

Issues in Three-Dimensional Depletion Analysis of Measured Data Near the End of a Fuel Rod

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INTRODUCTION

The dynamics of reactor operation result in nonuniform axial-burnup profiles in fuel with any significant burnup. At the beginning of life in a pressurized water reactor (PWR), a near-cosine axial-shaped flux will begin depleting fuel near the axial center of a fuel assembly at a greater rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission-product buildup that occur near the center. However, because of the high leakage near the end of the fuel assembly, burnup will drop off rapidly near the ends. Partial-length absorbers or nonuniform axial fuel loadings can further complicate the burnup profile. In a boiling water reactor, the same phenomena come into play, but the burnup profile is complicated by the significant variation of axial moderator density and by nonuniform axial loadings of burnable poison rods.

Numerous studies of axial burnup effects have been published [1–4]. However, most analyses performed in estimation of isotopic distributions due to axial burnup have been based on a set of two-dimensional (2-D) calculations performed for burnups that represent the axial burnup distribution in a fuel assembly. In general, this approach works quite well because the in-core axial gradient of the neutron flux is small over most of the length of the fuel rod, and the 2-D approximation is appropriate.

Conversely, because the axial gradient becomes significant as one approaches either end of the fuel assembly, the 2-D approximation begins to break down at that point. It has been theorized that axial leakage will lead to a

reduced fast flux relative to the thermal flux, softening the spectrum near the ends of the fuel, and that a 2-D approximation is conservative in that it provides more plutonium production. This has not been put to the test, however, for two reasons—a lack of good three-dimensional (3-D) analysis methods acceptable for away-from-reactor applications and, more importantly, a scarcity of experimental measurements for fuel taken from the end regions of a fuel rod.

A number of 3-D depletion approaches based on Monte Carlo methods have been introduced in the past decade including, but not limited to, those listed in Refs. 5–7. A full listing would be quite extensive. Recent fuel-sample measurements from two discharged assemblies of the Takahama Unit 3 PWR provide data for fuel samples taken very close to the top of the active region of the fuel rod. This paper discusses results of TRITON-based 3-D depletion calculations completed in the analysis of the Takahama fuel samples.

APPROACH

The T6-DEPL sequence of the TRITON module in SCALE 5.1 [5] was used for isotopic calculations of spent fuel based on the depletion history provided for fuel sample SF97-1 [8]. This sample, located at an estimated height of 0.4 cm from the top of the fuel rod, was burned to 17.69 GWd/MTU. A top-half assembly KENO-VI model was developed using reflection at the axial midplane. Isotopic concentrations were individually tracked for the sample rod with remaining fuel rods modeled as a group. All fuel rod models were divided into 15 axial segments, with the 7 top segments sized at 0.9 cm each,

based on the fuel pellet size. (A finite sample size is required to collect sufficient neutron histories for reasonable statistics.) The assembly model also included ten burnable poison rods, each represented with four equal-volume rings to properly capture burnup effects; fuel rods were modeled using a single radial zone. Calculations were performed with 2500 generations of 1000 neutrons each, with the first 500 generations skipped.

The first calculation was performed with a burnup of 17.69 GWd/MTU assumed in the first 0.9 cm segment. As discussed below, results showed poor spectral agreement with measured data, and additional calculations were performed assuming the sample was located in each of the remaining six pellet-sized axial locations. For each calculation, the burnup of 17.69 GWd/MTU was applied for the region assumed for the sample. Isotopic concentrations were extracted for each of the seven axial cells and converted to units of milligrams per initial gram of uranium.

RESULTS

Table I shows the ratio of calculated to measured concentrations for each pellet location. Initial calculations for the first pellet location, corresponding to the estimated 0.4 cm location of the fuel sample, showed very poor prediction of all nuclides, especially the plutonium nuclides. This result suggested that the spectrum within the last pellet was significantly softer than that of the measured sample. However, some uncertainty was associated with the measured location of the sample. The size of the top end plug was assumed to be 2.0 cm because actual data were not available. Additionally, it is not clear if plenum size specifications reflected the as-built dimensions, or if axial fuel swelling was accounted for in the provided fuel dimensions. Finally, information on structural materials located beyond the fuel end plugs was not available.

Table I. Calculated-to-Experimental Nuclide Number Density Ratios for SF97-1.

Nuclide	3-D Calculations (cm)							2-D Calculations (cm)
	Pellet 1 (0–0.9)	Pellet 2 (0.9–1.8)	Pellet 3 (1.8–2.7)	Pellet 4 (2.7–3.6)	Pellet 5 (3.6–4.5)	Pellet 6 (4.5–5.4)	Pellet 7 (5.4–6.3)	
U-234	1.131	1.107	1.102	1.074	1.073	1.070	1.052	1.057
U-235	0.986	1.011	1.030	1.048	1.049	1.059	1.065	1.058
U-236	0.947	0.939	0.939	0.932	0.935	0.936	0.933	0.939
U-238	1.002	1.000	0.999	0.998	0.998	0.998	0.997	0.997
Np-237	0.686	0.879	0.941	1.077	1.105	1.143	1.184	1.174
Pu-238	0.550	0.759	0.895	1.094	1.159	1.175	1.260	1.235
Pu-239	0.814	0.933	1.109	1.214	1.222	1.303	1.340	1.335
Pu-240	0.940	1.020	1.021	1.107	1.077	1.141	1.141	1.136
Pu-241	0.662	0.805	0.971	1.139	1.182	1.272	1.391	1.279
Pu-242	0.726	0.883	0.983	1.070	1.112	1.145	1.209	1.163
Am-241	0.816	1.053	1.291	1.443	1.529	1.614	1.714	1.653
Am-243	0.621	0.806	1.112	1.416	1.585	1.611	1.694	0.084
Cm-242	0.542	0.709	0.813	0.994	1.032	1.126	1.205	1.697
Cm-244	0.371	0.607	0.935	1.403	1.502	1.622	1.938	1.131
Nd-143	0.985	0.984	0.982	0.980	0.980	0.979	0.978	1.832
Nd-144	1.060	1.040	1.025	1.010	1.008	1.000	0.994	0.978
Nd-145	1.017	1.008	0.999	0.989	0.989	0.984	0.981	1.000
Nd-146	0.998	0.996	0.997	0.996	0.997	0.994	0.993	0.981
Nd-148	0.995	0.994	0.995	0.994	0.995	0.993	0.993	0.997
Nd-150	0.978	0.993	1.005	1.014	1.016	1.019	1.022	0.992
Cs-137	0.962	0.960	0.959	0.957	0.957	0.956	0.956	1.018
Cs-134	0.667	0.707	0.756	0.855	0.864	0.903	0.921	0.956

TABLE I. (continued)

Nuclide	3-D Calculations (cm)							2-D Calculations (cm)
	Pellet 1 (0–0.9)	Pellet 2 (0.9–1.8)	Pellet 3 (1.8–2.7)	Pellet 4 (2.7–3.6)	Pellet 5 (3.6–4.5)	Pellet 6 (4.5–5.4)	Pellet 7 (5.4–6.3)	
Eu-154	0.728	0.800	0.945	1.084	1.075	1.154	1.169	0.938
Sb-125	1.183	1.220	1.242	1.274	1.275	1.294	1.305	1.196
Ru-106	0.729	0.799	0.840	0.898	0.897	0.931	0.943	1.295
Average	0.844	0.920	0.995	1.082	1.104	1.137	1.175	1.125
Actinide								
Average	0.771	0.894	1.010	1.144	1.183	1.230	1.295	1.139

Table 1 shows the calculated-to-experimental (C/E) ratios for nuclides for which measured data were available. The last two rows show the average of the C/E ratios for all nuclides, and for actinides only. These results indicate that pellet location 3 (1.8 to 2.7 cm from the top of the active fuel region) provides the best agreement with measured data. The two locations above this region show an underprediction of plutonium nuclides. These nuclides are much better predicted in the pellet 3 location. Note that moving farther from the top of the rod results in overprediction of plutonium nuclides coming from spectral hardening (i.e., the loss of fewer fast neutrons). The final column of Table I provides C/E ratios computed using a 2-D approximation in KENO-VI (i.e., a model with axially uniform fuel and reflective boundary conditions).

CONCLUSIONS

Results indicate that uncertainties in the placement of the SF97-1 fuel sample render this measurement of limited value in the validation of 3-D depletion calculations. The results presented indicate that the location of the sample was perhaps 1.5 to 2.5 cm farther from the top of the rod than reported. Certainly the 3-D leakage effects render this sample inappropriate for 2-D calculations. Additional calculations (not reported here) have been performed to determine the effect of varying the moderator/steel ratio beyond the end of the fuel rod. These results show a strong dependence of isotopic predictions to the structure located at the end of a sample fuel rod, but they do not alter the conclusions above.

TRITON's T5-DEPL and T6-DEPL sequences, based on the Monte Carlo codes KENO V.a and KENO-VI, respectively, have been validated in the past for a number of

effectively 2-D radiochemical assay measurements and by comparison to the 2-D deterministic T-DEPL sequence [5]. Although this 3-D measurement does not serve to validate T6-DEPL because of uncertainties in the measurement itself, it does appear to demonstrate the ability of the sequence to capture 3-D effects. Issues associated with Monte Carlo depletion (propagation of uncertainties and variance reduction) remain to be addressed. However, in this case the spectral effects of spatial positioning appear to outweigh the effect of stochastic uncertainty, and a spectral shift with position is clearly seen.

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