REACTOR PHYSICS DESIGN AND CROSS SECTION NORMALIZATION

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The accuracy desired in the physics design of thermal power reactors is generally greater than a priori can be expected when considering the errors in theory and uncertainties in the nuclear data. In an attempt to obtain the desired accuracy, standard engineering practice has been to utilize experimental information as a basis for normalizing calculated results to fit measured values. The adjustment of neutron cross sections has historically been the major vehicle for the adjustments. In this paper, we present results of theory-experiment correlations which illustrates some of the problems encountered with this practice.

An extensive program has been conducted at Battelle-Northwest of comparing calculated and measured reactor parameters for UO_2 , Al-Pu, and UO_2 -Pu O_2 fueled thermal systems.

One invariant result of these studies is the lack of simultaneous agreement in correlation of clean lattice parameters and parameters obtained from irradiation experiments. The discrepancies are such that attempting to adjust cross sections to bring about agreement for the critical experiment parameters leads to increasing the discrepancy in the irradiation experiment parameters. We illustrate with data for some UO_2 fueled systems. The data are: the effective multiplication values for some 2.70 wt.% UO_2 - H_2O critical assemblies,¹ and the isotopic data obtained from the Yankee Core Evaluation Program.²

The calculations were made using the HRG3,³ BRT-1,⁴ and HFN⁵ codes for the critical assemblies and the ZODIAC G code⁶ for a pin cell pertaining to the Yankee Rowe asymptotic region. The neutron cross sections were derived from the Battelle-Northwest Master Library⁷ which were, with some exceptions, identical to those of version 1 - ENDF/B.⁸ The result of the calculations are compared to measured values in Table 1 under the base case column. As noted fairly good agreement is obtained in multiplication values,

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for the criticals. For the burnup data, the calculated ^{239}Pu concentration is ~15% lower than the measured value. In an attempt to bring about better agreement in isotopics during irradiation, another set of ^{238}U resonance cross sections was assembled which increased the ^{238}U resonance neutron capture rate. The radiation width, Γ_γ , of each level was increased to the upper limit of the quoted uncertainty until the infinite dilution resonance integral for ^{238}U (0.5 eV cutoff) became ~282b. (from 269b at mean value). Substituting these ^{238}U resonance cross sections in HRG3 the calculations were repeated and the results are shown in Table 1 under the perturbed column. A large discrepancy occurs in correlation of k_{eff} ; however, the isotopic concentration of ^{239}Pu agrees more favorably with measured values. Thus, simply adjusting the ^{238}U resonance cross sections does not simultaneously yield agreement in k_{eff} and isotopics. A similar trend is noted in the correlations of calculated and measured values for Al-Pu-H₂O systems and UO₂-PuO₂-H₂O systems.

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TABLE 1

Comparison Of Measured & Calculated ResultsYankee Criticals

Lattice Pitch (in)	K_{eff} (Measured = 1.0000)		Change $\Delta K(\%)$ Base/Perturbed
	Base Case	Perturbed Case	
0.405	1.0028	0.9788	2.45
0.435	1.0009	0.9816	1.93
0.470	0.9994	0.9868	1.26
0.573	0.9985	0.9888	0.98
0.615	0.9978	0.9895	0.82
0.665	0.9968	0.9898	0.70

Yankee Burnup

Isotope	Pu-Isotopic Concentration (Atoms/b-cm) at 22,400 MWD/MTM		
	Base Case	Perturbed Case	Measured
^{239}Pu	1.37×10^{-4}	1.58×10^{-4}	1.62×10^{-4}
^{240}Pu	3.5×10^{-5}	3.8×10^{-5}	3.8×10^{-5}
^{241}Pu	2.5×10^{-5}	2.6×10^{-5}	2.7×10^{-5}
^{242}Pu	4.8×10^{-6}	5.1×10^{-6}	4.5×10^{-6}

REFERENCES

1. Davison, P. W., et al., "Yankee Critical Experiments - Measurements on Lattices of Stainless Steel Clad Slightly Enriched Uranium Dioxide Fuel Rods in Light Water," USAEC Report YAEC-94, Westinghouse Electric Corp., April, 1959, and Grob, V. E., et al., "Results of Critical Experiments in Loose Lattices of UO_2 Rods in H_2O ," USAEC Report WCAP-1412, Westinghouse Electric Corp., March, 1960.
2. R. J. Noovick, "Evaluation of Mass Spectrometric and Radiochemical Analysis of Yankee Core I Spent Fuel," WCAP-6068, Westinghouse Electric Corp., March, 1966, and R. P. Matsen, "An Analysis of Yankee-Rowe Burnup Data," BNWL-1122, Pacific Northwest Laboratory, July, 1969.
3. J. L. Carter, "HRG3: A Code for Calculating the Slowing-Down Spectrum in the P_1 or B_1 Approximation," BNWL-1432, Pacific Northwest Laboratory, (1970).
4. C. L. Bennett and W. L. Purcell, "BRT-1: Battelle-Revised-THERMOS," BNWL-1434, Pacific Northwest Laboratory, (1970).
5. J. R. Lilley, "Computer Code HRN-Multigroup, Multiregion Neutron Diffusion Theory in One Space Dimension," HW-71545, Hanford Laboratories, November 1961.
6. R. H. Holeman and D. D. Matsumoto, "ZODIAC-G: A Code for Multigroup, Multiregion Burnup Calculations," BNWL-1559, Pacific Northwest Laboratory, May, 1971.
7. K. B. Stewart, "BNW Master Library," BNWL-CC-325, Pacific Northwest Laboratory, September, 1965, and subsequent revisions.
8. H. C. Honeck, "ENDF/B Specifications for an Evaluated Nuclear Data File for Reactor Applications," BNL-50066(T-467), Brookhaven National Laboratory, May 1966, Revised July 1967.