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BIASES FOR CURRENT FFTF CALCULATIONAL METHODS

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ABSTRACT

Uncertainties in nuclear data and approximate calculational methods used in safety design, and operational support of a reactor yield biased as well as uncertain results. Experimentally based biases for use in Fast Flux Test Facility (FFTF) core calculations have been evaluated and are presented together with a description of calculational methods. Experimental data for these evaluations were obtained from an Engineering Mockup Critical (EMC) of the FFTF core built at the Argonne National Laboratory (ANL). The experiments were conceived and planned by the Hanford Engineering Development Laboratory (HEDL) in cooperation with the Westinghouse Advanced Reactors Division (WARD) and ANL personnel, and carried out by the ANL staff. All experiments were designed specifically to provide data for evaluation of current FFTF core calculational methods. These comprehensive experiments were designed to allow simultaneous evaluations of biases and uncertainties in calculated reactivities, fuel subassembly and material reactivity worths, small sample worths, absorber rod worths, spatial fission rate distributions, power tilting effects and spatial neutron spectra. Modified source multiplication and reactivity anomaly methods have also been evaluated.

Uncertainties in the biases have been established and are sufficiently small to attain a high degree of confidence in the design, safety and operational aspects of the FFTF core.

I. INTRODUCTION

The design of the FFTF core was based on a series of experiments designed to evaluate methods used to calculate core neutronic parameters. The purpose of the experimental program was to determine biases and uncertainties associated with the various aspects of the design, safety and operation of the FFTF core to assure that all core components are designed

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with a high degree of confidence and will function reliably, as designed.

By the late 1960s, considerable improvements in nuclear data, core data and calculational methods were realized as a result of extensive zero power reactor (ZPR-3, ZPR-6, ZPR-9 and ZPPR) experimentation and basic cross section research. However, FFTF core-like engineering mockup data were nonexistent and it was rationalized[1] that for FFTF, additional experiments would be in order. Concepts of basic data adjustments for specific types of core designs had not yet been considered a viable route. Hence, experiments, which were conceived and planned by HEDL[1] in cooperation with WARD and ANL, were carried out by the ANL staff in a full-scale EMC[2] of the FFTF core assembled in the ZPR-9 facility at ANL, Illinois.

The early EMC experimental program converged on a configuration to reproduce as close as possible the actual design of the FFTF core. This was followed by a program which provided the final evaluations of FFTF nuclear design methods and important parameters related to operation and safety. Specifically, these EMC experiments were designed to provide data for use in the evaluation of the existing calculational methods being used in the design of the FFTF itself. Experiments were dedicated to safety, initial startup optimization, and economy of future testing operations. Experiment-to-theory correlations were planned to allow the establishment of calculational biases and calculational uncertainties for a comprehensive array of FFTF nuclear characteristics. Results of assessments of methods currently in use in the FFTF project are presented in this paper.

II. ASSESSMENTS OF BIASES FOR FFTF CALCULATIONS

The basic calculational method used for the design of the FFTF reactor and the analysis of the EMC experiments was multienergy group diffusion theory in either two dimensions with space- and energy-independent buckling, or three dimensions. Cross sections were essentially space and energy self-shielded values derived from the ENDF/B-III cross section library and were used in various energy group structures ranging from four to forty-two groups. In the analysis of the EMC experiments, the heavy metal isotopes were heterogeneously resonance self-shielded and adjusted to account for spatial structural heterogeneities. All measured reactivity values for the EMC reported herein are based upon a conversion factor of 1.0 percent $\Delta k/k$ = 995.22 Ih (Inhours) and a β = 0.0031 derived from ENDF/B-IV delayed neutron data. In addition, two-dimensional perturbation theory was used with two-dimensional diffusion theory fluxes to calculate reactivity worths in some cases.

The experiments analyzed were designed to allow simultaneous evaluations of biases and uncertainties in calculated reactivities, fuel subassembly and material worths, small sample worths, spatial fission rate distributions, power tilting effects and spatial neutron spectra. Modified source multiplication and reactivity anomaly methods have also been evaluated. For each analysis, an analytical model corresponding to the configuration of the EMC at the time of the measurement was developed. A few results of these analyses are provided below and in Table 1, together with one standard deviation (1σ) uncertainties.

Table 1. Biases and Uncertainties Applicable to FFTF Calculations

Parameter	Bias Value	Uncertainty	Units
Eigenvalue K_{eff}	0.0072	±0.0036	$\Delta k = K_{eff} - 1$
Reactivity worth of center fuel subassembly	1.064	±0.020	C/E[a]
Small sample worth			
^{239}Pu	1.153	±0.001	C/E
^{10}B	0.986	±0.003	C/E
Eu_2O_3	1.091	±0.002	C/E
Absorber worth (B_4C)			
Row 3 (safety rods) and Row 5 (control rods)	0.997	±0.025	C/E
Row 7 (peripheral shim rods)	0.932	±0.036	C/E
Doppler constant, BOL	0.0	±0.001	$T\frac{dk}{dT}$
Sodium void reactivity worth	-0.0005	±0.0017[b] (range)	$\%\frac{\Delta k}{k}/kg$[c]
Power density (^{239}Pu fission rate)			
Rows 1 to 4	1.015	±0.02	C/E
Row 5	0.990	±0.02	C/E
Row 6	0.980	±0.04 (range)	C/E

[a] C/E = calculation-to-experiment ratio.

[b] $1.0\%\frac{\Delta k}{k}$ = 995.22 In (Inhours) where $\frac{\Delta k}{k} = \frac{K_{eff-1}}{K_{eff}}$.

[c] Maximum uncertainty (range varies with voiding pattern).

a. Core Eigenvalue Bias

Core eigenvalues calculated with current cross sections and calculational methods and models are, on the average, biased low by 0.0072 Δk as established in analyses of the EMC experiments.[3] It should be noted that diffusion theory is used for both EMC and FFTF. No transport corrections are applied to eigenvalue or reactivity calculations, since it is expected that the transport corrections for the EMC and FFTF are the same.

The 1σ uncertainty associated with this bias value arises from two sources: a component 0.0020 Δk resulting from platelet heterogeneity effects in the critical assemblies, which are not present in the same form in the FFTF core, and a component 0.0030 Δk resulting from the difficulty of constructing EMC configurations that are exactly representative of the FFTF core loadings for which the bias is to be used. The uncertainty, ± 0.0036 Δk, given in Table 1, results from statistically combining the two components 0.002 Δk and 0.003 Δk.

Since the calculated eigenvalue is less than the experimental value, it is necessary to add the bias (0.0072 Δk) to a calculated value to estimate an experimental one; or, as in calculational enrichment searches, the bias is subtracted from the desired experimental eigenvalue to obtain the target calculated value.

b. Fuel Subassembly Worth Bias

Fuel subassembly removal with sodium inflow, will be a frequent gross reactivity event in FFTF refueling. The large driver fuel worths[4] are biased by C/E = 1.064$\pm$0.020 as shown by totally removing a center fuel subassembly in the EMC. The worth of totally removing the subassembly and replacing it with sodium channel material was obtained by calculating the reactivity of the two resulting configurations using 3-D diffusion theory with twelve energy group data and taking the reactivity difference. The model predicts a total worth approximately six percent higher than the experimental measurement. The axial worth profile of removing the fuel subassembly was accurately described by a 2-D, R-Z model which predicts a total worth nine percent higher than experiment (C/E = 1.09$\pm$0.02). However, when normalized to the measured total worth, the calculated axial worth profile of the central fuel subassembly agrees with the measured profile to within the experimental uncertainty of $\pm$0.0124 percent $\Delta k/k$.

c. Small Sample Worth Biases

The central reactivity worths of ten samples containing important FTR core materials[5] including stainless steel, iron oxide, europium oxide, depleted uranium and isotopes of plutonium and boron were calculated and compared with experimentally measured results in the EMC. Reactivity worths for each of the isotopes or elements in the samples were computed using first-order perturbation theory using real and adjoint fluxes from 42 group, 2-D diffusion theory calculations.

Worth biases and C/E values obtained at the core center are 1.153± 0.001 for ^{239}Pu, 0.986±0.003 for ^{10}B, and 1.092±0.002 for europia (ENDF/B-IV data).

At core center the C/E values for other materials are 1.15±0.016 to 1.35±0.021 for plutonium samples, 1.13±0.010 for a depleted uranium sample, 0.98±0.003 to 1.06 0.009 for natural boron, 0.906±0.001 for europia using ENDF/B-III data, 1.41±0.009 for stainless steel, and 2.64±0.95 for iron oxide.

These data are based on a percent $\Delta k/k/Ih$ conversion factor derived from ENDF/B-IV delayed neutron data and represent a 7 percent reduction over previous values calculated on the basis of ENDF/B-III delayed neutron data.

The importance of small sample central worth discrepancy to FFTF design is discounted since one depends on other analyses such as the worth analysis of equivalent drivers and control rods to establish the experiment/theory correlation. However, since the adjustment in the reactivity conversion factor partially accounts for the reduction in the central worth discrepancy, the small sample analysis has provided, indirectly, an incentive for reevaluation of the delayed neutron data.

d. Absorber Worth Biases

Worths of simulated FFTF safety, control, and peripheral shim rods were measured in the EMC for several different rod configurations that produced significant flux tilting and rod shadowing to obtain individual worths and to examine the effects of control rod interactions. Total rod worths were computed in thirty energy groups with a 2-D, X-Y model using diffusion theory. Worth profiles and incremental worths of the absorber rods were computed in four energy groups using a 3-D, X-Y-Z model and diffusion theory.

Experiment/theory correlations show that natural boron carbide absorber rod reactivity worth biases[6] vary with position. For both the row 3 safety rods and row 5 control rods, the bias is C/E = 0.997±0.025. The row 7 peripheral shim rod bias is C/E = 0.932±0.036. The C/E biases for the total worth of the in-core rods are approximately 6 percent higher than for peripheral shim rods. A comparison of the 3-D calculations with experiment indicated that there are modeling problems near the core/reflector boundary.

The results show that the interactions between safety rods were not strong (no larger than a 2 percent change in the worth of one rod due to inserting another rod). However, rod shadowing and flux tilting produced large interactions between the safety rod and control rods located in a single trisector. Individual total rod worths varied by as much as a factor of two depending on the position of the other rods. The shape of a control rod worth profile was significantly changed by inserting the adjacent control rod from fully withdrawn to half inserted.

e. Doppler Constant

The Doppler constant, $(T\frac{dk}{dT}) = -0.0050$, is taken to be unbiased but uncertain by 20 percent based upon evaluations of SEFOR II experimental results[7] and small sample Doppler measurements in the EMC as they apply to the FFTF reactor core.

The uncertainty in the nominal calculated Doppler constant for a particular core configuration has been evaluated with two techniques:

Method I, by which the variation of the calculated Doppler coefficient was established for changes in the ^{238}U resonance parameters, calculational methods, and models of FFTF, about best nominal values and conditions. The uncertainties associated with the above changes result in an overall uncertainty of 15 percent.

Method II, in which the uncertainty evaluation utilized the C/E ratios of the UO_2 small sample Doppler measurements in FFTF criticals and the Doppler measurements of SEFOR Cores I and II.

Extensive small sample Doppler measurements were performed in the EMC program. These measurements were performed at several locations: the core center and off-center, near inserted and withdrawn control rods, and at the core/reflector interface. The statistical analysis of the results of these experiments included the analysis of Doppler measurements in early FFTF criticals (ZPR-3/Assy. 48, 51 and ZPR-9). The C/E values for the early FFTF criticals ranged from 0.8 to 0.92 indicating that the calculations underpredicted the experimental results by 10 to 20 percent. The analysis of the EMC core center Doppler measurements gave a range of C/E values from 0.85 to 0.98, whereas the off-center Doppler measurements yielded C/E values in the range of 1.0 to 1.30 with an average of $\sim$1.10. A statistical analysis of all the C/E values yielded a mean value of 1.0 with a standard deviation of ±0.12. SEFOR superprompt critical experiments were analyzed[8] using FFTF methods to yield a nearly direct assessment of the bias and uncertainty of the FFTF methods. Core II results yielded a small bias of 2 percent and an uncertainty of 11 percent. Extrapolation to FFTF, to account for differences between SEFOR and FFTF, yields an additional uncertainty of ±8 percent. Statistically combining these two values yields a total uncertainty of ±14 percent, which is in good agreement with the results of Method I. From these results, it is expected that the uncertainty is ±15 percent at the 1σ level. However, because the Doppler constant is sensitive to variations in core configurations, an additional allowance is made for possible future departures from the reference core configuration for which the Doppler coefficient was calculated. Thus the total uncertainty is increased to ±20 percent for the purposes of design and safety analysis.

f. Sodium Void Bias

The sodium void reactivity effects[9] are overcalculated by 0.0005 percent $\Delta k/k/Kg$. This bias is spatially independent. The reactivity effects associated with the loss of sodium were obtained using 2-D direct and adjoint fluxes, calculated with diffusion theory, in first order perturbation theory. The calculational procedure predicts the sodium void experiments conducted in the EMC reasonably well if a uniform bias of -0.0005 percent $\Delta k/k/Kg$ is applied to all calculated values. The uncertainty which should be attributed to these worths depends on the nature of the voiding pattern. The uncertainties are taken conservatively to be the maximum deviation between experiment and calculation and range from 0.0007 percent $\Delta k/k/Kg$ to 0.0019 percent $\Delta k/k/Kg$ of voided sodium. Reference 7 gives an uncertainty of 0.0019 percent $\Delta k/k/Kg$ as compared to a calculated core center sodium void worth in the FTR of 0.0128 percent $\Delta k/k/Kg$. This implies a 15 percent uncertainty at the (1σ) confidence level.

g. Fission Rate Biases

Measurements and calculations were made in a row 4 and a row 6 test location in the EMC to obtain ^{239}Pu, ^{238}U and ^{235}U axial fission rate profiles with two different control rod configurations. Results of 3-D diffusion theory calculations[10] show that in the unbanked case, when the calculated and measured fission rates are normalized at the reactor midplane, they agree to within 5 percent out to the axial core boundaries. When the control rods are all banked at half insertion, the agreement is slightly worse with differences as large as 7 percent. In all cases, the agreement is poor near the axial reflector.

Fission rate ratios were measured and calculated for isotopes of plutonium and uranium relative to ^{239}Pu. The measurements were made at the reactor midplane at the center of the core and in rows 4 and 6 test loops.

Three-dimensional results show that the calculated ratios of fission rates of plutonium and uranium isotopes to ^{239}Pu can be calculated with an accuracy of 3 percent for test locations at the center and 2 to 5 percent out to and including row 4. However, agreement with experiment degenerates in and near the reflector due to rapid spectral softening that is not well described by the diffusion theory models. In power density calculations for the FFTF core, the biases used were the fission rate biases and uncertainties derived from analysis of foil irradiation experiments in the EMC given in Reference 11. The values are $C/E = 1.015 \pm 0.02$ for rows 1 to 4, $C/E = 0.990 \pm 0.02$ for row 5 and $C/E = 0.980 \pm 0.02$ for row 6. These biases for each of the three zones were based on averaging the C/E values of all the foils irradiated within the zone. These biases are given with an associated uncertainty based on the variability of the C/E data within the region.

III. CONCLUSIONS

The analysis and evaluation of the data obtained from the EMC experimental program is comprehensive; however, the discussion of the experiment/theory correlations in this paper has been limited to the parameters more important to the design and operation of the FFTF. A more detailed analysis of these and other experiments is provided in a summary document, Reference 12. The analyses of many experiments performed in the EMC have provided a set of bias factors for adjusting the FFTF calculated results and as shown in this paper, demonstrate the adequacy of the core model and calculational methods to predict the experimental results. As a result of the EMC experimental program and the analysis of the experiments, neutron parameters important to the design and operation of the FFTF are known with a sufficient degree of confidence to ensure that the core will function reliably as designed.

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