U , ^{235}U , ^{236}U , and ^{238}U ranged from 0.945 to 1.033, 0.976 to 1.019, 0.852 to 1.165, and 0.957 to 1.036, respectively. The differences between the ^{236}U C/E ratios and unity were the largest, but no trend was discovered. The majority of the ^{234}U C/E ratios were less than unity, and thus it is postulated that the ^{234}U capture cross section and/or the ^{235}U (n,2n) cross section may be too low. Since the fuel is highly enriched, the effect on k_{eff} is negligible.

The calculated and experimental post-irradiation ^{235}U atomic percentage (ratio of ^{235}U measured in discharged sample relative to total U in discharged sample expressed as a percent) as a function of axial and radial position is illustrated in Fig. 1 for fuel element 49-I. In this plot, E and C stand for experimental and calculated results while H, O, and I stand for central, outer, and inner radial regions, respectively.

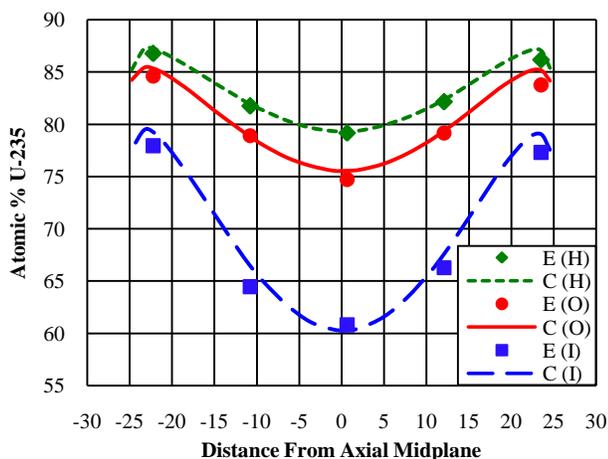


Fig. 1. IFE Post-Irradiation ^{235}U Atomic Percentages for Fuel Element 49-I (Cycle 35, 2319 MWd).

The beginning-of-life (BOL) k_{eff} values were calculated 1 - 3 dollars ($\beta_{\text{eff}} = 0.0076$) greater than unity,

and the end-of-life (EOL) k_{eff} values were calculated 40 - 50 cents less than unity, except for cycle 4. However, cycle 4 was terminated prematurely due to an electrical failure. Sensitivity analyses were performed on the BOL k_{eff} and the placement of the control elements within each burn-up step, and both analyses show that the slightly high BOL k_{eff} has no significant impact on the EOL uranium isotopic compositions. Refer to Ref. 2 for a more detailed explanation on the causes and concerns of producing a k_{eff} slightly greater than unity and other sensitivity analyses performed for this study.

CONCLUSIONS

Atomic percentages of post-irradiation uranium isotopes were calculated and good agreement between the calculated and experimental measurements was observed, which reveal that the methodology developed in this study can be used with reasonable confidence for calculating burn-up dependent nuclide inventories.

The precise calculation of the ^{235}U distribution coupled with previously reported critical experiment calculations [9] indicates that the methodology could be used to accurately calculate the power distribution variation as a function of space and time (burn-up), which is important since the spatial power distribution in HFIR changes significantly during the cycle due to the control element movement. Time and spatially dependent heat generation source terms needed for reactor core thermal hydraulic analyses, if derived from this methodology, have been shown to be accurate for HEU fuel.

No trends were discovered while examining the results, so it was determined that a bias factor isn't needed for current fuel design studies for HEU fuel. Since no nuclear data/computational bias has been identified, none is being applied in current design studies for low enriched uranium (LEU) fuel.

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