Testing of the JEF-2.2 and ENDF/B-VI.2 through MCNP Analysis of BWR-simulated Critical Experiments

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Abstract- The JEF-2.2 and ENDF/B-VI.2 nuclear data are tested for applications in light water reactor analyses. Two sets of the BWR-simulated critical experiments at temperatures representative of cold as well as operating conditions are analyzed by using the MCNP-4A code with continuous energy data. Eigenvalue and fission density distribution results from the MCNP simulations with the JEF-2.2 and ENDF/B-VI.2 data show good agreements with the measurements in the all considered temperatures.

1. Introduction

In the past few years, a number of new revisions of major evaluated nuclear data files such as JEF-2.2, ENDF/B-VI.2, JENDL-3.2 and BROND-2.2 have been released and validated through analyses of the Cross Section Evaluation Working Group benchmarks at room temperature. The MCNP[1)] code is used as the basis for developing adequately accurate lattice physics models, which in turn are used in generating homogenized cross section data for codes that simulate and monitor reactor core performance. The use of the MCNP code to simulate critical experiments is an excellent way to check, in an integral sense, the cross section data for use in nuclear design and applications. S. Sitaraman et al.[2)] reported the detailed specifications for the two sets of criticals which were made up of actual power reactor fuel bundle in critical configurations at both room and operating temperatures. In order to validate an applicability of the JEF-2.2 and ENDF/B-VI.2 data for a LWR analysis, the two sets of the BWR-simulated critical experiments varying parameters such as temperature, burnable absorber content, and control poison amounts were analyzed by using the MCNP-4A code with continuous energy cross section data.

2. Description of the Two Critical Experiments

2-1. Experiment Set 1 : A Small Core Critical With and Without Poison Curtains

The core configuration consisted of fuel bundles arranged in a 4 x 4 matrix. Each fuel bundle contained (1.19, 1.67 and 2.42 w/o) ^{235}U enriched UO_2 fuel rods arranged in a 7 x 7. In one case, four central bundles were surrounded by borated stainless steel curtains; the other had no curtains. Both cores were simulated in a full three dimensional configuration with water reflectors radially as well as below the active fuel region. Figure 1 shows the configuration for the small core experiment and MCNP calculational model. The total core height was 403 cm. The active fuel region and the reflector region below it extended 38 cm contained water. The region above the active fuel region had no moderator. The isotopic compositions of the three different fuel types, based on nominal stack fuel densities, are given in Table 1. The rounded coners were replaced in the model with right angle coners. the fuel rods were modeled with the fuel smeared into the gap. The effect of these approximations on the calculational results was found to be negligible in the reference. The case with poison curtains had a critical height of 175.489 cm, and the one without the poison curtains had a critical height of 55.07 cm. The experiments were conducted at room temperature(20 ° C). In addition to the effective multiplication factors, fission density measurements in some selected fuel rods were available in one central bundle of each of the two experiments(data for 42 and 44 fuel rods), respectively with and without poison curtains.

2-2. Experiment Set 2 : Set of Small Core Criticals with Burnable Absorbers

The core configurations and the MCNP calculational model are shown in Fig. 2-1 ~ 2-3. Three basic configurations were considered in these experiments (e.g. BA3GD16, BA3GD4, BA5GD4). All the bundles contained UO_2 fuel rods with five different ^{235}U enrichment arranged in an 8 x 8 matrix. Some selected fuel rods(referred to as gadolinium rods) were poisoned with Gd_2O_3 as a burnable absorber. The bundle types were determined by their gadolinium rod arrangements. Axially, the tie plate region below the active fuel extended 2.3 cm, and the end plugs that were present in this region were explicitly modeled. The end plug consisted of a cylindrical 0.32027 cm radius Zircaloy plug clad in 0.29223 cm thich stainless steel. In all cases, the fuel bundles were modeled to their full height of 365 cm with no moderator above the critical height. A combination of burnable absorber configuration, temperature, and other criticality parameters resulted in a set of ten cases being studied. The critical parameters for each of these ten cases are listed in Table 2.

3. Cross Section Generation and MCNP Calculation

Continuous energy cross section data for MCNP code were prepared with NJOY91.91[3] nuclear data processing system in ACE formats from both JEF-2.2 and ENDF/B-VI.2 data. The cross section sets at three temperatures (300, 363, and 515 K) were generated in reconstruction and thinning tolerance of 0.5 % in the NJOY modules. About the S(α,β) data for hydrogen in water, each of 296, 350, and 500 K value was used for this study. All MCNP-4A simulations were performed on a HP735 workstation. The converged source was obtained in each case from an initial run of 310 cycles and 210 cycles of 1 000 hitories(10 settle cycles) for the first and the second set, respectively. The final runs for calculations of eigenvalues and fission density distributions in the fuel rods were made with 210 cycles of 5 000 histories(10 settle cycles) starting with their corresponding converged sources.

4. Results

In addition to the criticalities, fission density measurements in some selected fuel rods were taken in one central bundle of each of the two experiments(curtained and noncurtained) in the first set. It was thus possible to make comparisons of the rod-by-rod fission density, in addition to the eigenvalues. The calculated eigenvalues for each of the two experiments are presented in Table 3. In both cases using JEF-2.2 and ENDF/B-VI.2, the calculated results agree well with the measurements. Table 3 indicates that the calculated eigen values with JEF-2.2 are slightly higher than those with ENDF/B-VI.2 in all cases. The simple average of calculated eigenvalues of the three temperatures(300, 363 and 515 K) with JEF-2.2 are 1.00033(0.00150), 0.99920(0.00119) and 1.00211(0.00135), respectively. Overall average for the 12 cases with JEF-2.2 is 1.00064(0.00235) and 0.99854(0.00228) with ENDF/B-VI.2. These results suggest that there is no temperature dependent bias in the eigenvalue with the JEF-2.2 and ENDF/B-VI.2 for LWR analysis. The overall root mean square deviations and maximum errors of fission density distributions from the measurements are presented in Table 4. For the experiments with and without poison curtain, Table 5~6 presents the rod-by-rod fission density distributions compared with the measurements with the JEF-2.2 and ENDF/B-VI.2, respectively. In the first set, the overall root mean square deviations of fission density distributions from the measurements with JEF-2.2 and ENDF/B-VI.2 are 1.71% and 1.73% with curtains, and 1.01% and 1.22% without curtains, respectively. Considering the experimental uncertainties(about 2%) and fractional standard deviations of the MCNP calculations(within 1.3%), the results with both JEF-2.2 and ENDF/B-VI.2 show excellent agreements with measured values. The results of fission density distribution with JEF-2.2 agree especially well. This shows that the result using JEF-2.2 data is more or less better than the result with ENDF/B-VI.2.

5. Conclusion

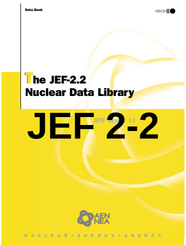
New Data files such as JEF-2.2 and ENDF/B-VI.2 were tested through BWR-simulated experiments in this study. One may conclude that there is no eigenvalue bias between this diverse set of experiments(different fuel configurations and temperatures) and calculations. Since the critical experiments were simulated with approximations that had no consequence, in a statiscal sense, on the MCNP results, one can conclude that the newly released evaluated data are quite adequate for LWR analysis from the results.

References

1. J. Briesmeister (Editor), "MCNP-A General Monte Carlo Code N-Particle Transport Code Version 4A," LA-12625-M (November 1993)

2. S. Sitaraman and F. Rahnema, "Criticality Analysis of Heterogeneous Light Water Reactor Configurations," Nucl. Sci. & Eng., 113, 239-250 (1993)

3. R.E. Macfarlane and D.W. Muir, "The NJOY Nuclear Data Processing System, Version 91," LA-12740-M (October 1994)



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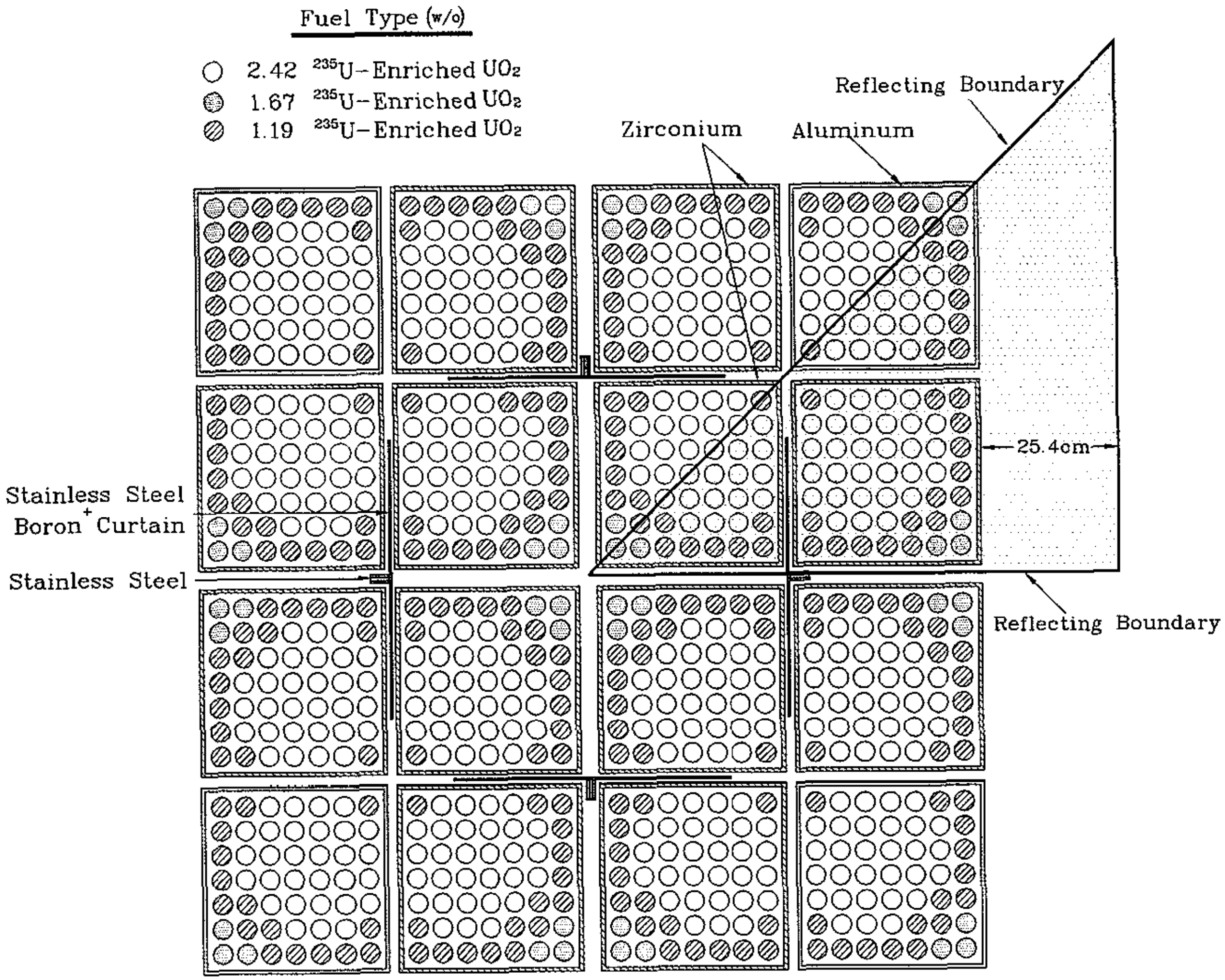
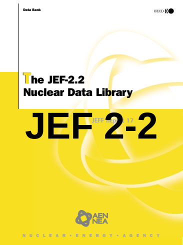


Fig. 1. Core configuration for experiment set 1 (1/8 MCNP Calculational Model)



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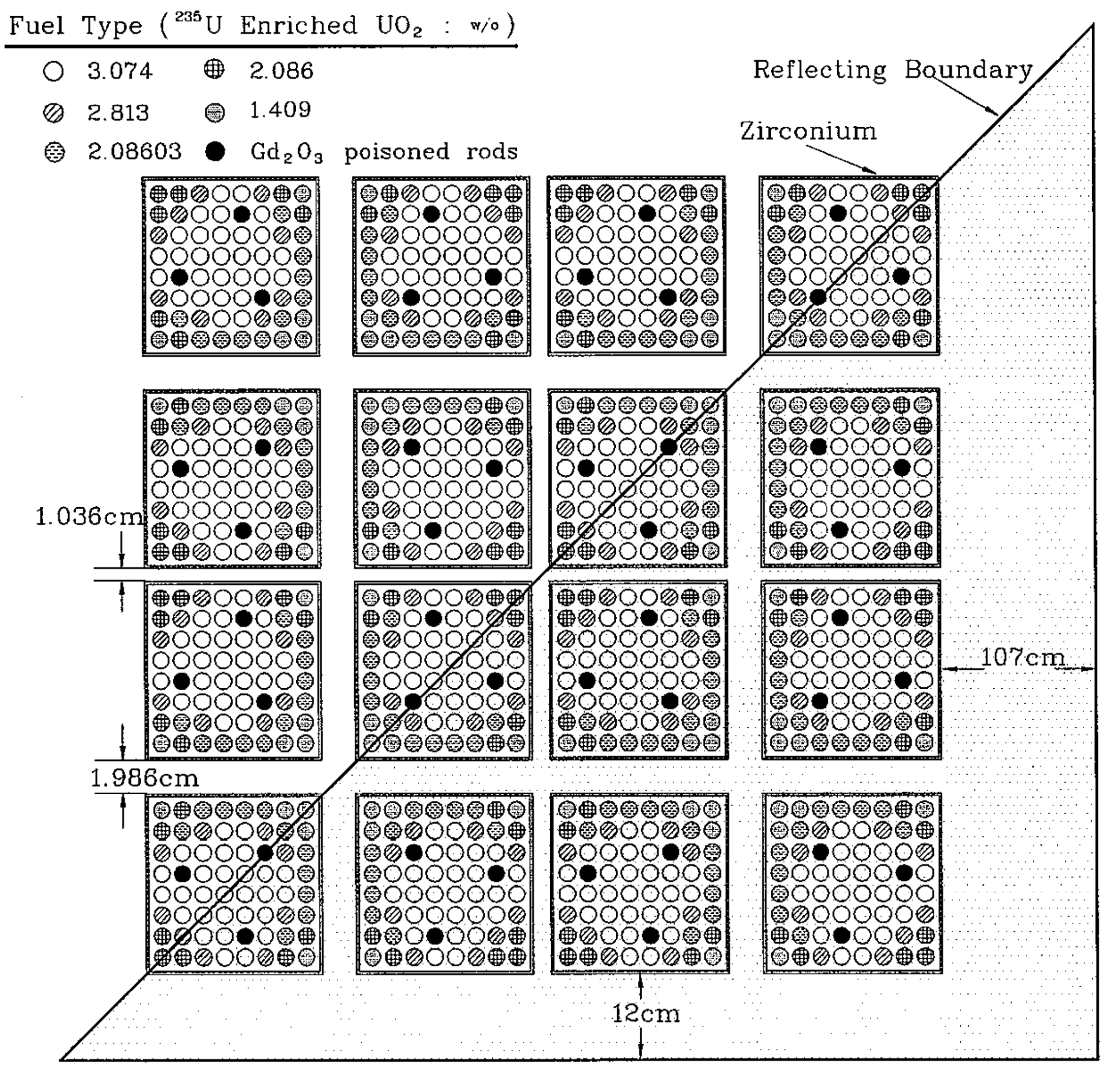
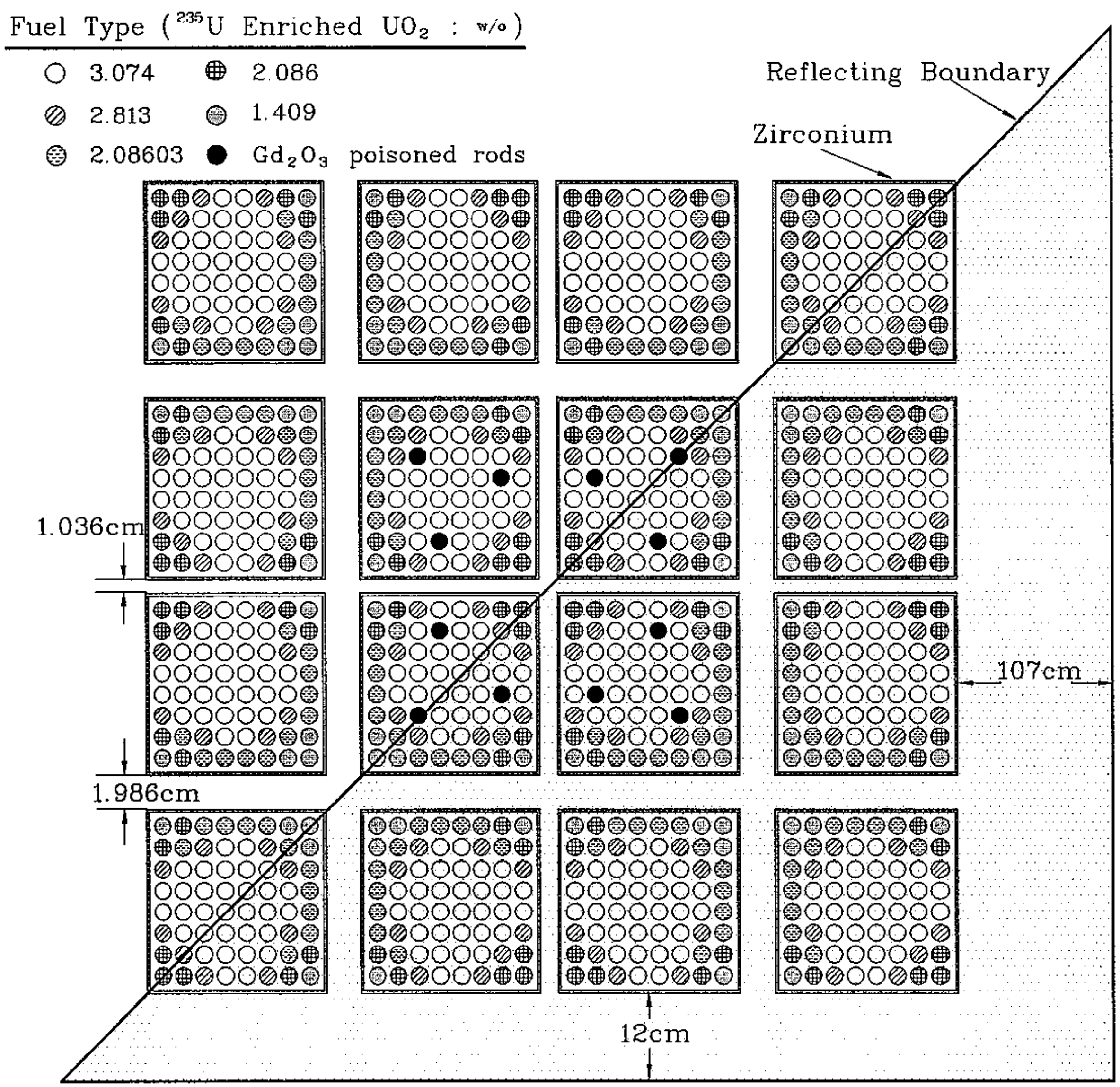


MCNP Calculational Model

Fig. 2-1. Core configuration for experiment set 2 (BA3GD16)



MCNP Calculational Model

Fig. 2-2. Core configuration for experiment set 2 (BA3GD4)

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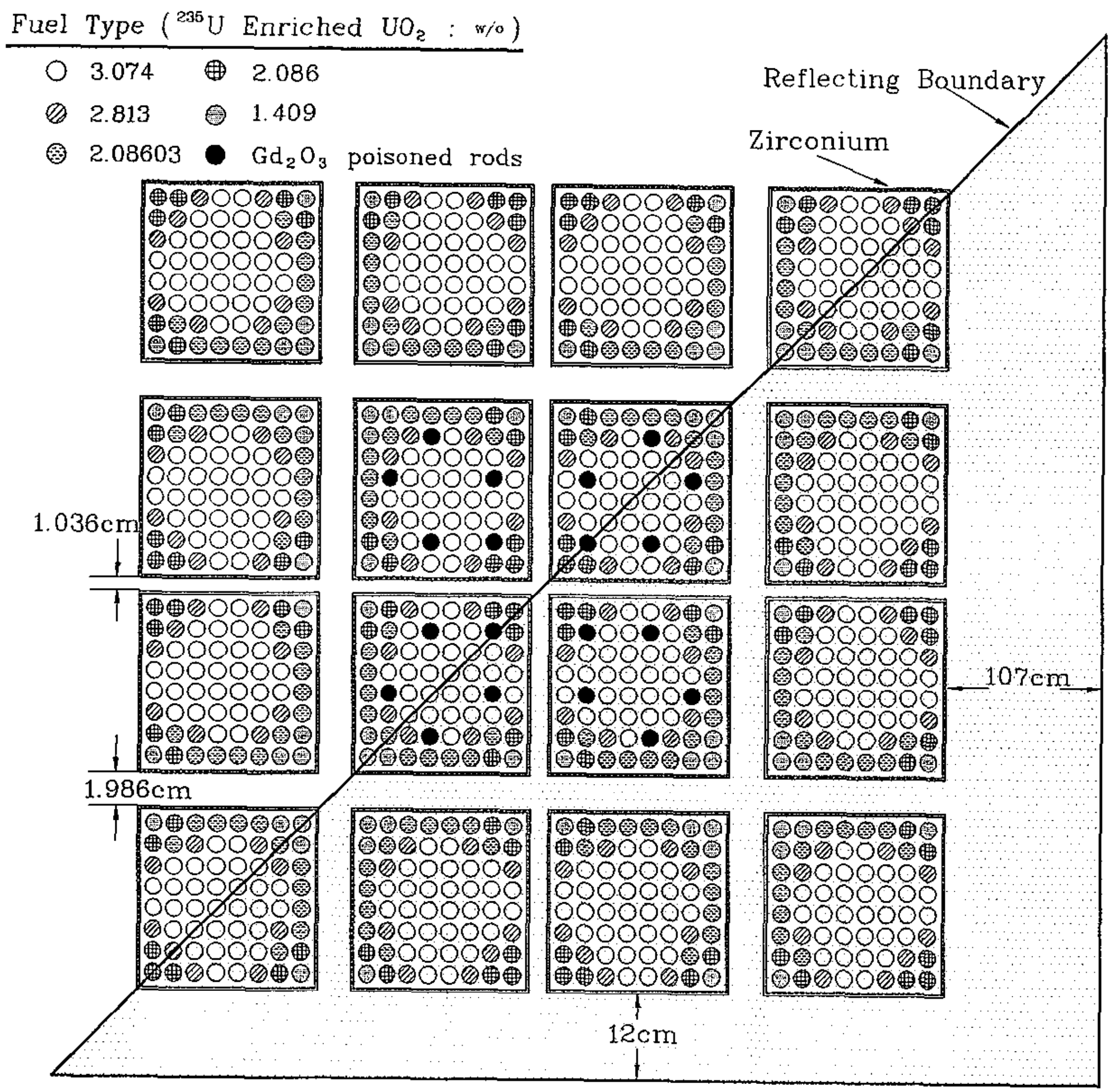

MCNP Calculational Model

Fig. 2−3. Core configuration for experiment set 2 (BA5GD4)

Table 1. Isotopic Compositions and Corresponding Atom Densities (atom/b · cm) for Fuel Types Experiment Set 1

Fuel Type (wt%)	^{235}U	^{238}U	^{16}O
2.42 ^{235}U-enriched UO_2	5.36816E-4 *	2.13722E-2	4.38181E-2
1.67 ^{235}U-enriched UO_2	3.70452E-4	2.15368E-2	4.38145E-2
1.19 ^{235}U-enriched UO_2	2.63968E-4	2.16421E-2	4.38121E-2

* Read as 5.36816×10^{-4}.

Table 2. Criticality Parameters for the Burnable Absorber Criticals

Experiment	Critical Water Height (cm)	Temperature (°C)	Moderator Density (atom or molecule/b · cm)	
			Water	Natural Boron *
BA3GD4	168.30	243	2.69606E-2 **	7.88536E-6
	99.37	245	2.69497E-2	4.26222E-6
	117.40	90	3.22844E-2	8.89366E-6
	108.64	20	3.33689E-2	8.06908E-6
BA5GD4	170.20	242	2.69606E-2	2.85284E-6
	92.61	89	3.22844E-2	1.72170E-6
	94.16	20	3.33689E-2	1.77397E-6
BA3GD16	183.50	242	2.69606E-2	8.08753E-8
	157.89	88	3.22940E-2	3.30989E-6
	92.31	19	3.33751E-2	2.55856E-7

* The natural boron isotopic composition used was 19.9% ^{10}B and 80.1% ^{11}B.
** Read as 2.69606×10^{-2}.

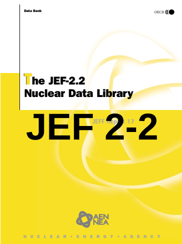

Table 3. Calculated k-eff. Values for Critical Experiments

Core Name *	Temp. (°C)	JEF-2.2	ENDF/B-VI.2
Small Core with Curtain	20	1.00371 (.00067)**	1.00178 (.00071)
Small Core without Curtain	20	0.99814 (.00069)	0.99711 (.00066)
BA3GD4	20	0.99816 (.00067)	0.99672 (.00071)
BA5GD4	20	0.99985 (.00067)	0.99643 (.00062)
BA3GD16	19	1.00180 (.00066)	1.00016 (.00069)
BA3GD4	90	0.99777 (.00066)	0.99573 (.00064)
BA5GD4	89	0.99770 (.00068)	0.99630 (.00067)
BA3GD16	88	1.00214 (.00072)	1.00175 (.00061)
BA3GD4	243	1.00238 (.00067)	0.99891 (.00066)
BA3GD4	245	1.00062 (.00070)	0.99670 (.00069)
BA5GD4	242	1.00097 (.00065)	0.99800 (.00064)
BA3GD16	242	1.00446 (.00068)	1.00288 (.00060)

* from Ref. 2
** fractional standard deviation

Table 4. The Overall Root Mean Square(rms) Deviations and Maximum Errors of the Fission Density Distributions from the Measurements

BWR Core	Temp. (°C)	Relative Error	JEF-2.2	ENDF/B-VI.2
Small Core with Poison Curtain	20	rms * (%)	1.71	1.73
		max. err. (%)	3.35	3.22
Small Core without Poison Curtain	20	rms (%)	1.01	1.22
		max. err. (%)	2.54	3.47

* Root Mean Square

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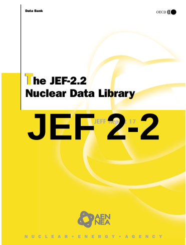

Table 5. The Fission Density Distributions in the Experiment Set 1 With Poison Curtain

JEF-2.2

.948 [a] .968 [b] (-2.03) [c]	.755 .760 (-.72)	.893 .910 (-1.88)	.858 .874 (-1.84)		1.059 1.059 (-.04)	.972 .982 (-1.05)
1.050 1.048 (.23)	1.038 1.032 (.58)	.907 .921 (-1.50)	.874 .889 (-1.63)	.875 .887 (-1.38)	.919 .925 (-.63)	1.059 1.075 (-1.53)
	1.097 1.087 (.90)	.944 .961 (-1.77)	.901 .910 (-.95)	.874 .897 (-2.57)	.875 .887 (-1.38)	
1.176 1.162 (1.23)	1.163 1.161 (.18)	.995 1.008 (-1.31)	.000 .000 (.00)	.901 .918 (-1.82)	.874 .882 (-.85)	.858 .879 (-2.40)
1.295 1.257 (3.02)	.973 .958 (1.55)	1.084 1.094 (-.95)	.995 1.002 (-.72)	.944 .957 (-1.36)	.907 .922 (-1.60)	.893 .904 (-1.23)
1.114 1.082 (2.92)	1.123 1.111 (1.09)	.973 .956 (1.76)	1.163 1.165 (-.16)			.755 .749 (.74)
1.333 1.311 (1.71)	1.114 1.078 (3.30)	1.295 1.253 (3.35)	1.176 1.163 (1.14)		1.050 1.047 (.33)	.948 .926 (2.41)

[a] MCNP
[b] Experiment
[c] (MCNP / Experiment - 1) x 100 (%)

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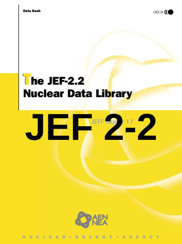

Table 5. (continued)

<u>ENDF/B-VI.2</u>

.944 [a] .968 [b] (−2.49) [c]	.747 .760 (−1.71)	.900 .910 (−1.14)	.853 .874 (−2.41)		1.044 1.059 (−1.41)	.966 .982 (−1.60)
1.060 1.048 (1.12)	1.047 1.032 (1.48)	.915 .921 (−.61)	.872 .889 (−1.97)	.878 .887 (−.96)	.920 .925 (−.57)	1.044 1.075 (−2.88)
	1.088 1.087 (.10)	.952 .961 (−.96)	.904 .910 (−.66)	.876 .897 (−2.36)	.878 .887 (−.96)	
1.196 1.162 (2.94)	1.155 1.161 (−.53)	.997 1.008 (−1.09)		.904 .918 (−1.53)	.872 .882 (−1.19)	.853 .879 (−2.97)
1.281 1.257 (1.89)	.969 .958 (1.11)	1.097 1.094 (.28)	.997 1.002 (−.50)	.952 .957 (−.54)	.915 .922 (−.72)	.900 .904 (−.48)
1.100 1.082 (1.70)	1.120 1.111 (.82)	.969 .956 (1.33)	1.155 1.165 (−.87)			.747 .749 (−.26)
1.353 1.311 (3.22)	1.100 1.078 (2.08)	1.281 1.253 (2.21)	1.196 1.163 (2.85)		1.060 1.047 (1.22)	.944 .926 (1.93)

[a] MCNP
[b] Experiment
[c] (MCNP / Experiment − 1) x 100 (%)

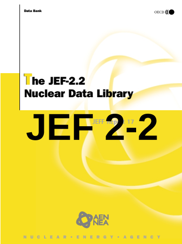

Table 6. The Fission Density Distributions in the Experiment Set 1 Without Poison Curtain

JEF-2.2

1.207 [a] 1.207 [b] (.02) [c]	.944 .941 (.35)	1.109 1.098 (.98)	1.035 1.031 (.38)		1.068 1.058 (.96)	.911 .931 (-2.19)
1.087 1.074 (1.24)	1.080 1.078 (.17)	.928 .932 (-.43)	.866 .885 (-2.18)	.875 .863 (1.36)	.902 .903 (-.06)	1.068 1.076 (-.73)
	1.038 1.036 (.23)	.897 .892 (.52)	.845 .866 (-2.48)	.843 .846 (-.32)	.875 .869 (.66)	
1.092 1.083 (.79)	1.069 1.065 (.36)	.910 .925 (-1.66)		.845 .853 (-.99)	.866 .885 (-2.18)	1.035 1.039 (-.39)
1.170 1.162 (.65)	.873 .869 (.47)	.984 .996 (-1.24)	.910 .920 (-1.12)	.897 .898 (-.15)	.928 .931 (-.32)	1.109 1.118 (-.82)
.992 .967 (2.54)	1.001 1.015 (-1.40)	.873 .873 (.01)	1.069 1.069 (-.02)	1.038 1.028 (1.01)	1.080 1.080 (-.01)	.944 .944 (.03)
1.194 1.185 (.73)	.992 .986 (.56)	1.170 1.157 (1.08)	1.092 1.083 (.79)		1.087 1.081 (.58)	1.207 1.206 (.10)

[a] MCNP
[b] Experiment
[c] (MCNP / Experiment - 1) x 100 (%)

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Table 6. (continued)

ENDF/B-VI.2

1.222 [a] 1.207 [b] (1.25) [c]	.948 .941 (.73)	1.085 1.098 (-1.14)	1.042 1.031 (1.08)		1.054 1.058 (-.42)	.931 .931 (.02)
1.101 1.074 (2.49)	1.075 1.078 (-.23)	.922 .932 (-1.06)	.880 .885 (-.56)	.868 .863 (.57)	.900 .903 (-.37)	1.054 1.076 (-2.09)
	1.029 1.036 (-.66)	.887 .892 (-.60)	.836 .866 (-3.47)	.838 .846 (-.94)	.868 .869 (-.13)	
1.097 1.083 (1.32)	1.070 1.065 (.51)	.914 .925 (-1.19)		.836 .853 (-2.00)	.880 .885 (-.56)	1.042 1.039 (.30)
1.166 1.162 (.37)	.880 .869 (1.32)	1.000 .996 (.45)	.914 .920 (-.65)	.887 .898 (-1.26)	.922 .931 (-.95)	1.085 1.118 (-2.91)
.986 .967 (1.92)	1.015 1.015 (-.01)	.880 .873 (.85)	1.070 1.069 (.13)	1.029 1.028 (.11)	1.075 1.080 (-.42)	.948 .944 (.41)
1.190 1.185 (.39)	.986 .986 (-.05)	1.166 1.157 (.80)	1.097 1.083 (1.32)		1.101 1.081 (1.82)	1.222 1.206 (1.33)

[a] MCNP
[b] Experiment
[c] (MCNP / Experiment − 1) x 100 (%)