MOL.19990920.0167

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Civilian Radioactive Waste Management System
Management & Operating Contractor

Summary Report of Laboratory Critical Experiment Analyses Performed for the Disposal Criticality Analysis Methodology

B00000000-01717-5705-00076 REV 02

September 1999

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Under Contract Number
DE-AC08-91RW00134

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HISTORY OF CHANGE PAGE

Initial Issuance, REV 00 .... September 4, 1997

Revised, REV 01 ....July 16, 1998

Replaced all of Section 2.2
Added Section 3.2.6
Replaced all of Section 4
Deleted the HXST and HXT calculations due to redundancy

Revised, REV 02 .... September 1999

Complete revision to include calculations performed with latest version of MCNP code (MCNP4B2), to evaluate results produced with use of different cross-section libraries, and to evaluate the results from additional cases

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1. INTRODUCTION

This report, *Summary Report of Laboratory Critical Experiment Analyses Performed for the Disposal Criticality Analysis Methodology*, contains a summary of the laboratory critical experiment (LCE) analyses used to support the validation of the disposal criticality analysis methodology.

1.1 BACKGROUND

The United States Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM) is developing a methodology for criticality analysis to support disposal of commercial spent nuclear fuel in a geologic repository. A supplement to the Disposal Criticality Analysis Methodology Topical Report is scheduled to be submitted to the United States Nuclear Regulatory Commission (NRC) for formal review later in 1999. This technical report is one of a series of reports that provides a summary of the results of the analyses that support the development of the disposal criticality analysis methodology.

1.2 OBJECTIVE

The objective of this report is to present a summary of the LCE analyses' results. These results demonstrate the ability of MCNP to accurately predict the critical multiplication factor (k_{eff}) for fuel with different configurations. Results from the LCE evaluations will support the development and validation of the criticality models used in the disposal criticality analysis methodology. These models and their validation have been discussed in the *Disposal Criticality Analysis Methodology Topical Report* (CRWMS M&O 1998a).

1.3 SCOPE

The scope of this *Summary Report* includes the LCE analytical results for the following types of critical experiments:

- Lattice Criticals
- Homogeneous Mixture Criticals
- Fast Metal Fuel Criticals.

Additional types of critical experiments may be added in future revisions to this report.

1.4 QUALITY ASSURANCE

The Quality Assurance (QA) program applies to the development of this report. The data provided in this report will indirectly be used to develop the methodology for evaluating the Monitored Geologic Repository (MGR) waste package and engineered barrier segment. The QAP-2-3 (*Classification of Permanent Items*) evaluation entitled *Classification of the Preliminary MGDS Repository Design* (p. IV-11, CRWMS M&O 1999a) has identified the waste package as an MGR item important to radiological safety and waste isolation. The Waste Package Operations responsible manager has evaluated the technical document development activity in accordance with QAP-2-0, *Conduct of Activities*. The QAP-2-0 activity evaluation, *Neutronics Methodology* (CRWMS M&O 1999e), has determined that the preparation and review of this technical document is subject to *Quality Assurance Requirements and Description* (DOE OCRWM 1998) requirements. As specified in NLP-3-18, *Documentation of QA Controls on Drawings, Specifications, Design Analyses, and Technical Documents*, this activity is subject to QA controls.

1.5 USE OF COMPUTER SOFTWARE

As discussed in CRWMS M&O (1999b), CRWMS M&O (1999c), and CRWMS M&O (1999d), the MCNP code was used to calculate the eigenvalue for the LCE configurations. The software specifications are as follows:

Program Name: MCNP
Version/Revision Number: Version 4B2LV
CSCI Number: 30033 V4B2LV
Computer Type: HP 9000 Series Workstations
Computer Processing Unit Number: (Bloom) 700887

The input and output files for the various MCNP calculations are documented throughout CRWMS M&O (1999b), CRWMS M&O (1999c), and CRWMS M&O (1999d), such that an independent repetition of the software use may be performed. The MCNP software used was: (a) appropriate for the application of eigenvalue calculations, (b) used only within the range of validation as documented throughout Briesmeister (1997) and CRWMS M&O (1998b), and (c) obtained from the Software Configuration Manager in accordance with appropriate procedures.

Title: Excel
Version/Revision Number: Microsoft® Excel 97

The Excel spreadsheet program was used for simple graphical displays of the results as presented in Section 4 of this report.

2. ANALYSIS MODEL

This section provides a description of the configurations used to generate the supporting analytic results reported in this document.

2.1 CRITICALITY MODEL

The criticality models used to calculate the reactivity of the various experiments are discussed in CRWMS M&O (1999b), CRWMS M&O (1999c), and CRWMS M&O (1999d). The tool used to assess the reactivity of each of the experiments is MCNP 4B (Briesmeister 1997 and CRWMS M&O 1998b), which is an implementation of the Monte Carlo method. The computer inputs for the LCE calculations are provided in Attachments I, III, and V of CRWMS M&O (1999b), CRWMS M&O (1999c), and CRWMS M&O (1999d).

2.2 CROSS SECTIONS

Table 2.2-1 lists all of the MCNP cross section library identifiers (ZAIDs) used in the LCE reactivity calculations documented in CRWMS M&O (1999b), CRWMS M&O (1999c), and CRWMS M&O (1999d). The MCNP ZAID is used to identify a cross section library. The ZAID consists of a four- or five-digit element and isotope identifier followed by a cross section library suffix. The first one or two digits in the ZAID refer to the atomic number of the corresponding element. The three digits preceding the decimal always refer to the isotopic mass number. The suffix identifies the library. Details are in Appendix G of Briesmeister (1997).

Each of the critical experiments were run with three sets of cross-section libraries – ENDF/B-V, ENDF/B-VI, and a hybrid set that is a combination of the two referred to as the WPO Selected set which comes from CRWMS M&O (1998c). The WPO Selected set uses isotopic cross-section data that is from ENDF/B-VI in place of elemental cross-sections that came from ENDF/B-V for certain elements. A complete listing is provided in CRWMS M&O (1998c).

Table 2.2-1. MCNP Cross Section Libraries Used in the LCE Reactivity Calculations

Element/Isotope	MCNP ZAID	Temperature (K)	Library name	Data source
H-1	1001.50c	294.0	rmccs	ENDF/B-V.0
H-1	1001.60c	294.0	endf60	ENDF/B-VI.1
H-2	1002.55c	294.0	rmccs	LANL/T-2
H-2	1002.60c	294.0	endf60	ENDF/B-VI.0
Li-6	3006.50c	294.0	rmccs	ENDF/B-V.0
Li-6	3006.60c	294.0	endf60	ENDF/B-VI.1
Li-7	3007.55c	294.0	rmccs	ENDF/B-V.2
Li-7	3007.60c	294.0	endf60	ENDF/B-VI.0
Be-9	4009.50c	294.0	rmccs	ENDF/B-V.0
Be-9	4009.60c	294.0	endf60	ENDF/B-VI.0
B-10	5010.50c	294.0	rmccs	ENDF/B-V.0
B-10	5010.60c	294.0	endf60	ENDF/B-VI.1
B-11	5011.56c	294.0	newxs	LANL/T-2
B-11	5011.60c	294.0	endf60	ENDF/B-VI.0
C-nat	6000.50c	294.0	rmccs	ENDF/B-V.0
C-nat	6000.60c	294.0	endf60	ENDF/B-VI.1
C-12	6012.50c	294.0	rmccs	ENDF/B-V.0

Table 2.2-1. MCNP Cross Section Libraries Used in the LCE Reactivity Calculations

Element/Isotope	MCNP ZAID	Temperature (K)	Library name	Data source
N-14	7014.50c	294.0	rmccs	ENDF/B-V.0
N-14	7014.60c	294.0	endf60	LANL/T-2
N-15	7015.55c	294.0	rmccsa	LANL/T-2
N-15	7015.60c	294.0	endf60	ENDF/B-VI.0
O-16	8016.50c	294.0	rmccs	ENDF/B-V.0
O-16	8016.60c	294.0	endf60	ENDF/B-VI.0
O-17	8017.60c	294.0	endf60	ENDF/B-VI.0
F-19	9019.50c	294.0	endf5p	ENDF/B-V.0
F-19	9019.60c	294.0	endf60	ENDF/B-VI.0
Na-23	11023.50c	294.0	endf5p	ENDF/B-V.0
Na-23	11023.60c	294.0	endf60	ENDF/B-VI.1
Mg-nat	12000.50c	294.0	endf5u	ENDF/B-V.0
Mg-nat	12000.60c	294.0	endf60	ENDF/B-VI.0
Al-27	13027.50c	294.0	rmccs	ENDF/B-V.0
Al-27	13027.60c	294.0	endf60	ENDF/B-VI.0
Si-nat	14000.50c	294.0	endf5p	ENDF/B-V.0
Si-nat	14000.60c	294.0	endf60	ENDF/B-VI.0
P-31	15031.50c	294.0	endf5u	ENDF/B-V.0
P-31	15031.60c	294.0	endf60	ENDF/B-VI.0
S-32	16032.50c	294.0	endf5u	ENDF/B-V.0
S-32	16032.60c	294.0	endf60	ENDF/B-VI.0
Cl-nat	17000.50c	294.0	endf5p	ENDF/B-V.0
Cl-nat	17000.60c	294.0	endf60	ENDF/B-VI.0
K-nat	19000.50c	294.0	endf5u	ENDF/B-V.0
K-nat	19000.60c	294.0	endf60	ENDF/B-VI.0
Ca-nat	20000.50c	294.0	endf5u	ENDF/B-V.0
Ca-nat	20000.60c	294.0	endf60	ENDF/B-VI.0
Ti-nat	22000.50c	294.0	endf5u	ENDF/B-V.0
Ti-nat	22000.60c	294.0	endf60	ENDF/B-VI.0
V-nat	23000.50c	294.0	endf5u	ENDF/B-V.0
V-nat	23000.60c	294.0	endf60	ENDF/B-VI.0
Cr-nat	24000.50c	294.0	rmccs	ENDF/B-V.0
Cr-50	24050.60c	294.0	endf60	ENDF/B-VI.1
Cr-52	24052.60c	294.0	endf60	ENDF/B-VI.1
Cr-53	24053.60c	294.0	endf60	ENDF/B-VI.1
Cr-54	24054.60c	294.0	endf60	ENDF/B-VI.1
Mn-55	25055.50c	294.0	endf5u	ENDF/B-V.0
Mn-55	25055.60c	294.0	endf60	ENDF/B-VI.0
Fe-nat	26000.55c	294.0	rmccs	LANL/T-2
Fe-54	26054.60c	294.0	endf60	ENDF/B-VI.1
Fe-56	26056.60c	294.0	endf60	ENDF/B-VI.1
Fe-57	26057.60c	294.0	endf60	ENDF/B-VI.1
Fe-58	26058.60c	294.0	endf60	ENDF/B-VI.1
Ni-nat	28000.50c	294.0	rmccs	ENDF/B-V.0
Ni-58	28058.60c	294.0	endf60	ENDF/B-VI.1
Ni-60	28060.60c	294.0	endf60	ENDF/B-VI.1
Ni-61	28061.60c	294.0	endf60	ENDF/B-VI.1
Ni-62	28062.60c	294.0	endf60	ENDF/B-VI.1
Ni-64	28064.60c	294.0	endf60	ENDF/B-VI.1
Cu-nat	29000.50c	294.0	rmccs	ENDF/B-V.0
Cu-63	29063.60c	294.0	endf60	ENDF/B-VI.2
Cu-65	29065.60c	294.0	endf60	ENDF/B-VI.2
Ga-nat	31000.50c	294.0	rmccs	ENDF/B-V.0
Ga-nat	31000.60c	294.0	endf60	ENDF/B-VI.0
Zr-nat	40000.56c	294.0	misc5xs	ENDF/B-V: XTM
Zr-nat	40000.60c	294.0	endf60	ENDF/B-VI.1

Table 2.2-1. MCNP Cross Section Libraries Used in the LCE Reactivity Calculations

Element/Isotope	MCNP ZAID	Temperature (K)	Library name	Data source
Nb-93	41093.50c	294.0	endf5p	ENDF/B-V.0
Nb-93	41093.60c	294.0	endf60	ENDF/B-VI.1
Mo-nat	42000.50c	294.0	endf5u	ENDF/B-V.0
Mo-nat	42000.60c	294.0	endf60	ENDF/B-VI.0
Cd-nat	48000.50c	294.0	endf5u	ENDF/B-V.0
Ba-138	56138.50c	294.0	rmccs	ENDF/B-V.0
Ba-138	56138.60c	294.0	endf60	ENDF/B-VI.0
Gd-152	64152.50c	294.0	endf5u	ENDF/B-V.0
Gd-152	64152.60c	294.0	endf60	ENDF/B-VI.0
Gd-154	64154.50c	294.0	endf5u	ENDF/B-V.0
Gd-154	64154.60c	294.0	endf60	ENDF/B-VI.0
Gd-155	64155.50c	294.0	endf5u	ENDF/B-V.0
Gd-155	64155.60c	294.0	endf60	ENDF/B-VI.0
Gd-156	64156.50c	294.0	endf5u	ENDF/B-V.0
Gd-156	64156.60c	294.0	endf60	ENDF/B-VI.0
Gd-157	64157.50c	294.0	endf5u	ENDF/B-V.0
Gd-157	64157.60c	294.0	endf60	ENDF/B-VI.0
Gd-158	64158.50c	294.0	endf5u	ENDF/B-V.0
Gd-158	64158.60c	294.0	endf60	ENDF/B-VI.0
Gd-160	64160.50c	294.0	endf5u	ENDF/B-V.0
Gd-160	64160.60c	294.0	endf60	ENDF/B-VI.0
W-nat	74000.55c	294.0	rmccs	ENDF/B-V.2
W-182	74182.60c	294.0	endf60	ENDF/B-VI.0
W-183	74183.60c	294.0	endf60	ENDF/B-VI.0
W-184	74184.60c	294.0	endf60	ENDF/B-VI.0
W-186	74186.60c	294.0	endf60	ENDF/B-VI.0
U-233	92233.50c	294.0	rmccs	ENDF/B-V.0
U-233	92233.60c	294.0	endf60	ENDF/B-VI.0
U-234	92234.50c	294.0	endf5p	ENDF/B-V.0
U-234	92234.60c	294.0	endf60	ENDF/B-VI.0
U-235	92235.50c	294.0	rmccs	ENDF/B-V.0
U-235	92235.60c	294.0	endf60	ENDF/B-VI.2
U-236	92236.50c	294.0	endf5p	ENDF/B-V.0
U-236	92236.60c	294.0	endf60	ENDF/B-VI.0
U-238	92238.50c	294.0	rmccs	ENDF/B-V.0
U-238	92238.60c	294.0	endf60	ENDF/B-VI.2
Pu-238	94238.50c	294.0	endf5p	ENDF/B-V.0
Pu-238	94238.60c	294.0	endf60	ENDF/B-VI.0
Pu-239	94239.55c	294.0	rmccs	ENDF/B-V.2
Pu-239	94239.60c	294.0	endf60	ENDF/B-VI.2
Pu-240	94240.50c	294.0	rmccs	ENDF/B-V.0
Pu-240	94240.60c	294.0	endf60	ENDF/B-VI.2
Pu-241	94241.50c	294.0	endf5p	ENDF/B-V.0
Pu-241	94241.60c	294.0	endf60	ENDF/B-VI.1
Pu-242	94242.50c	294.0	endf5p	ENDF/B-V.0
Pu-242	94242.60c	294.0	endf60	ENDF/B-VI.0
Am-241	95241.50c	294.0	endf5u	ENDF/B-V.0
Am-241	95241.60c	294.0	endf60	LANL/T-2

pp. 6 through 9 of CRWMS M&O 1999b

NOTE: * nat = natural element composition

3. DESCRIPTION OF THE EXPERIMENT SYSTEMS

The represented experiments are described in Baldwin et al. (1979) through Wittekind (1992). The data from OECD-NEA (1998) is from a standard handbook, is generally accepted by the scientific and engineering community, and used in a number of license applications and validation reports through out the nuclear industry. The data in this reference is therefore considered "Accepted Data". Throughout the rest of this section, information from the references specified in Table 3-1 should be considered to be verified (TBV) in that they are not considered accepted data sources per the retroactive procedural requirement of AP-SIII.2Q initiated by the July 27, 1999 issuance of the DOE Letter, "Accepted Data Call", from R.E. Spence to J.L. Younker (DOE 1999).

Table 3-1. TBV References and Associated Tracking Numbers

Reference	TBV Tracking Number
Baldwin et al. 1979	TBV-1357
Newman 1984	TBV-1358
Taylor 1965	TBV-1359
Wittekind 1992	TBV-1360
Bierman 1990	TBV-1361
Bierman et al. 1977	TBV-1362
Bierman et al. 1981	TBV-1363
Bierman and Clayton 1981	TBV-1364
Durst et al. 1982	TBV-1365
Bierman et al. 1984	TBV-1366
Brown et al. 1965	TBV-1367
DeHardt and Bowman 1995	TBV-1368
Mele et al. 1994	TBV-1369
Miyoshi et al. 1997	TBV-1370
Bierman et al. 1979	TBV-1371

It should be noted that some of the experiments were not true critical configurations, but were subcritical approaches extrapolated to critical. The number of digits in the values cited herein may be the results of a calculation or may reflect the input from another source; consequently, the number of digits should not be interpreted as an indication of accuracy. In the following sections, intermediate-enriched uranium (IEU) is defined as having a ^{235}U concentration greater than 10 wt% but less than 80 wt%. Low-enriched uranium (LEU) and high-enriched uranium (HEU) are outside this range.

3.1 LATTICE EXPERIMENTS

The fresh fuel LCEs presented in this section represent moderated lattice configurations containing fissile oxide fuel. Each of the LCE configurations described in this section have been analyzed with the MCNP code system using the cross section library previously described (Section 2.2). An experiment identifier for each configuration is provided for subsequent reference when the results are reported.

3.1.1 Critical Configurations of Subcritical Clusters of 2.35 wt% Enriched UO_2 Rods in Water with Fixed Neutron Absorber Plates

Experiments with subcritical clusters of low-enrichment UO_2 fuel rods were performed at the Pacific Northwest Laboratory and documented by Bierman et al. (1977). The four experiments modeled with MCNP consisted of three rectangular arrays of aluminum-clad fuel rods. The fuel rods comprising the arrays had a uniform enrichment of 2.35 wt% ^{235}U. The three arrays of fuel were arranged in a row and, in three of the experiments, sheets of neutron poison were interposed between adjacent arrays. The structure of the experimental assembly was provided by aluminum structural members on the margins of the fuel arrays. Axial support for the fuel rods was provided by an acrylic base plate. The lateral alignment of the fuel rods was provided by another acrylic plate. The experimental apparatus was closely reflected by full-density water.

The pertinent differences among these four experiments are shown in Table 3.1-1. These critical experiments help demonstrate the ability of MCNP to accurately predict the critical multiplication factor for configurations containing light-water reactor fuel separated by absorber plates.

Table 3.1-1. Differences in Absorber Plates used for Clusters of 2.35 wt% UO_2 Fuel Rods

Experiment identifier	Interposed plate
exp1	none
exp2	Boral™
exp3	Type 6061 Aluminum
exp4	Type 304 Stainless Steel

3.1.2 Water-Reflected Fuel Rod Clusters in Square Pitched Arrays

A series of critical experiments with clusters of aluminum clad UO_2 fuel rods in a large water-filled tank was performed over a period of several years at the Critical Mass Laboratory at Pacific Northwest Laboratories (PNL). Eight cases were analyzed under this category that correspond to water-reflected clusters at 2.032 cm square pitch with no absorber plates, reflecting walls, dissolved poison, or gadolinium impurity. Table 3.1-2 provides a brief description of the experiments. Each of the experiments used 2.35 wt% ^{235}U enriched UO_2 fuel with an average loading of 17.08 g of ^{235}U per rod (OECD-NEA 1998, p. 7 LCT-001).

Table 3.1-2. Water-Reflected Fuel Rod Cluster Critical Experiments

Experiment identifier	Description (p. 10, OECD-NEA 1998, LCT-001) number of rods[1] (X x Y), number of clusters, cluster separation	H/X ratio[2]
Case 1	20 x 18.08, 1 cluster	459
Case 2	20 x 17, 3 clusters, 11.92 ± 0.04 cm separation	487
Case 3[3]	20 x 16, 3 clusters, 8.41 ± 0.05 cm separation	469
Case 4	20 x 16 (center), 22 x 16 (two outer), 3 clusters, 10.05 ± 0.05 cm separation	474
Case 5	20 x 15, 3 clusters, 6.39 ± 0.05 cm separation	459
Case 6	20 x 15 (center), 24 x 15 (two outer), 3 clusters, 8.01 ± 0.06 cm separation	462
Case 7	20 x 14, 3 clusters, 4.46 ± 0.10 cm separation	449
Case 8[4]	19 x 6, 3 clusters, 7.57 ± 0.04 cm separation	467

NOTES: [1] For three-cluster configurations, the first dimension is along the direction of the cluster placement. The second dimension is the width of facing sides, as shown in Figure 5 of OECD-NEA 1998, on p. 11 LCT-001.

[2] The H/X ratio is the ratio of hydrogen to fissile material per unit cell. These values are from p. 9 of CRWMS M&O 1999c.

[3] The cluster separation referenced was 8.41 cm, but footnote (d) on p. 10 LCT-001 of OECD-NEA 1998, states that the cluster separation should be 0.762 cm less. Thus, 7.648 cm was represented in the MCNP case for the cluster separation.

[4] The cluster separation referenced was 7.57 cm, but footnote (d) on p. 10 LCT-001 of OECD-NEA 1998, states that the cluster separation should be 0.762 cm less. Thus, 6.808 cm was represented in the MCNP case for the cluster separation.

3.1.3 LEU Systems Typical of N-Reactor Fuel in the K Basin

Three cases which analyzed a lattice of actual N-Reactor MKIA fuel elements were performed. The MKIA cases analyzed 101.2 fuel elements, 67.4 fuel elements, and 90.3 fuel elements with corresponding pitches of 7.112 cm, 7.874 cm, and 8.636 cm, respectively (p. 52, Wittekind 1992). The MKIA experiments were performed for three different lattice pitches resulting in three experimental values to achieve a k_{eff} of unity. The lattice pitches and corresponding critical number of fuel elements are listed in Table 3.1-3. It should be noted that the MKIA experiments used 26.2 in. (66.548 cm) long fuel elements stacked two high for a 52.4 in. (133.096 cm) fuel column per lattice (p. 5, Brown et al. 1965) with 121 filled fuel lattices (p. 52, Wittekind 1992).

Table 3.1-3. MKIA Fuel Assembly Experiments

Experiment identifier	Lattice pitch	Critical number of fuel elements experimentally determined	H/X ratio[1]
SUBC2P8H	7.112 cm	101.2	994
SUBC3P1H	7.874 cm	67.4	1414
SUBC3P4H	8.636 cm	90.3	1876

NOTE: [1] Values are from p. 11 of CRWMS M&O 1999c

3.1.4 Critical Configurations with Subcritical Clusters of 4.31 wt% Enriched UO_2 Rods in Water with Reflecting Walls

These three experiments were also performed at the PNL and were documented in Bierman et al. (1981) and Bierman and Clayton (1981). In these experiments three similar fuel assemblies were laterally surrounded by reflectors of different compositions. The fuel lattices in each critical experiment contained 4.31 wt% ^{235}U enriched UO_2 fuel rods on a square pitch of 1.892 cm. The distinguishing characteristics of each experiment are given in Table 3.1-4. These critical experiments demonstrate the ability of MCNP to accurately predict the critical multiplication factor for configurations with different shielding materials used for reflectors.

Table 3.1-4. Differences in Experimental Configurations for Clusters of 2.35 wt% UO_2 Fuel Rods

Experiment identifier	Reference	Reflector
exp5	Bierman et al. 1981	uranium
exp6	Bierman et al. 1981	lead
exp7	Bierman and Clayton 1981	stainless steel

3.1.5 Critical Configurations with 4.31 wt% ^{235}U Enriched UO_2 Rods in Highly Borated Water Lattices

This set of four experiments was performed at the PNL and documented by Durst et al. (1982). These experiments used 4.31 wt% ^{235}U uniformly enriched UO_2 fuel rods arranged in square-pitch, water-moderated lattices of different size with various amounts of boric acid in the moderator. The fuel rods were clad with aluminum and were loaded into polypropylene lattice templates fastened inside a plexiglass tank. The plexiglass tank was surrounded on all four sides by an unborated water reflector and was positioned on top of a plexiglass slab. The borated water was restricted to the water volume inside the plexiglass tank.

Rectangular critical arrays were constructed by sequentially filling rows of the lattice template starting at the plexiglass tank wall. The water level in the tank was held constant by removing an appropriate volume of water as each fuel rod was loaded. These experiments were denoted as "exp8" through "exp11."

3.1.6 Critical Configurations with Neutron Flux Traps

Pacific Northwest Laboratories performed experiments studying the effect of neutron flux traps on criticality. These experiments were documented by Bierman (1990) and served as the source for two configurations modeled with MCNP. These two critical experiments were each composed of four fuel rod arrays arranged in a square and separated by a neutron flux trap region. Each fuel lattice in a given configuration was nearly equal in size. Two polypropylene lattice templates were used to position the fuel rods. The fuel rods were composed of aluminum-clad 4.31 wt% ^{235}U enriched UO_2 fuel. The neutron flux traps were created by positioning two plates of Boral™ between interacting faces of each fuel lattice. The experimental configurations were moderated and closely reflected by full-density water. These experiments are denoted as "exp12" and "exp13."

3.1.7 Electric Power Research Institute 2.35 wt% Enriched Light Water Reactor Fuel Critical Configurations

Criticality experiments were sponsored by Electric Power Research Institute (EPRI) for light water reactor fuel configurations. These were documented by EPRI and subsequently described by Dehart and Bowman (1995). Two critical experiment configurations composed of water-moderated lattices of 2.35 wt% enriched UO_2 fuel rods were modeled with MCNP. The fuel rods were supported in a core structure composed of "eggcrate" type lattice plates with an upper lead shield. The configuration was closely reflected by full-density water laterally and below the fuel. These experiments were designated as "exp14" and "exp15."

3.1.8 Water-Moderated, Lead-Reflected Uranium Dioxide Rod Array

OECD-NEA (1998) LEU-COMP-THERM-027 documents a series of four experiments involving lead-reflected, water-moderated arrays of low-enriched UO_2 fuel rods. The experiments were subcritical approaches extrapolated to critical; the multiplication factor reached was very close to 1.000 (within 0.1%). The experiments were tests of the lead reflector effect. Only the first case was evaluated in CRWMS M&O (1999d). This case consisted of a 14 x 14 array of 4.74 wt% enriched UO_2 fuel rods reflected on four sides by 30-cm-thick lead reflectors with no water gap between the array and the lead reflectors. This experiment was denoted as lct27-1.

3.1.9 Laboratory Critical Experiments from the Urania-Gadolinia: Nuclear Model Development and Critical Experiment Benchmark Report

A number of critical experiments were performed by Babcock and Wilcox for urania fuel incorporating gadolinia as an integral burnable absorber. These experiments were documented in Newman (1984). The configurations modeled with MCNP included critical configurations containing arrangements of 2.46 wt% ^{235}U enriched UO_2 fuel rods, 4.02 wt% ^{235}U enriched UO_2 fuel rods, combination 4 wt% Gd_2O_3 and 96 wt% (1.944 wt% ^{235}U enriched) UO_2 fuel rods, Ag-In-Cd absorber rods, and B_4C absorber rods. The fuel rods were supported by a top and bottom aluminum "eggcrate" type grid plate. The fuel rods rested on an aluminum base plate. The central 45 x 45 array of rod lattice cells was separated into nine 15 x 15 arrays of rod lattice cells. These arrays were intended to simulate pressurized water reactor fuel assembly lattices.

Descriptions of the experimental configurations are shown in Table 3.1-5.

Table 3.1-5. Urania-Gadolinia Critical Experiment Descriptions

Exp. ID[a]	Number of 2.46 wt% ^{235}U fuel rods	Number of 4.02 wt% ^{235}U fuel rods	Number of Gd_2O_3 fuel rods	Number of B_4C rods	Number of Ag-In-Cd rods	Number of void rods	Number of water holes
ugd1	4808	0	0	0	0	0	153
ugd2	4808	0	0	0	16	0	137
ugd3	4788	0	20	0	0	0	153
ugd4	4788	0	20	0	16	0	137
ugd5	4780	0	28	0	0	0	153
ugd6	4780	0	28	0	16	0	137
ugd7	4780	0	28 (Annular)	0	0	0	153
ugd8	4772	0	36	0	0	0	153
ugd9	4772	0	36	0	16	0	137
ugd10	4772	0	36	0	0	16	137
ugd12	3920	888	0	0	0	0	153
ugd13	3920	888	0	16	0	0	137
ugd14	3920	860	28	0	0	0	153
ugd15	3920	860	28	16	0	0	137
ugd16	3920	852	36	0	0	0	153
ugd17	3920	852	36	16	0	0	137
ugd18	3676	944	0	0	0	0	180
ugd19	3676	928	16	0	0	0	180
ugd20	3676	912	32	0	0	0	180

NOTE: [a]ID = Identifier

3.1.10 Saxton UO_2 and PuO_2-UO_2 Critical Configurations

Single- and multi-region uranium and plutonium oxide fueled cores, water moderated, clean, and borated, have been used in a series of critical experiments at the Westinghouse Reactor Evaluation Center in support of the Saxton Plutonium Program. In this series of experiments, criticality was achieved entirely by varying the water level inside the core tank. The fuel used in the experiments were UO_2 fuel with 5.74 wt% ^{235}U enrichment, and mixed oxide (MOX) fuel containing 6.6 wt% PuO_2 and natural enriched UO_2 (p. A-1, Taylor 1965). This work was documented by Taylor (1965) and subsequently described by DeHart and Bowman (1995). This section includes eight single-region configurations and six multi-region configurations. The fuel rods were loaded into a single rectangular array for each critical experiment. The fuel rods were supported by three aluminum grid plates with holes for rod emplacement. The fuel rod type, pitch, array size, moderator height, and boron concentration were adjusted in each LCE. Table 3.1-6 presents a description of the various single-region experiments and Table 3.1-7 presents a description of the multi-region experiments.

Table 3.1-6. Saxton Single-Region Critical Configuration Parameters

Experiment identifier	Fuel	Pitch (cm)	Core configuration	Critical water height (cm)
ssr83	UO_2	1.3208	449 cylindrical	95.25
ssr48	UO_2	1.4224	19 x 19 square	83.71
ssr70	MOX	1.3208	22 x 23 square	84.56
ssr57	MOX	1.4224	19 x 19 square	82.46
ssr27	MOX	1.4224	21 x 21 square	89.70
ssr66	MOX	1.8669	13 x 13 square	70.11
ssr53	MOX	2.0117	12 x 12 square	78.43
ssr74	MOX	2.6416	11 x 11 square	81.17

Table 3.1-7. Saxton Multi-Region Critical Configuration Parameters

Experiment identifier	Core configuration	Critical water height (cm)
smr1	19 x 19 square: 11 x 11 MOX center region; UO_2 outer region	91.07
smr9	19 x 19 square: 11 x 11 MOX center region; UO_2 outer region; Al plate at the fuel interface	92.07
smr5	27 x 27 square: 19 x 19 UO_2 center region; MOX outer region	86.70
smr11	27 x 27 square: 19 x 19 MOX center region; UO_2 outer region; water slot at the region boundary	99.80
smr12	27 x 27 square: 19 x 19 MOX center region; UO_2 outer region; Al slab at the interface	106.35
smr8	27 x 27 square: 19 x 19 MOX center region; UO_2 outer region; L shaped UO_2 insert in MOX region	92.19

3.1.11 Critical Configurations Simulating Light Water Reactor Fuel in Close Proximity Water Storage

Babcock and Wilcox performed experiments simulating neutron multiplication in pool storage racks. These were documented in Baldwin et al. (1979). Twenty such critical configurations, each containing a 3 x 3 array of 14 x 14 fuel rod assemblies, were modeled with MCNP. Two different methods were utilized to support the fuel assemblies in the critical experiment core. The first support method used top and bottom grid plates to hold the fuel rods in place. The second support method used a bottom grid plate and vertical alignment system consisting of locating bars and fastening plates. The gaps between assemblies contained a number B_4C rods and water, stainless steel sheets and water, borated aluminum sheets and water, or only water.

The critical experiment arrays were assembled in an aluminum core tank. The fuel rods were composed of 2.46 wt% ^{235}U enriched UO_2 clad in Type 6061 aluminum. The B_4C rods were aluminum tubes filled with B_4C powder. Six sets of borated aluminum sheets were used in the critical experiments. The soluble boron concentration and moderator heights were adjusted to obtain a critical configuration.

The key parameters which distinguish the twenty critical configurations are shown in Table 3.1-8.

Table 3.1-8. Close Proximity Critical Benchmark Characterization Parameters

Experiment identifier	Assembly spacing, rod pitch[1]	Number of B_4C rods	Metal between unit assemblies
core2	0	0	n/a[2]
core3	1	0	n/a
core4	1	84	n/a
core5	2	64	n/a
core6	2	64	n/a
core7	3	34	n/a
core8	3	34	n/a
core9	4	0	n/a
core10	3	n/a	none
core11	1	n/a	SS[3]
core12	2	n/a	SS
core13	1	n/a	B/Al set 5
core14	1	n/a	B/Al set 4
core15	1	n/a	B/Al set 3
core16	2	n/a	B/Al set 3
core17	1	n/a	B/Al set 2
core18	2	n/a	B/Al set 2
core19	1	n/a	B/Al set 1
core20	2	n/a	B/Al set 1
core21	3	n/a	B/Al set 1

NOTES: [1] number of rod pitches
[2] n/a = not applicable
[3] SS = stainless steel

3.1.12 Electric Power Research Institute Mixed Oxide Critical Configurations

DeHart and Bowman (1995) describe criticality tests with mixed oxide fuel performed for the Electric Power Research Institute. Six critical experiment configurations composed of unborated and borated water moderated lattices of 2 wt% PuO_2 (8 wt% ^{240}Pu)/98 wt% UO_2 (natural) fuel rods were modeled with MCNP. The fuel rods were clad with aluminum and were supported in a core structure composed of "eggcrate" type lattice plates with an upper lead shield. The configurations were closely reflected with full-density water laterally and below the core. These experiments are denoted as “exp22” through “exp27.”

3.1.13 Critical Triangular Lattice of MOX & UO_2 Fuel Rods

Bierman et al. (1984) documented critical experiments performed at PNL incorporating both urania and mixed-oxide (MOX) fuel rods in a triangular lattice. One such experiment, designated “exp34”, contained a triangular lattice of uniformly distributed PuO_2-UO_2 and UO_2

fuel rods. The fuel rods were placed in a uniform distribution with a Pu/^{235}U ratio approximating that of a 20,000 MWd/MTU burnup. Each PuO_2-UO_2 fuel rod was surrounded by six UO_2 fuel rods with a triangular lattice pitch. The fuel rods were supported by three polypropylene lattice plates.

3.1.14 TRIGA (Training, Research, Isotopes, General Atomics) Fuel Rod Experiments

These benchmark experiments documented in Mele et al. (1994) used fresh stainless steel clad TRIGA fuel rods in a TRIGA Mark II reactor. The configuration was a cylindrical water filled reactor with an annular graphite reflector. The fuel elements were arranged in six concentric rings, and were made up of 20 wt% ^{235}U mixed with zirconium hydride. The fuel had a 1.65 hydrogen-zirconium atom ratio. Two experiments of this type were evaluated and are identified as "tri17" and "tri18".

3.1.15 SPERT-D Fuel Experiments

Twenty-three critical experiments involving lattices of SPERT-D fuel elements were performed at Oak Ridge National Laboratories (ORNL). The fuel elements consisted of plates of uranium-aluminum alloy. Each fuel element contained approximately 300 grams (p. 1, OECD-NEA 1998, HEU-MET-THERM-006 [HMT-006]) of ^{235}U in 22 aluminum clad fuel plates. The average enrichment was 93.17 wt% ^{235}U (p. 10, OECD-NEA 1998). Table 3.1-9 provides a listing of the various SPERT-D cases and the experiment identifiers. The reflector and moderator for cases 1 through 18 was demineralized water, and an aqueous solution of uranyl nitrate $U(92.6)O_2(NO_3)_2$ in cases 19 through 23 (p. 10, OECD-NEA 1998, HMT-006).

Table 3.1-9. SPERT-D Fuel Element Critical Experiments

Experiment identifier	Global description	Reflector above fuel (cm) (p. 23, OECD-NEA 1998, HMT-006)
spert1	4 x 3.77 lattice[1], 4.63 kg U^{235}, 0.0 in. spacing	9.825
spert2	4 x 3.16 lattice[1], 3.87 kg U^{235}, 0.25 in. spacing	12.8982
spert3	4 x 3.09 lattice[1], 3.79 kg U^{235}, 0.50 in. spacing	9.8401
spert4	circular, 3.48 kg U^{235}, 0.50 in. spacing	7.3136
spert5	4 x 3.16 lattice[1], 3.87 kg U^{235}, 0.75 in. spacing	17.21
spert6	4 x 3.70 lattice[1], 4.54 kg U^{235}, 1.00 in. spacing	13.853
spert7	5 x 4.03 lattice[1], 6.16 kg U^{235}, 1.25 in. spacing	13.0186
spert8	6 x 5.34 lattice[1], 9.82 kg U^{235}, 1.50 in. spacing	11.3984
spert9	7 x 6.68 lattice[1], 6.16 kg U^{235}, 1.60 in. spacing	12.0359
spert10	4 x 3.2 x 3 lattice[1], 11.78 kg U^{235}, 0.0 in. spacing	9.4309
spert11	3 x 3.36 x 3 lattice[1], 9.28 kg U^{235}, 0.50 in. spacing	12.0994
spert12	4 x 4 x 3 lattice[1], 14.71 kg U^{235}, 1.25 in. spacing	8.0
spert13	slab 16 x 2.32[1], 11.37 kg U^{235}, 1.25 in. spacing	11.6162
spert14	slab 16 x 3[1], 14.71 kg U^{235}, 0.50 in./2.19 in. spacing	7.5728
spert15	slab 16 x 4[1], 19.62 kg U^{235}, 0.50 in./2.56 in. spacing	10.75
spert16	2 slabs 16 x 2[1], 19.62 kg U^{235}, 0.50 in./0.50 in./6.37 in. spacing	12.7351
spert17	slab 4 x 5.0[1] w/ Cd, 6.19 kg U^{235}, 0.0 in./0.75 in. spacing	10.7471
spert18	slab 4 x 7.04[1] w/ Cd, 8.64 kg U^{235}, 0.0 in./0.75 in. spacing	13.8573
spert19	U Nitrate (3.99 g U^{235}/liter) & 3 x 3.09[1], 2.86 kg U^{235}, 0.5 in. spacing, 0.0 g B/liter	6.8208
spert20	U Nitrate (3.99 g U^{235}/liter) & 4 x 4.20[1], 5.15 kg U^{235}, 0.5 in. spacing, 0.389 g B/liter	8.3311
spert21	U Nitrate (3.99 g U^{235}/liter) & 5 x 4.41[1], 6.76 kg U^{235}, 0.5 in. spacing, 0.579 g B/liter	4.6946

Table 3.1-9. SPERT-D Fuel Element Critical Experiments

Experiment identifier	Global description	Reflector above fuel (cm) (p. 23, OECD-NEA 1998, HMT-006)
spert22	U Nitrate (3.99 g U^{235}/liter) & 6 x 4.96[1], 8.90 kg U^{235}, 0.5 in. spacing, 0.773 g B/liter	5.5725
spert23	U Nitrate (3.99 g U^{235}/liter) & 6 x 5.55[1], 10.15 kg U^{235}, 0.5 in. spacing, 0.871 g B/liter	7.7118

NOTE: [1] Dimensions of lattice (number of elements)

3.1.16 Water-Moderated Hexagonally Pitched Lattices of Highly Enriched Fuel Rods of Cross-Shaped Cross Section

A series of critical experiments with water moderated hexagonally pitched lattices of highly enriched fuel rods of cross-shaped cross section was performed over several years in the Russian Research Center (RRC) "Kurchatov Institute". Each of the experiments used UO_2 plus copper fuel, and were taken from the benchmark compilation OECD-NEA (1998), Vol. II. These experiments were categorized under the HEU-Comp-Therm (HCT) class of experiments in the reference. The 28 experiments analyzed under this category consist of the following:

1) Fifteen critical two-zone lattice experiments corresponding to different combinations of inner and peripheral zones of cross-shaped fuel rods at two pitches. Descriptions for these experimental configurations and their experiment identifiers are presented in Table 3.1-10.

2) Four critical configurations of hexagonal lattices of fuel rods with Gd or Sm rods. These experiments consisted of double lattices of fuel rods and absorber rods containing Gd or Sm. Descriptions of these experimental configurations and their experiment identifiers are presented in Table 3.1-11.

3) One critical configuration of hexagonal pitched clusters of lattices of fuel rods with Cu rods. The configuration was arranged with a 5.2-mm pitch and a total of 2257 fuel rods. In the reference, two cases were evaluated: one considered the fuel rod as a cylinder in a simplified model, and the other represented the fuel rod in a detailed cross-shaped model; only the detailed case is represented in this report.

4) Three critical configurations with uniform hexagonal lattices with pitch values of 5.6, 10.0, and 21.13 mm. Descriptions for these experimental configurations and their experiment identifiers are presented in Table 3.1-12.

5) Three critical configurations with double hexagonal lattices of fuel rods and zirconium hydride rods. Descriptions for these experimental configurations and their experiment identifiers are presented in Table 3.1-13. In the reference, two sets of cases were evaluated: one considered the fuel rod as a cylinder in a simplified model, and the other represented a detailed cross-shaped fuel rod model; only the detailed cases are represented in this report.

6) Two critical configurations with double hexagonal lattices of fuel rods and boron carbide rods. Descriptions for these experimental configurations and their experiment identifiers are presented in Table 3.1-14.

Table 3.1-10. Benchmarks for HCT-003 Class of Experiments

Experiment identifier	Configuration description
hct3-1	Center zone: 12.2 mm pitch, 19 rods Outer zone: 6.1 mm pitch, 1390 rods
hct3-2	Center zone: 12.2 mm pitch, 61 rods Outer zone: 6.1 mm pitch, 1182 rods
hct3-3	Center zone: 12.2 mm pitch, 121 rods Outer zone: 6.1 mm pitch, 897 rods
hct3-4	Center zone: 12.2 mm pitch, 199 rods Outer zone: 6.1 mm pitch, 577 rods
hct3-5	Center zone: 12.2 mm pitch, 271 rods Outer zone: 6.1 mm pitch, 325 rods
hct3-6	Center zone: 6.1 mm pitch, 1099 rods Outer zone: 12.2 mm pitch, 167 rods
hct3-7	Center zone: 6.1 mm pitch, 793 rods Outer zone: 12.2 mm pitch, 250 rods
hct3-8	Center zone: 6.1 mm pitch, 757 rods Outer zone: 12.2 mm pitch, 249 rods
hct3-9	Center zone: 6.1 mm pitch, 445 rods Outer zone: 12.2 mm pitch, 319 rods
hct310	Center zone: 6.1 mm pitch, 217 rods Outer zone: 12.2 mm pitch, 372 rods
hct311	Center zone: 6.1 mm pitch, 85 rods Outer zone: 12.2 mm pitch, 415 rods
hct312	Center zone: 18.3 mm pitch, 121 rods Outer zone: 6.1 mm pitch, 985 rods
hct313	Center zone: 18.3 mm pitch, 301 rods Outer zone: 6.1 mm pitch, 426 rods
hct314	Center zone: 6.1 mm pitch, 763 rods Outer zone: 18.3 mm pitch, 186 rods
hct315	Center zone: 6.1 mm pitch, 337 rods Outer zone: 18.3 mm pitch, 325 rods

Table 3.1-11. Benchmarks for HCT-004 Class of Experiments

Experiment identifier	Configuration description
hct4-1	106 Gd rods on 27.54 mm pitch, 2760 fuel rods
hct4-2	55 Gd rods on 36.72 mm pitch, 2520 fuel rods
hct4-3	121 Sm rods on 27.54 mm pitch, 3198 fuel rods
hct4-4	58 Gd rods on 36.72 mm pitch, 2727 fuel rods

Table 3.1-12. Benchmarks for HCT-006 Class of Experiments

Experiment identifier	Configuration description
hct6-t1	1819 fuel rods on a 5.6 mm pitch
hct6-t2	457 fuel rods on a 10.0 mm pitch
hct6-t3	554 fuel rods on a 21.13 mm pitch

Table 3.1-13. Benchmarks for HCT-007 Class of Experiments

Experiment identifier	Configuration description
hct7-4	523 Zr rods on 10.5655 mm pitch, 1064 fuel rods
hct7-5	121 Zr rods on 21.1310 mm pitch, 1400 fuel rods
hct7-6	31 Zr rods on 42.2620 mm pitch, 1484 fuel rods

Table 3.1-14. Benchmarks for HCT-008 Class of Experiments

Experiment identifier	Configuration description
hct8-1	217 B_4C rods (1.0 g B/rod) on 21.2 mm pitch, 3460 fuel rods
hct8-2	169 B_4C rods (3.5 g B/rod) on 26.5 mm pitch, 4130 fuel rods

3.1.17 Critical Experiments of EBOR Fuel Pins in Water

Twenty-one critical experiments involving lattices of EBOR (Experimental Beryllium Oxide Reactor) fuel pins were performed in 1967 at ORNL. The fuel pins consisted of compressed ceramic pellets contained in Hastelloy X-280 tubes. The pellets were a homogeneous mixture of $U(62.4)O_2$ and BeO. Two sets of experiments were conducted. The first set of experiments (total of 15 experiments) consisted of EBOR fuel pins arranged in various lattice configurations moderated and reflected by water (p. 8, OECD-NEA 1998, HEU-COMP-THERM-010 [HCT-010]). The second set (6 experiments) consisted of EBOR fuel pins arranged in various lattice configurations with boron and/or with uranyl nitrate in the water (p. 14, OECD-NEA 1998, HCT-010). Experimental configuration descriptions and identifiers are presented in Table 3.1-15.

Table 3.1-15. Benchmarks for HCT-010 Class of Experiments

Experiment identifier	Surface separation (cm)	Critical number of pins	Critical height of solution above the fuel in the fuel tank (cm)[a]	Solution inside the core
hct101	0.290	222	15.2000	Water
hct102	0.290	223	-50.3000	Water
hct103	0.536	138	30.8000	Water
hct104	0.790	102	-21.3000	Water
hct105	1.046	85	15.2000	Water
hct106	1.046	86	-60.8000	Water
hct107	1.323	78	15.2000	Water
hct108	1.323	79	-39.0000	Water
hct109	1.300	77	-3.90000	Water
hct110	1.554	75	15.2000	Water
hct111	1.554	76	-61.3000	Water
hct112	1.826	77	-43.2000	Water
hct113	2.042	83	-34.1000	Water
hct114	1.544 1.585[b]	96	-10.4000	Water
hct115	1.6477	75	-12.2000	Water
hct116	1.5478	99	19.3675	Water
hct117	1.5478	114	19.3675	Aqueous solution of boron
hct118	1.5478	113	19.3675	Aqueous solution of boron
hct119	1.5478	133	19.3675	Aqueous solution of boron

Table 3.1-15. Benchmarks for HCT-010 Class of Experiments

Experiment identifier	Surface separation (cm)	Critical number of pins	Critical height of solution above the fuel in the fuel tank (cm)[a]	Solution inside the core
hct120	1.5478	83	19.3675	Aqueous solution of uranyl nitrate
hct121	1.5478	133	19.3675	Aqueous solution of uranyl nitrate and boron

NOTES: [a] Negative water heights refer to distance below top of fuel.
[b] The average surface separation was 1.544 cm between pins in the 16-pin direction and 1.585 cm in the 6-pin direction.

3.1.18 Intermediate Heterogeneous Assembly with Highly Enriched Uranium Dioxide and Sand/Water Radial Reflector

An experiment was performed at the RRC "Kurchatov Institute" to investigate accidental sand and water immersion criticality safety of the thermionic intermediate reactor-converter with highly enriched fuel (approximately 96% ^{235}U), zirconium hydride moderator, and end beryllium reflectors (p. 1, OECD-NEA 1998, HEU-COMP-INTER-002 HCI-002). Described in this category are five configurations of the critical assemblies simulating water ingress into different reactor cavities, as well as surrounding the reactor with sand and water. The experimental descriptions of each configuration are presented in Table 3.1-16. Figure 3.1-1 illustrates the thermionic intermediate reactor as simulated. In configurations 1 and 2, wet sand is the reflector material and the rotating drums are filled with wet sand. All cavities are filled with water. The number of drums and the position of the control drums vary. In configurations 3, 4, and 5, the drum channels were removed, water is the radial reflector, and the number of drums and their positions vary.

Table 3.1-16. Benchmarks for HCI-002 Class of Experiments

Experiment identifier	Material		Rotating drums	
	Radial reflector	Core cavities	Total number	Position in assemblies[1]
hci2-1	SiO_2 +H_2O	H_2O	12	CD-5, φ=36° CD-1 ↑,CD-3 ↓, CD-6 ↓, the other CD and all SD ↑
hci2-2	SiO_2 +H_2O	H_2O	11	CD-5, φ=121° CD-3 removed, CD-1, 2, 4, 6 ↓, CD-6 ↓, all SD ↑
hci2-3	H_2O	H_2O	12	CD-5, φ=119.5° the other CD and all SD ↑
hci2-4	H_2O	H_2O	11	CD-5, φ=95.5° CD-3 removed, the other CD and all SD ↑
hci2-5	H_2O	H_2O	10	CD-5, φ=76° CD-3 and SD-3 removed, the other CD and all SD ↑

NOTES: [1] ↑ -Control drums turned out (φ=180°),
↓ -Control drums turned in (φ=0°),
Rotation angle φ is shown in Figure 3.1-1.

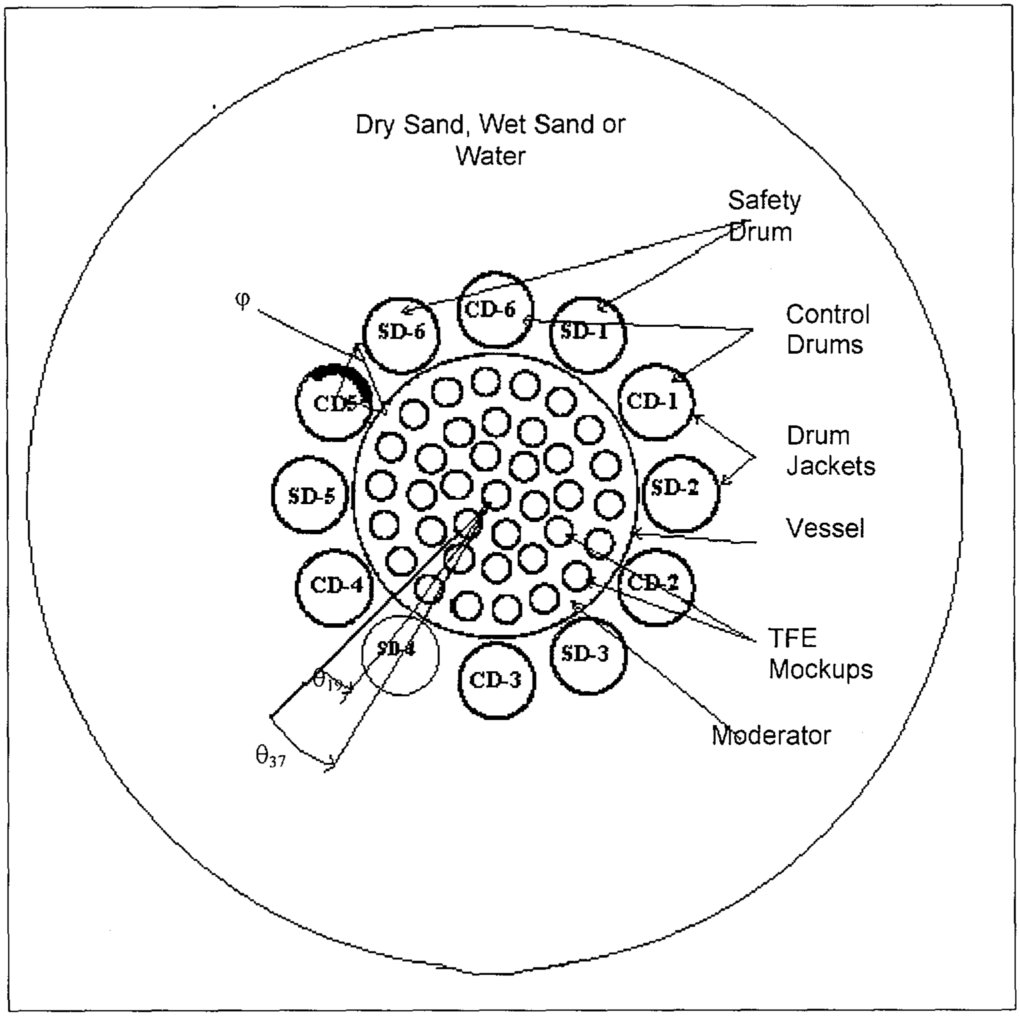

Figure 3.1-1. Cross Section of the Critical Assembly Representation Through the Core Center (φ is the angle of the drum rotation) (p. 3, OECD-NEA [1998], HCI-002)

3.1.19 FFTF Fuel Pin Array Experiments

These experiments were from predicted critical configurations extrapolated from near critical experiments performed by Bierman with plutonium oxide-uranium oxide fuel pins containing about 20 wt% plutonium with light water moderation and reflection (p. 1, OECD-NEA 1998, MIX-COMP-THERM-001 [MCT-001]). Six experiments were evaluated under this classification. These experiments were performed at the PNL Critical Mass Laboratory. The experimental configuration was comprised of an array of fast test reactor fuel pins within a large tank containing water. An axial description of the experimental arrangement is shown on page 4 of OECD-NEA (1998), MCT-001. In the case, the steel grids were replaced with polypropylene grids (p. 144, Bierman et al. 1979). Various lattice pitches were used in the array, resulting in different numbers of fuel rods being required to obtain criticality. The fuel used for the experimental program was a mixture of PuO_2 and UO_2, with the pins comprised of either 19.84 or 24.39 wt% plutonium (p. 141, Bierman et al. 1979). The plutonium contained 11.5 wt% ^{240}Pu, and the uranium in the PuO_2-UO_2 mixture was natural uranium. The remainder of the pin consisted of end-caps, plenum, and other types of hardware (e.g., natural UO_2 insulators and

Inconel reflectors). Table 3.1-17 provides a brief description of these experiments along with their experiment identifier names.

Table 3.1-17. FFTF Bierman Array Critical Experiments

Experiment identifier	Configuration description
fftf001	18 pin lattice width, 1.2588 cm lattice spacing, 279 total pins
fftf003r	36 pin lattice width, 0.7671 cm lattice spacing, 1037 total pins
fftf004	18 pin lattice width, 1.5342 cm lattice spacing, 205 total pins
fftf005	28 pin lattice width, 0.9525 cm lattice spacing, 605 total pins
fftf006	14 pin lattice width, 1.9050 cm lattice spacing, 162 total pins
fftf029	28 pin lattice width, 0.9677 cm lattice spacing, 580 total pins

3.1.20 High Enriched Uranium Metal Fast (HMF) Systems

Reactivity calculations involving Fast Metal Fuel are presented in this section. In the two following sections, a series of critical experiments performed by the Institute for Experimental Physics of the Russian Federal Nuclear Center at Arzamas-16 and the Institute for Technical Physics of the Russian Federal Nuclear Center at Chelyabinsk-70 are described. Detailed experimental configurations and material compositions for these experiments are included in OECD-NEA (1998). The series of experiments studied in this report include uranium metal systems of intermediate enrichment, high enrichment, and plutonium systems reflected by the following materials: depleted uranium, steel, aluminum, graphite, and polyethylene. An experiment identifier for each configuration is provided for subsequent reference in this document in Tables 3.1-18 and 3.1-19.

Table 3.1-18. HMF Russian Criticality Safety Benchmark Experiments

Reference location	^{235}U wt%	Core dimensions (cm)		Reflector material	Reflector thickness (cm)	Experiment identifier
		Radius	Height			
HEU-MET-FAST-001	93.71	8.74	--	None	--	HMF1G
HEU-MET-FAST-003	93.5	6.463	--	Nickel	20.32	HMF3Ni
HEU-MET-FAST-008	90	10.150	--	None	--	HMF8
HEU-MET-FAST-011	90	7.550	--	Polyethylene	13.230	HMF11
HEU-MET-FAST-012	90	9.150	--	Aluminum	0.850, 2.850	HMF12
HEU-MET-FAST-013	90	8.350	--	Steel	3.650	HMF13
HEU-MET-FAST-014	90	7.750	--	Depleted U	4.650	HMF14
HEU-MET-FAST-015	96	9.995	11.130	None	--	HMF15
HEU-MET-FAST-018	90	9.150	--	None	--	HMF18
HEU-MET-FAST-019	90	9.150	--	Graphite	3.450	HMF19
HEU-MET-FAST-020	90	8.350	--	Polyethylene	1.450	HMF20
HEU-MET-FAST-021	90	7.550	--	Steel	9.700	HMF21
HEU-MET-FAST-022	90	8.350	--	Duralumin	3.900	HMF22
HEU-MET-FAST-024	90	7.550	--	Steel, polyethylene	0.850, 2.850	HMF24
HEU-MET-FAST-028	93.27	6.116	--	Natural uranium	18.009	HMF28

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3.1.21 Plutonium Metal Fast (PMF) Systems

Table 3.1-19. PMF Russian Criticality Safety Benchmark Experiments

Handbook identifier	^{239}Pu at%[1]	Core dimensions (cm)		Reflector material	Reflector thickness (cm)	Case name
		Radius	Height			
PU-MET-FAST-020	89	5.350	--	Depleted U	7.650	PMF20
PU-MET-FAST-022	98	6.670	--	None	--	PMF22
PU-MET-FAST-023	98	6.000	--	Graphite	2.350	PMF23
PU-MET-FAST-024	98	6.000	--	Polyethylene	1.550	PMF24
PU-MET-FAST-025	98	6.000	--	Steel	1.550	PMF25
PU-MET-FAST-026	98	5.350	--	Steel	11.900	PMF26
PU-MET-FAST-027	89	5.350	--	Polyethylene	5.580	PMF27
PU-MET-FAST-028	89	5.350	--	Steel	19.650	PMF28
PU-MET-FAST-029	88	6.670	--	None	--	PMF29
PU-MET-FAST-030	88	4.660	--	Graphite	4.490	PMF30
PU-MET-FAST-031	88	4.660	--	Polyethylene	3.690	PMF31
PU-MET-FAST-032	88	4.660	--	Steel	4.490	PMF32

Volume I, OECD-NEA 1998

NOTE: [1] at% = atom percent

3.1.22 Intermediate Enriched Uranium Metal Fast (IMF) Systems

3.1.22.1 The Early Jemima Experiments: Bare Cylindrical Configurations of Enriched and Natural Uranium

The early Jemima experiments, performed at the Pajarito critical assembly facility at Los Alamos (1952-1954), was to determine critical conditions for bare uranium cylinders of intermediate enrichment. Vertical cylindrical columns were constructed by stacking thin disks of enriched uranium (Oralloy, or Oy, 93.4 wt% ^{235}U) and natural uranium (Tuballoy, or Tu) in different orders. A total of five critical cylindrical configurations of uranium disks, partial disks (in the shape of 45° circular sectors), and layers of rectangular blocks were assembled. Documentation and detailed drawings of these experiments are shown on pages 16 through 37 of OECD-NEA (1998) IEU-MET-FAST-001 (IMF-001). For the four experiments, the reference listed detailed and idealized cases. Only the results of the detailed cases were reported in this report. The cases are identified as "IMF1-1", "IMF1-2", "IMF1-3", and "IMF1-4".

3.1.22.2 Natural Uranium Reflected Assembly of Enriched and Natural Uranium Plates

This critical experiment was a cylindrical assembly with a core of alternating plates of enriched and natural uranium surrounded by a natural uranium reflector. The core average enrichment was 16 wt% ^{235}U (p. 1, OECD-NEA 1998, IMF-002). This experiment was performed at the Los Alamos Pajarito critical assembly facility and can be considered an extension of the earlier Jemima experiments that determined the critical conditions of bare natural and enriched uranium disks of combined intermediate enrichments (29-94 wt% ^{235}U; p. 1, OECD-NEA 1998, IMF-002). This experiment was designated as "imf2-1".

3.1.22.3 Spherical Assembly of 36 wt% ^{235}U

A series of critical experiments with a spherical assembly of 36 wt% ^{235}U was performed over the course of several years (1993-1996) by the Institute for Experimental Physics of the Russian Federal Nuclear Center. The assembly core included different layers of fissile material (lower core limit) and could be covered by different layers of graphite, steel, or duralumin (upper core limit). The upper core layers are the reflector layers. Table 3.1-20 shows the descriptions for the experimental configurations. More detailed descriptions for this set of experiments is provided in Sections 3.1.22.4 through 3.1.22.8.

Table 3.1-20. IMF Experimental Description

Reference identifier	Description	Experiment identifier
IEU-MET-FAST-003	Bare spherical assembly (detailed)	IMF3-1
IEU-MET-FAST-004	Graphite reflected spherical assembly (detailed)	IMF4-1
IEU-MET-FAST-005	Steel reflected spherical assembly (detailed)	IMF5-1
IEU-MET-FAST-006	Duralumin reflected spherical assembly (detailed)	IMF6-1
IEU-MET-FAST-008	Uranium depleted reflected spherical assembly	IMF8-1

p. 14, OECD-NEA (1998), IEU-MET-FAST-003 (IMF-003)
p. 13, OECD-NEA (1998), IEU-MET-FAST-004 (IMF-004)
p. 13, OECD-NEA (1998), IEU-MET-FAST-005 (IMF-005)
p. 14, OECD-NEA (1998), IEU-MET-FAST-006 (IMF-006)
p. 14, OECD-NEA (1998), IEU-MET-FAST-007 (IMF-008)

3.1.22.4 Bare Spherical Assembly of 36 wt% ^{235}U

A criticality measurement of a bare spherical assembly of 36 wt% ^{235}U was performed. The assembly had a core of 36 wt% ^{235}U incorporating 10 spherical layers of fissile material. Characteristics for these core layers are presented in Table 3.1-21. This case is identified as "imf3-1".

Table 3.1-21. Dimensions and Mass Characteristics of the IMF-003 Critical Assembly

Detailed 10 shell case			
Layer no.	Radius (cm)		Layer mass (g)
	Inner	Outer	
1	0.00	2.00	623.29
2	2.00	6.00	16256
3	6.00	7.55	16630
4	7.55	9.15	25864
5	9.15	11.00	43638
6	11.00	12.25	39021
7	12.25	13.25	37156
8	13.25	14.00	31449
9	14.00	15.00	48363
10	15.00	15.324	17128

p. 12, OECD-NEA 1998, IMF-003

3.1.22.5 Graphite-Reflected Spherical Assembly of 36 wt% ^{235}U

A criticality measurement of a graphite reflected spherical assembly of 36 wt% ^{235}U was performed. The assembly core had a central cavity of 2-cm radius and incorporated 7 spherical layers of fissile material. Characteristics of these core layers are presented in Table 3.1-22. The graphite reflector was a single spherical layer with an outer radius of 17.2 cm. This case is identified as "imf4-1".

Table 3.1-22. Dimensions and Mass Characteristics of the IMF-004 Critical Assembly

Detailed 8 shell case			
Layer no.	Radius (cm)		Layer mass (g)
	Inner	Outer	
Fuel			
1	2.788	6.000	15267
2	6.000	7.550	16791
3	7.550	9.150	26169
4	9.150	11.000	44087
5	11.000	12.250	39542
6	12.250	13.250	37804
7	13.250	14.000	32214
Graphite reflector			
1	14.000	17.200	15222

p. 11, OECD-NEA 1998, IMF-004

3.1.22.6 Steel-Reflected Spherical Assembly of 36 wt% ^{235}U

A criticality measurement for a steel reflected spherical assembly of 36 wt% ^{235}U was performed. The assembly core had a central cavity of 2.686-cm radius and incorporated 6 spherical layers of fissile material. Characteristics of these core layers are presented in Table 3.1-23. Five spherical layers that differ slightly in density represented the steel reflector. The outermost layer had an outer radius of 21.5 cm. The reflector was represented in as 2 spherical layers that differ slightly in density. This case was identified as "imf5-1".

Table 3.1-23. Dimension and Mass Characteristics of the IMF-005 Critical Assembly

Detailed 8 shell case			
Layer no.	Radius (cm)		Layer mass (g)
	Inner	Outer	
Fuel			
1	2.686	6.000	15366
2	6.000	7.550	16630
3	7.550	9.150	25864
4	9.150	11.000	43638
5	11.000	12.250	39021
6	12.250	13.250	37156
Steel reflector			
1	13.250	15.000	32860
2	15.000	21.500	208500

p. 11, OECD-NEA 1998, IMF-005

3.1.22.7 Duralumin-Reflected Spherical Assembly of 36 wt% ^{235}U

A criticality measurement for a duralumin reflected spherical assembly of 36 wt% ^{235}U was performed. The assembly core had a central cavity of 2.1 cm radius and incorporated 6 spherical layers of fissile material. Characteristics of these core layers are presented in Table 3.1-24. Seven spherical layers that differ slightly in density represented the duralumin reflector. The outermost layer had an outer radius of 25 cm. The reflector was represented as 2 spherical layers that differ slightly in density. This case was designated as "imf6-1".

Table 3.1-24. Dimensions and Mass Characteristics of the IMF-006 Critical Assembly

Detailed 8 shell case			
Layer no.	Radius (cm)		Layer mass (g)
	Inner	Outer	
Fuel			
1	2.10[a]	6.00	16157
2	6.00	7.55	16630
3	7.55	9.15	25864
4	9.15	11.00	43638
5	11.00	12.25	39021
6	12.25	13.25	37156
Duralumin reflector			
1	13.25	15.00	11150
2	15.00	25.00	129550

pp. 11, 12, 13, OECD-NEA 1998, IMF-006

3.1.22.8 Depleted Uranium-Reflected Spherical Assembly of 36 wt% ^{235}U

A criticality measurement for a depleted uranium reflected spherical assembly of 36 wt% ^{235}U was performed. The assembly core had a central cavity of 2 cm radius and incorporated 6 spherical layers of fissile material. Characteristics of these core layers are presented in Table 3.1-25. Three spherical layers that differ slightly in density represented the depleted uranium reflector. The outermost layer had an outer radius of 16.5 cm. This case was designated as "imf8-1".

Table 3.1-25. Dimensions and Mass Characteristics of the IMF-008 Critical Assembly

Detailed 9 shells case			
Layer no.	Radius (cm)		Layer mass (g)
	Inner	Outer	
Fuel			
0	1.40[a]	2.00	395.7
1	2.00	6.00	16256
2	6.00	7.55	16630
3	7.55	9.15	25864
4	9.15	11.00	43638
5	11.00	12.25	39021
6	12.25	13.25	37156
Depleted uranium reflector			
1	13.25	14.00	31510
2	14.00	15.00	48390
3	15.00	16.50	85890

p. 12, OECD-NEA 1998, IMF-008

3.2 HOMOGENEOUS SOLUTION EXPERIMENTS

The LCEs presented in this section represent solutions containing uranium, plutonium, or both uranium and plutonium. Each of the LCE configurations described in this section have been analyzed with the MCNP code system using the cross section sets previously described in Section 2.2 of this document. An experiment identifier for each configuration is provided for subsequent reference in this document. With a few exceptions that are noted in the text, the vast majority of the assessed benchmarks come from the OECD compilation (OECD-NEA 1998).

The following sections briefly describe the LCEs according to the grouping in which the results are presented.

3.2.1 Mixed Plutonium and Natural Uranium Nitrate Solutions

The experiments involving plutonium and uranium with naturally occurring isotopic ratios are from OECD-NEA (1998), Volume VI and are listed in Table 3.2-1.

Table 3.2-1. Benchmark Problem Summary for Configurations Incorporating Mixed Plutonium and Natural Uranium Nitrate Solutions

Reference Identifier	Experiment Identifier
MIX-SOL-THERM-001	PNL3187
	PNL3391
	PNL3492
	PNL3593
	PNL3694
	PNL3795
	PNL3896
	PNL3897
	PNL3898
	PNL3808
	PNL3999
	PNL5300
MIX-SOL-THERM-002	PNL1158
	PNL1159
	PNL1161
MIX-SOL-THERM-003	awre1
	awre2
	awre3
	awre4
	awre5
	awre6
	awre7
	awre8
	awre9
	awre10
MIX-SOL-THERM-004	PNL1577
	PNL1678
	PNL1783
	PNL1868
	PNL1969
	PNL2070
	PNL2565
	PNL2666
	PNL2767

3.2.2 Plutonium Nitrate Solutions

The experiments involving plutonium are from OECD-NEA (1998), Volume I and are listed in Table 3.2-2.

Table 3.2-2. Benchmark Problem Summary for Configurations Incorporating Plutonium Nitrate Solutions

Reference Identifier	Experiment Identifier
PU-SOL-THERM-001	pust1t1
	pust1t2
	pust1t3
	pust1t4
	pust1t5
	pust1t6
PU-SOL-THERM-003	pu003-1
	pu003-2
	pu003-3
	pu003-4
	pu003-5
	pu003-6
	pu003-7
	pu003-8
PU-SOL-THERM-004	pu004-1
	pu004-2
	pu004-3
	pu004-4
	pu004-5
	pu004-6
	pu004-7
	pu004-8
	pu004-9
	pu04-10
	pu04-11
	pu04-12
	pu04-13
PU-SOL-THERM-005	pu005-1
	pu005-2
	pu005-3
	pu005-4
	pu005-5
	pu005-6
	pu005-7
	pu005-8
	pu005-9
PU-SOL-THERM-007	pu007-2
	pu007-3
	pu007-5
	pu007-6
	pu007-7
	pu007-8
	pu007-9
	pu07-10
PU-SOL-THERM-009	pust9-1
	pust9-2
	pust9-3
PU-SOL-THERM-0010	pu10091
	pu10092
	pu10093
	pu10111

Table 3.2-2. Benchmark Problem Summary for Configurations Incorporating Plutonium Nitrate Solutions

Reference Identifier	Experiment Identifier
	pu10112
	pu10113
	pu10114
	pu10115
	pu10116
	pu10117
	pu10121
	pu10122
	pu10123
	pu10124
PU-SOL-THERM-0011	pu11161
	pu11162
	pu11163
	pu11164
	pu11165
	pu11181
	pu11182
	pu11183
	pu11184
	pu11185
	pu11186
	pu11187

3.2.3 Highly Enriched Uranium Nitrate Solutions

The experiments involving highly enriched uranium are from OECD-NEA (1998), Volume II and are listed in Table 3.2-3.

Table 3.2-3. Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions

Reference Identifier	Experiment Identifier
HEU-SOL-THERM-001	hest1-1
	hest1-2
	hest1-3
	hest1-4
	hest1-5
	hest1-6
	hest1-7
	hest1-8
	hest1-9
	hest110
HEU-SOL-THERM-002	hest2-1
	hest2-2
	hest2-3
	hest2-4
	hest2-5
	hest2-6
	hest2-7
	hest2-8
	hest2-9
	hest2-10
	hest2-11
	hest2-12

Table 3.2-3. Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions

Reference Identifier	Experiment Identifier
	hest2-13
	hest2-14
HEU-SOL-THERM-003	heust31
	heust32
	heust33
	heust34
	heust35
	heust36
	heust37
	heust38
	heust39
	hest310
	hest311
	hest312
	hest313
	hest314
	hest315
	hest316
	hest317
	hest318
	hest319
HEU-SOL-THERM-007	heust71
	heust72
	heust73
	heust74
	heust75
	heust76
	heust77
	heust78
	heust79
	hest710
	hest711
	hest712
	hest713
	hest714
	hest715
	hest716
	hest717
HEU-SOL-THERM-008	heust81
	heust83
	heust86
	heust89
	hest813
HEU-SOL-THERM-013	hest131
	hest132
	hest133
	hest134
HEU-SOL-THERM-0014	hest141
	hest142
	hest143
HEU-SOL-THERM-0015	hest151
	hest152
	hest153
	hest154
	hest155

Table 3.2-3. Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions

Reference Identifier	Experiment Identifier
HEU-SOL-THERM-0016	hest161
	hest162
	hest163
HEU-SOL-THERM-0017	hest171
	hest172
	hest173
	hest174
	hest175
	hest176
	hest177
	hest178
HEU-SOL-THERM-0018	hest181
	hest182
	hest183
	hest184
	hest185
	hest186
	hest187
	hest188
	hest189
	hst1810
	hst1811
	hst1812
HEU-SOL-THERM-0019	hest191
	hest192
	hest193
HEU-SOL-THERM-012	hst-121
HEU-SOL-THERM-032	hst-321

3.2.4 Intermediate-Enrichment Uranium Solutions

The experiments involving intermediate-enrichment uranium are from OECD-NEA (1998), Volume III. All involve arrays of polyethylene-moderated U(30)F_4-polytetraflouroethlyene one-inch cubes. These experiments are denoted as IECT101 through IECT129.

3.2.5 Intermediate-Enriched Uranium Nitrate Solutions

A series of critical experiments with aqueous uranyl nitrate solutions with uranium enriched to 10 wt% ^{235}U was performed at the Solution Critical Facility of the Institute of Physics and Power Engineering, Obninsk, Russia. These experiments are from (p.12, OECD-NEA 1998, LEU-SOL-THERM-003 [LST-003]). Spheres with outer diameters of 66 cm, 88 cm, and 120 cm were used. Experiments differed from one another in geometry, size, and in uranium concentration in the solution. These experiments are denoted as lst3-1 through lst3-9.

3.2.6 Intermediate-Enriched Uranyl Sulfate Solutions

These experiments were performed at the RRC "Kurchatov Institute" in 1980-1981, and were to investigate nuclear safety issues for a special-purpose compact reactor with an aqueous solution of uranyl-sulphate (~20.9 at% ^{235}U) and graphite reflector. These experiments are from (p. 1, OECD-NEA 1998, IEU-SOL-THERM-001 [IST-001]). These experiments are denoted as cases "ist1-1" through "ist1-4".

3.2.7 Low-Enrichment Uranium Solutions

The first set of experiments involving low-enrichment uranium are from OECD-NEA (1998), Volume IV, the second set (case prefix "LEUJ") are from work at the Japan Atomic Energy Research Institute (Miyoshi et al. 1997), and the third set (case prefix SPHU9) are cases that look at UO_3-H_2O critical solutions (p. 43, Bierman et al. 1984). These problems are listed in Table 3.2-4.

Table 3.2-4. Benchmark Problem Summary for Configurations Incorporating Low-Enrichment Uranium Solutions

Reference Identifier	Experiment Identifier
LEU-SOL-THERM-002	LEUST21
	LEUST22
	LEUST23
JAERI	LEUJA01
	LEUJA29
	LEUJA33
	LEUJA34
	LEUJA46
	LEUJA51
	LEUJA54
	LEUJA14
	LEUJA30
	LEUJA32
	LEUJA36
	LEUJA49
SPHU9	SPHU9A
	SPHU9B
	SPHU9C
	SPHU9D
	SPHU9E
	SPHU9F
	SPHU9G
	SPHU9H
	SPHU9I
	SPHU9J
	SPHU9K
	SPHU9L

3.2.8 Low Enriched Uranyl Flouride Solutions

This experiment involved an aqueous solution of about 5 wt% enriched uranyl fluoride and is taken from OECD-NEA (1998), LEU-SOL-THERM-001 (LST-001). This experiment used the SHEBA-II (Solution High Energy Burst Assembly-II), which is a critical assembly experiment that was operated at the Los Alamos Critical Experiments Facility. This experiment is identified as lst1-1.

3.2.9 ^{233}U Fuel Homogeneous Criticals

The experiments involving ^{233}U Fuel are from OECD-NEA (1998), Volume V. All involve spheres of enriched ^{233}U Fuel. The first ten are fast-metal systems. These experiments are denoted as u2331a through u2336a. The other six are thermal solution systems. These experiments are denoted as u233s1 through u233s6.

4. LCE ANALYSES RESULTS

This section tabulates the MCNP k_{eff} results from the three different cross section sets and the average energy of a neutron causing fission (AENCF) results for the LCEs from CRWMS M&O (1999b), CRWMS M&O (1999c), and CRWMS M&O (1999d) according to experimental similarities. The cross section sets were the evaluated nuclear data file (ENDF)/B-V, ENDF/B-VI, and a combination of the two as selected in CRWMS M&O (1998c) referred to as the WPO Selected. Based on the AENCF values, the systems were divided into the following categories: Fast (1.0 Mev ≤ AENCF); Intermediate (0.1 Mev ≤ AENCF ≤ 1.0 Mev); and Thermal (AENCF ≤ 0.1 Mev). The AENCF value is the average energy of all fissions – fast, thermal, and intermediate – that occur in a given configuration, and are calculated from the MCNP outputs as described on page 7 of CRWMS M&O (1999c). It should be noted that due to the AENCF being an average over all energies, a system that has a small number of very high energy neutrons causing fission and a lot of low energy neutrons causing fission may have the same AENCF as a system that has many intermediate energy neutrons causing fission. Thus, this grouping structure is used for illustrative purposes only, and should not be considered the same as traditional spectrum nomenclature for fast, thermal, and intermediate systems.

In the following sections the index in the plots refers to the number (#) designation in the tables for each case.

4.1 LATTICE CRITICALS

Tables 4.1-1 through 4.1-10 and Figures 4.1-1 through 4.1-10 present the results for the LCEs according to the following distinct experimental classifications:

- Table and Figure 4.1-1: Moderated lattices containing mixed oxide fuel (Thermal System)
- Table and Figure 4.1-2: Moderated lattices containing mixed oxide fuel (Intermediate System)
- Table and Figure 4.1-3: Moderated lattices containing HEU fuel pins (Thermal System)
- Table and Figure 4.1-4: Moderated lattices containing HEU fuel pins (Intermediate System)
- Table and Figure 4.1-5: Moderated lattices containing HEU fuel plates (Thermal System)
- Table and Figure 4.1-6: Moderated single-zone lattices containing HEU cruciform fuel pins (Thermal System)
- Table and Figure 4.1-7: Moderated dual-zone lattices containing HEU cruciform fuel pins (Thermal System)
- Table and Figure 4.1-8: Moderated lattices containing IEU fuel pins (Thermal System)
- Table and Figure 4.1-9: Moderated lattices containing LEU fuel pins (Thermal System)
- Table and Figure 4.1-10: Moderated lattices containing LEU fuel pins (Intermediate System)

Future revisions of this report may include additional LCEs.

The tables include the calculated values for k_{eff}, standard deviation (sigma), and the average energy of a neutron causing fission (AENCF). The values are all taken from the MCNP output files as described in CRWMS M&O (1999b), CRWMS M&O (1999c), and CRWMS M&O (1999d).

Table 4.1-1. Mixed Oxide Fuel Pin Lattice Critical Experiments (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	fftf004[a]	1.00635	0.00361	0.0819	0.99333	0.00348	0.0816	1.00137	0.00337	0.0779
2	fftf006[a]	1.00558	0.00313	0.0609	1.01291	0.00285	0.0635	0.99947	0.00313	0.0646

NOTE: [a] Values based on TBV-1371

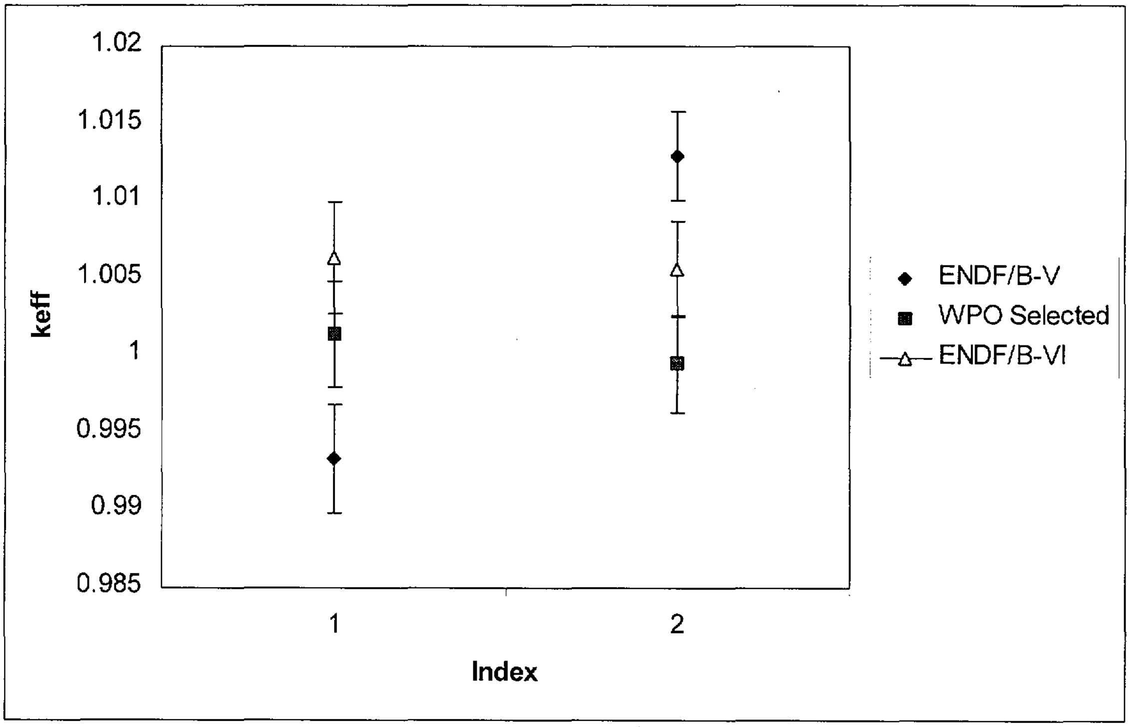

Figure 4.1-1. Mixed Oxide Fuel Pin Lattice Critical Experiments (Thermal)

Table 4.1-2. Mixed Oxide Fuel Pin Lattice Critical Experiments (Intermediate)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	exp22e5	0.99624	0.00174	0.25557	0.99574	0.00153	0.2559	0.99267	0.00161	0.25764
2	exp23e5	1.0005	0.00169	0.27397	1.00004	0.00157	0.27645	0.99307	0.00176	0.2732
3	exp24e5	1.00302	0.00171	0.16128	1.00819	0.00177	0.16053	0.99653	0.00176	0.16065
4	exp25e5	1.00835	0.00161	0.18944	1.00635	0.00161	0.18898	1.00214	0.00167	0.19096
5	exp26e5	1.00709	0.0016	0.13192	1.00864	0.00158	0.13287	0.999	0.00162	0.13166
6	exp27e5	1.00752	0.00155	0.15372	1.0045	0.0016	0.15299	1.00186	0.00155	0.15393
7	fftf001	1.00557	0.00355	0.1015	1.00567	0.00342	0.1044	0.99485	0.00344	0.1043
8	fftf003r	0.99049	0.00276	0.2453	0.98913	0.00307	0.2447	0.99388	0.00371	0.2388
9	fftf005	1.00373	0.0031	0.1728	0.99582	0.00361	0.1717	0.99835	0.00316	0.1727
10	fftf0029	0.99776	0.00278	0.1647	0.99891	0.00323	0.1694	0.9985	0.0032	0.1661
11	smr1	0.99783	0.00073	0.1715	0.99627	0.00072	0.1707	0.99302	0.00072	0.1705
12	smr5	0.99349	0.00073	0.1919	0.99442	0.00072	0.1928	0.99087	0.00074	0.1919
13	smr8	0.99956	0.00068	0.2051	0.99861	0.00064	0.2045	0.99444	0.00058	0.2037
14	smr9	0.99683	0.00078	0.1673	0.99468	0.00072	0.1683	0.99151	0.00074	0.1667
15	smr11	0.99783	0.00078	0.0205	0.99833	0.00072	0.2049	0.99247	0.00079	0.2027
16	smr12	0.99992	0.0008	0.2049	0.99869	0.00073	0.2041	0.99358	0.00077	0.2024
17	ssr27	0.99881	0.00082	0.2015	0.99728	0.00077	0.2013	0.99195	0.00077	0.2017
18	ssr53	1.00454	0.00066	0.1065	1.0034	0.00065	0.1063	0.99503	0.00074	0.1064
19	ssr57	0.99807	0.00075	0.1938	0.99673	0.00076	0.1933	0.99009	0.00072	0.1928
20	ssr66	1.00308	0.00073	0.1183	1.00272	0.00074	0.1182	0.99383	0.00073	0.1189
21	ssr70	0.99543	0.00072	0.2295	0.99369	0.00074	0.2294	0.98936	0.00071	0.229
22	ssr74	1.00505	0.00068	0.079	1.00551	0.00066	0.0799	0.99858	0.00072	0.08
23	exp34	0.9875	0.00168	0.37762	0.98999	0.00154	0.37796	0.99406	0.00152	0.37363

NOTE: Index numbers 1 through 6 carry TBV-1368, index numbers 7 through 10 carry TBV-1371, and index numbers 11 through 23 carry TBV-1359 and TBV-1368

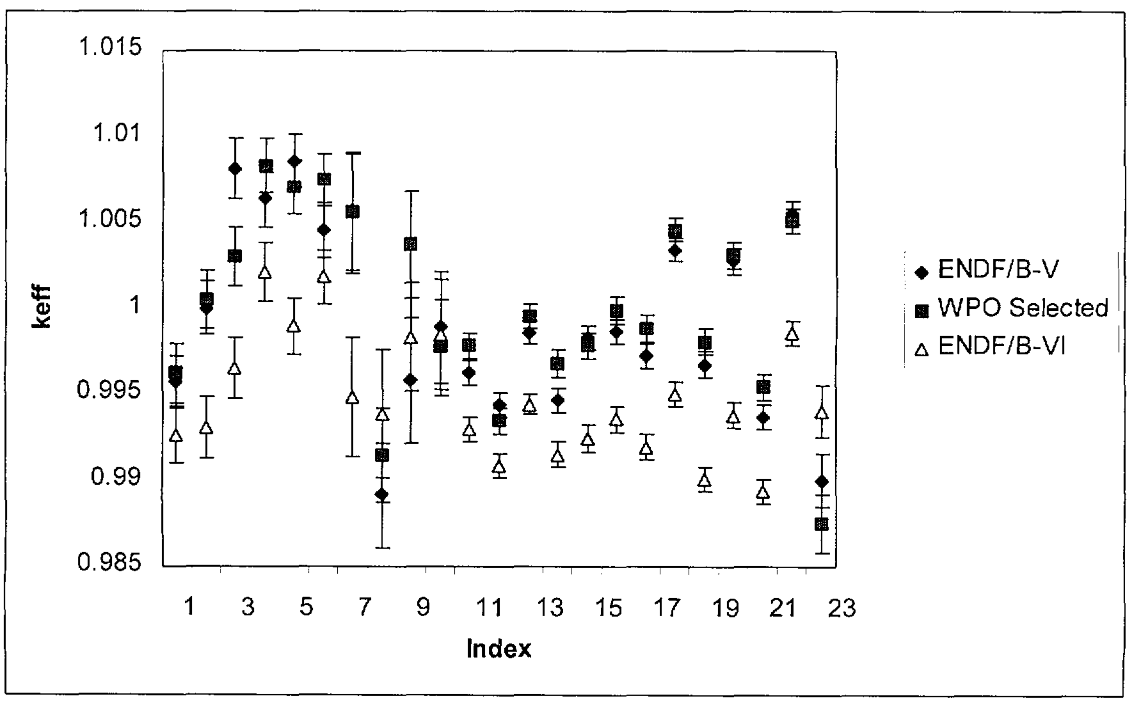

Figure 4.1-2. Mixed Oxide Fuel Pin Lattice Critical Experiments (Intermediate)

Table 4.1-3. High Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	hct101	0.98849	0.00121	0.0793	0.98716	0.00118	0.0797	0.98784	0.00112	0.0783
2	hct102	0.98532	0.00116	0.0799	0.98361	0.00115	0.0797	0.98632	0.00117	0.08
3	hct103	0.99428	0.00127	0.0555	0.99328	0.00125	0.0557	0.98998	0.00117	0.0551
4	hct104	0.99478	0.00119	0.0439	0.99468	0.00125	0.0431	0.99397	0.00126	0.0423
5	hct105	0.99679	0.00115	0.0357	0.99734	0.00119	0.036	0.99568	0.00113	0.0358
6	hct106	0.99644	0.00113	0.0359	0.99239	0.00121	0.0361	0.99419	0.00113	0.0358
7	hct107	0.99679	0.00115	0.0357	0.99976	0.00105	0.0308	0.99841	0.00113	0.0307
8	hct108	1.00263	0.00108	0.032	0.9991	0.00122	0.0315	0.99853	0.00121	0.0309
9	hct109	0.99882	0.00113	0.0316	0.99777	0.00109	0.0314	0.99798	0.00116	0.031
10	hct110	1.00461	0.00106	0.0288	1.00198	0.00104	0.0287	1.00261	0.00102	0.0282
11	hct111	1.00002	0.00115	0.0285	0.99913	0.00109	0.0291	0.99882	0.00111	0.0283
12	hct112	1.00103	0.00111	0.0262	1.00052	0.00105	0.0262	0.99642	0.00107	0.0257
13	hct113	1.00366	0.00102	0.0252	1.00237	0.001	0.025	1.0006	0.001	0.0244
14	hct114	1.00135	0.00114	0.0284	1.00112	0.00112	0.0281	0.99823	0.00113	0.028
15	hct115	1.00368	0.00113	0.0288	1.00059	0.00113	0.0289	0.99844	0.00115	0.0286
16	hct116	1.00594	0.00109	0.0282	1.00378	0.00114	0.0289	0.99892	0.00115	0.0281
17	hct117	1.00448	0.00107	0.0286	1.00377	0.0011	0.0293	1.00117	0.00124	0.0281
18	hct118	1.00267	0.00116	0.0288	1.0018	0.00117	0.0287	0.99989	0.0012	0.0283
19	hct119	1.0037	0.00108	0.0296	1.00151	0.00105	0.0295	0.99935	0.00126	0.0294
20	hct120	1.00489	0.00101	0.0229	1.0024	0.001	0.0234	1.00295	0.00096	0.0228
21	hct121	1.00342	0.00115	0.0277	1.00416	0.0011	0.0278	1.0025	0.00111	0.0272

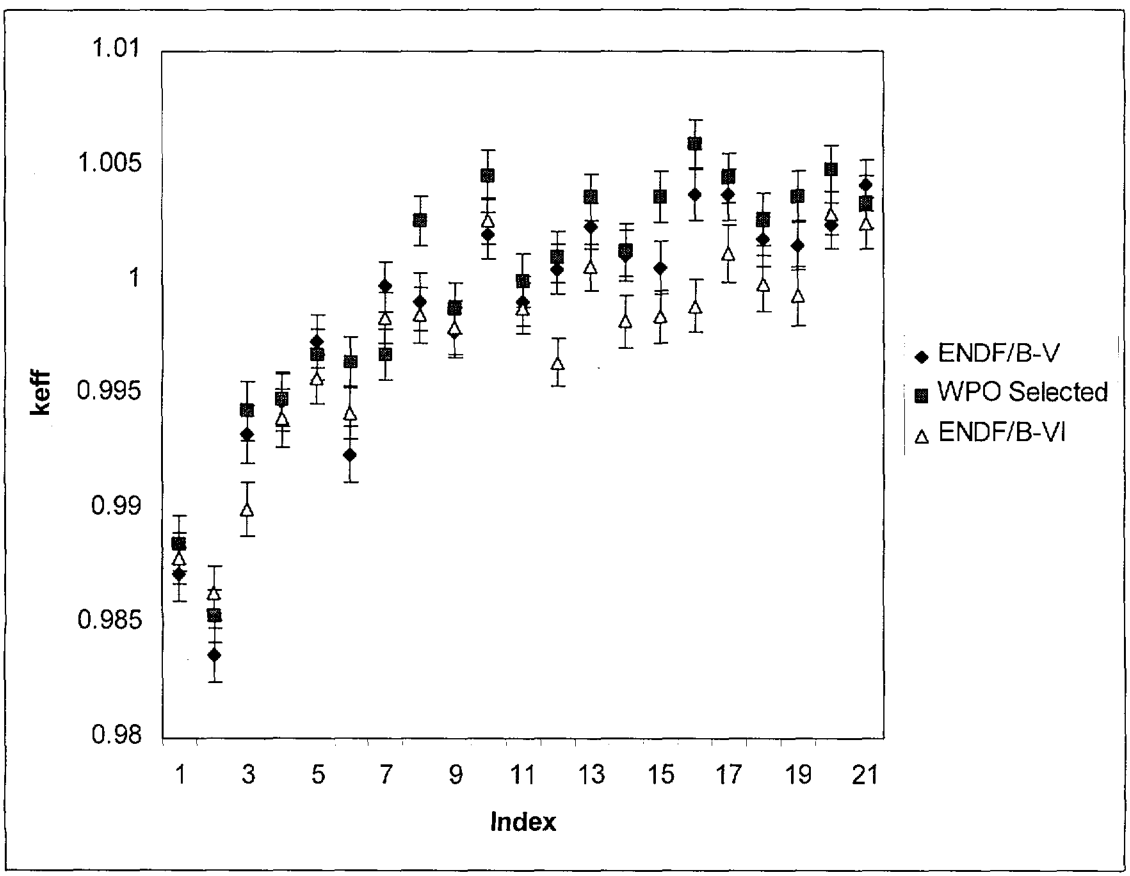

Figure 4.1-3. High Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Thermal)

Table 4.1-4. High Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Intermediate)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	hci2-1	0.99627	0.00079	0.2422	0.99236	0.00081	0.2430	0.99933	0.00081	0.2354
2	hci2-2	0.99614	0.00082	0.2413	0.99339	0.00107	0.2439	1.00136	0.00085	0.2357
3	hci2-3	0.99871	0.00083	0.2381	0.99603	0.00101	0.2377	1.00422	0.00087	0.2313
4	hci2-4	0.99892	0.00083	0.2376	0.99676	0.00101	0.2394	1.00469	0.00081	0.2312
5	hci2-5	1.00059	0.00078	0.2376	0.99736	0.001	0.2391	1.00278	0.00081	0.2313

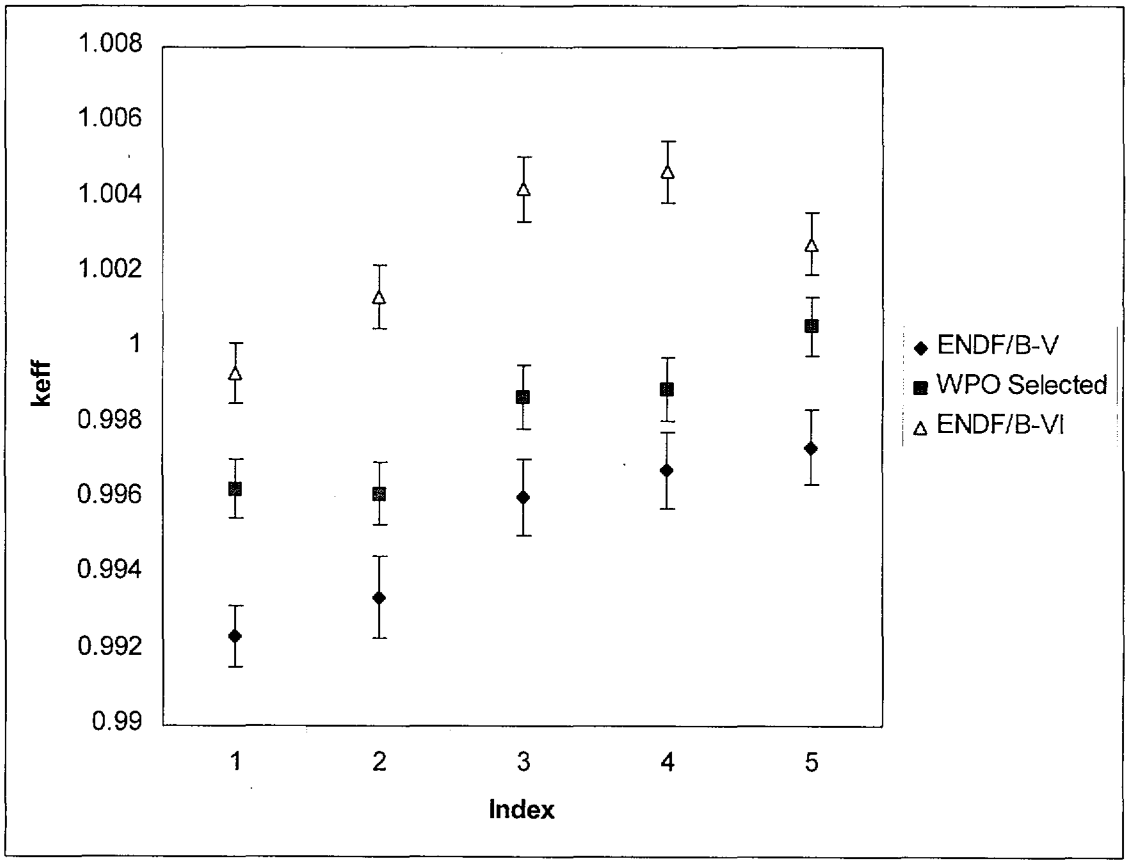

Figure 4.1-4. High Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Intermediate)

Table 4.1-5. High Enriched Uranium Oxide Fuel Plate Lattice Critical Experiments (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	spert1	0.99792	0.00184	0.0147	0.99792	0.00184	0.0147	0.99727	0.00176	0.014
2	spert2	0.99952	0.0019	0.0126	0.99952	0.0019	0.0126	0.99732	0.00186	0.0125
3	spert3	1.00676	0.00114	0.0117	1.00676	0.00114	0.0117	1.00384	0.00095	0.0112
4	spert4	0.99542	0.00173	0.011	0.99542	0.00173	0.011	0.98908	0.00179	0.0113
5	spert5	1.00104	0.00162	0.0105	1.00104	0.00162	0.0105	1.00072	0.00183	0.011
6	spert6	1.00133	0.00168	0.0102	1.00133	0.00168	0.0102	0.99683	0.00152	0.0098
7	spert7	0.99923	0.00163	0.0097	0.99923	0.00163	0.0097	0.9952	0.00147	0.0104
8	spert8	0.99843	0.00154	0.0098	0.99843	0.00154	0.0098	0.99009	0.0015	0.0096
9	spert9	1.00003	0.00143	0.0099	1.00003	0.00143	0.0099	0.99697	0.00149	0.0098
10	spert10	1.00608	0.00177	0.0147	1.00608	0.00177	0.0147	1.00534	0.0019	0.014
11	spert11	1.00565	0.00163	0.0115	1.00565	0.00163	0.0115	1.00283	0.00171	0.0113
12	spert12	1.00676	0.00156	0.0101	1.00676	0.00156	0.0101	1.00466	0.00175	0.0102
13	spert13	1.03289	0.00181	0.0143	1.03289	0.00181	0.0143	1.02936	0.00189	0.0141
14	spert14	0.99451	0.00158	0.0106	0.99451	0.00158	0.0106	0.99427	0.00159	0.0107
15	spert15	0.99355	0.00107	0.0106	0.99355	0.00107	0.0106	0.99179	0.00103	0.0105
16	spert16	1.00791	0.00175	0.012	1.00791	0.00175	0.012	1.00549	0.00183	0.0118
17	spert17	1.00569	0.00188	0.0131	1.00569	0.00188	0.0131	1.00206	0.00179	0.0133
18	spert18	1.00028	0.00198	0.014	1.00028	0.00198	0.014	1.00615	0.00181	0.0135
19	spert19	0.99482	0.00148	0.0097	0.99482	0.00148	0.0097	0.98982	0.00158	0.009
20	spert20	0.99652	0.00158	0.0114	0.99652	0.00158	0.0114	0.99315	0.00166	0.0115
21	spert21	1.00113	0.00186	0.0124	1.00113	0.00186	0.0126	1.00086	0.00178	0.0127
22	spert22	1.00272	0.00194	0.0133	1.00272	0.00194	0.0133	0.9984	0.00175	0.0131
23	spert23	1.00695	0.0011	0.0133	1.00695	0.0011	0.0132	1.00426	0.00099	0.0131

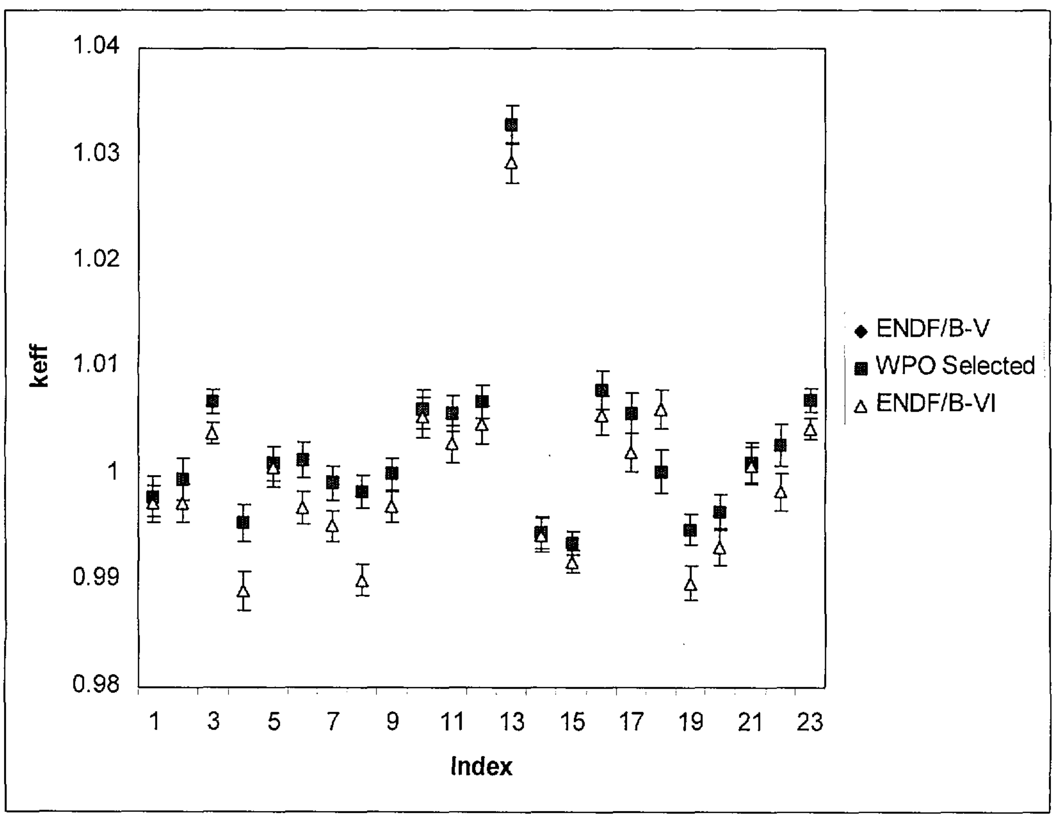

Figure 4.1-5. High Enriched Uranium Oxide Fuel Plate Lattice Critical Experiments (Thermal)

Table 4.1-6. High Enriched Uranium Oxide Single-Zone Cruciform Rod Lattice Critical Experiments (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	hct4-1	0.98875	0.00126	0.0740	0.98756	0.00122	0.0744	0.99143	0.00122	0.0735
2	hct4-2	0.98977	0.00124	0.0732	0.9889	0.0012	0.0736	0.99042	0.00123	0.0716
3	hct4-3	0.99049	0.00123	0.0765	0.99157	0.00119	0.0756	0.99272	0.00121	0.0747
4	hct4-4	0.99036	0.00118	0.0748	0.99116	0.00114	0.0742	0.98934	0.00122	0.0729
5	hct5-2	0.98493	0.0018	0.0776	0.98455	0.00128	0.0764	0.98887	0.00115	0.0765
6	hct6-t1	0.99034	0.00125	0.0715	0.98952	0.00137	0.0720	0.99212	0.00136	0.0718
7	hct6-t2	1.01252	0.00127	0.0231	1.0104	0.0013	0.0232	1.00798	0.00128	0.0229
8	hct6-t3	0.9993	0.001	0.0106	0.99557	0.00095	0.0104	0.99612	0.00099	0.0106
9	hct7-4	1.00086	0.00149	0.0340	0.99932	0.00164	0.0339	0.99992	0.00148	0.0334
10	hct7-5	0.99902	0.00164	0.0448	0.99492	0.00156	0.0458	0.99969	0.00152	0.0445
11	hct7-6	0.99556	0.00154	0.0486	0.99487	0.00154	0.0475	0.99545	0.00159	0.0470
12	hct8-1	0.98915	0.0011	0.0882	0.99042	0.00106	0.0882	0.99281	0.00108	0.0856
13	hct8-2	0.99273	0.00108	0.0919	0.98954	0.00112	0.0922	0.99117	0.00116	0.0912

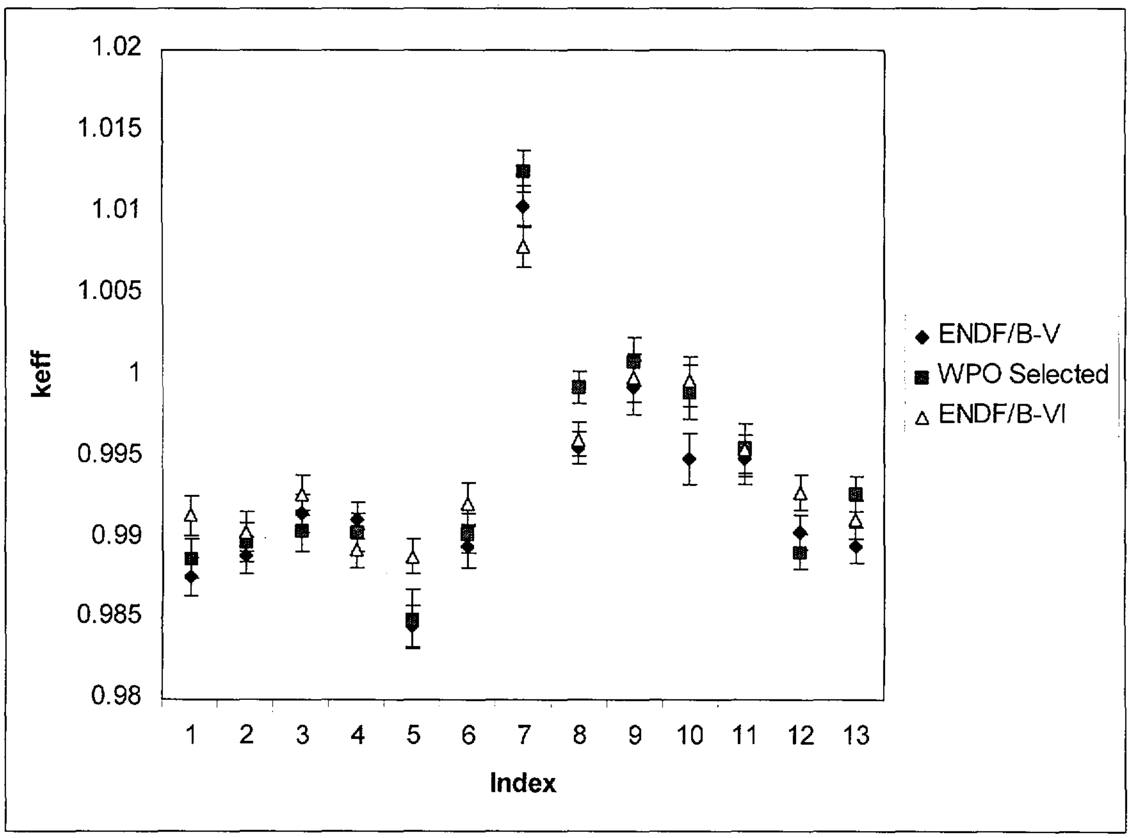

Figure 4.1-6. High Enriched Uranium Oxide Single-Zone Cruciform Rod Lattice Critical Experiments (Thermal)

Table 4.1-7. High Enriched Uranium Oxide Dual-Zone Cruciform Rod Lattice Critical Experiments (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	hct3-1	0.99475	0.0015	0.0466	0.99381	0.00136	0.0467	0.99473	0.00138	0.0460
2	hct3-2	0.99147	0.00152	0.0417	0.99746	0.00144	0.0404	0.9956	0.00144	0.0405
3	hct3-3	0.99699	0.00132	0.0330	0.99637	0.00144	0.0337	0.99473	0.00149	0.0333
4	hct3-4	1.00159	0.00141	0.0262	0.99796	0.00148	0.0259	0.99688	0.00148	0.0261
5	hct3-5	1.00175	0.00146	0.0207	1.00253	0.00137	0.0202	1.0006	0.00151	0.0199
6	hct3-6	1.00947	0.00153	0.0419	1.00616	0.00152	0.0405	1.00706	0.0016	0.0398
7	hct3-7	1.01327	0.00144	0.0345	1.01132	0.00143	0.0339	1.009	0.00145	0.0339
8	hct3-8	1.01001	0.0015	0.0342	1.00988	0.00155	0.0329	1.01081	0.00143	0.0334
9	hct3-9	1.01332	0.00147	0.0265	1.01309	0.00154	0.0263	1.01355	0.00145	0.0260
10	hct3-10	1.01014	0.00143	0.0202	1.00962	0.0014	0.0209	1.01004	0.00149	0.0200
11	hct3-11	1.01347	0.00137	0.0177	1.0133	0.0014	0.0177	1.01317	0.00152	0.0173
12	hct3-12	0.9933	0.00136	0.0263	0.98956	0.00133	0.0265	0.98921	0.00135	0.0265
13	hct3-13	1.00022	0.00126	0.0140	0.99677	0.00119	0.0139	0.99588	0.00126	0.0133
14	hct3-14	1.00853	0.00143	0.0305	1.005	0.00157	0.0300	1.00682	0.0015	0.0301
15	hct3-15	1.00648	0.00128	0.0185	1.00775	0.00142	0.0183	1.003	0.00141	0.0179

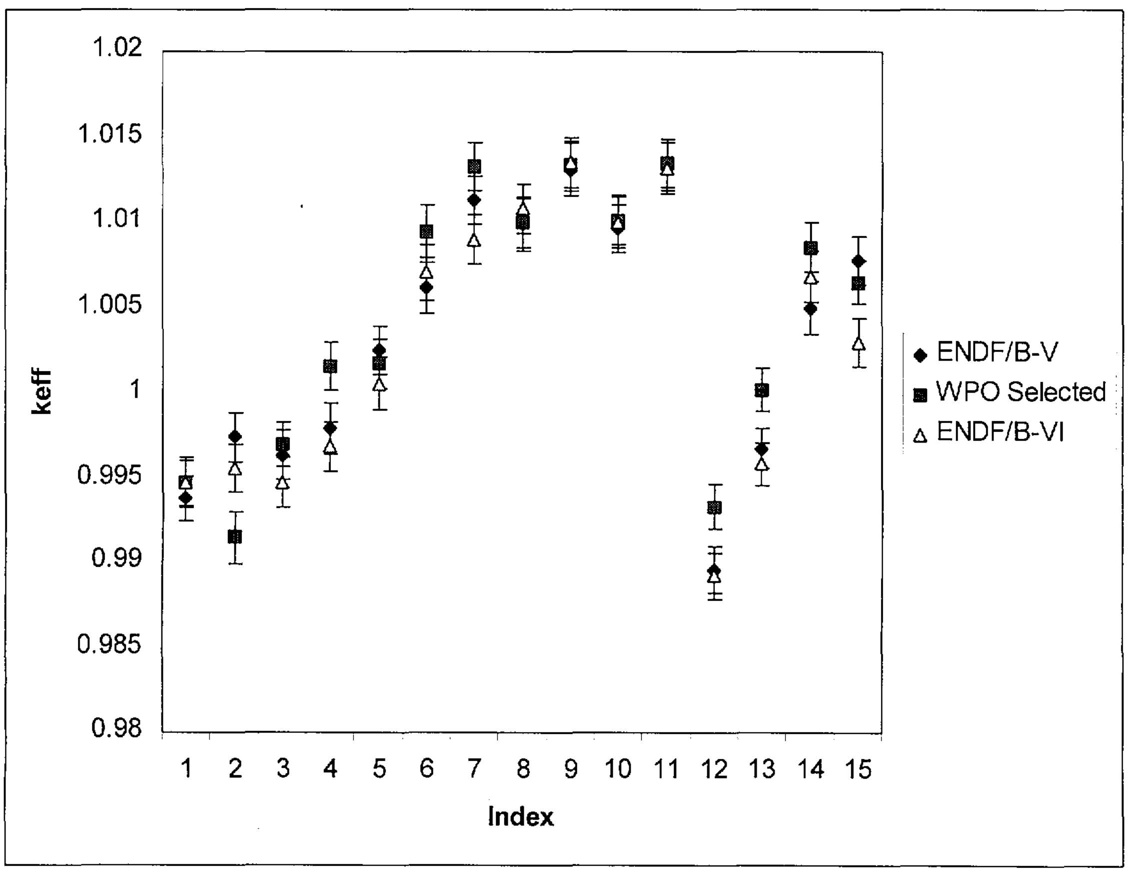

Figure 4.1-7. High Enriched Uranium Oxide Dual-Zone Cruciform Rod Lattice Critical Experiments (Thermal)

Table 4.1-8. Intermediate Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	tri17	1.00897	0.00141	0.0236	1.00737	0.00129	0.0233	1.00740	0.00132	0.0236
2	tri18	1.01314	0.00126	0.0240	1.00921	0.00132	0.0234	1.01195	0.00143	0.0238

NOTE: Values in table carry TBV-1369

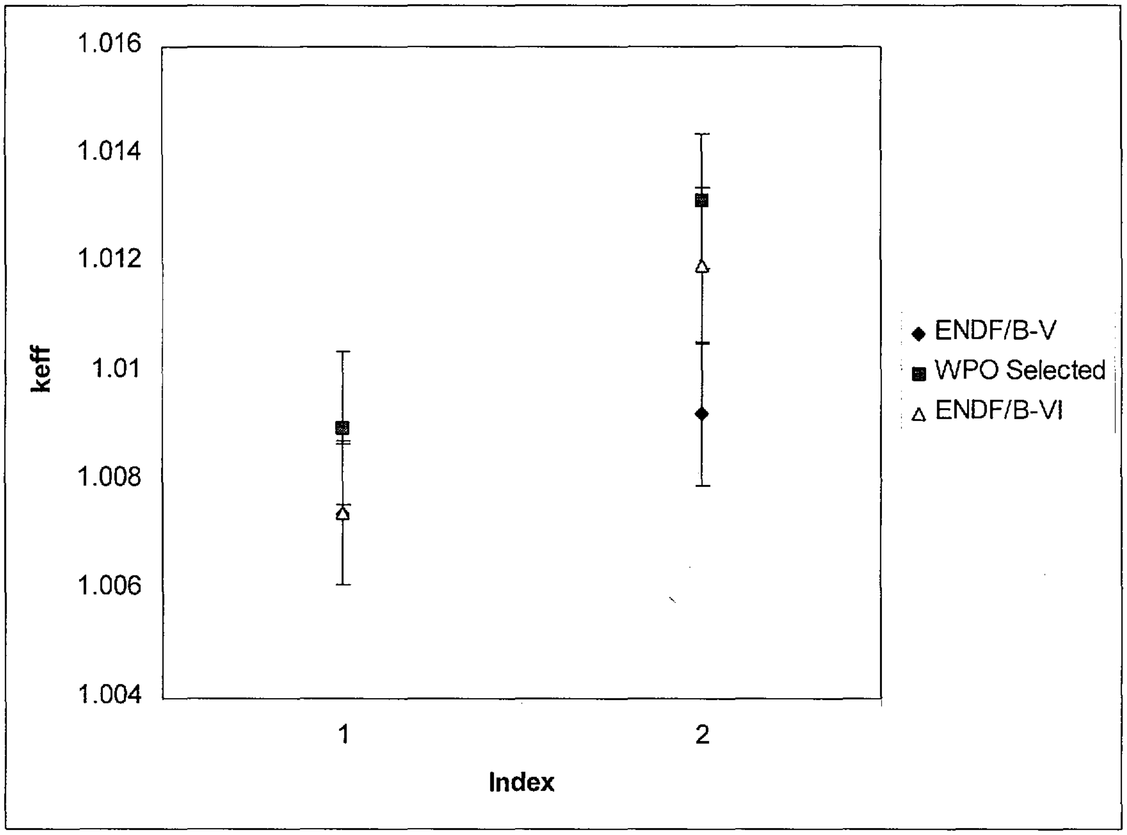

Figure 4.1-8. Intermediate Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Thermal)

Table 4.1-9. Low Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	exp18	1.00503	0.00167	0.08863	1.00192	0.00168	0.08861	0.99716	0.00169	0.08890

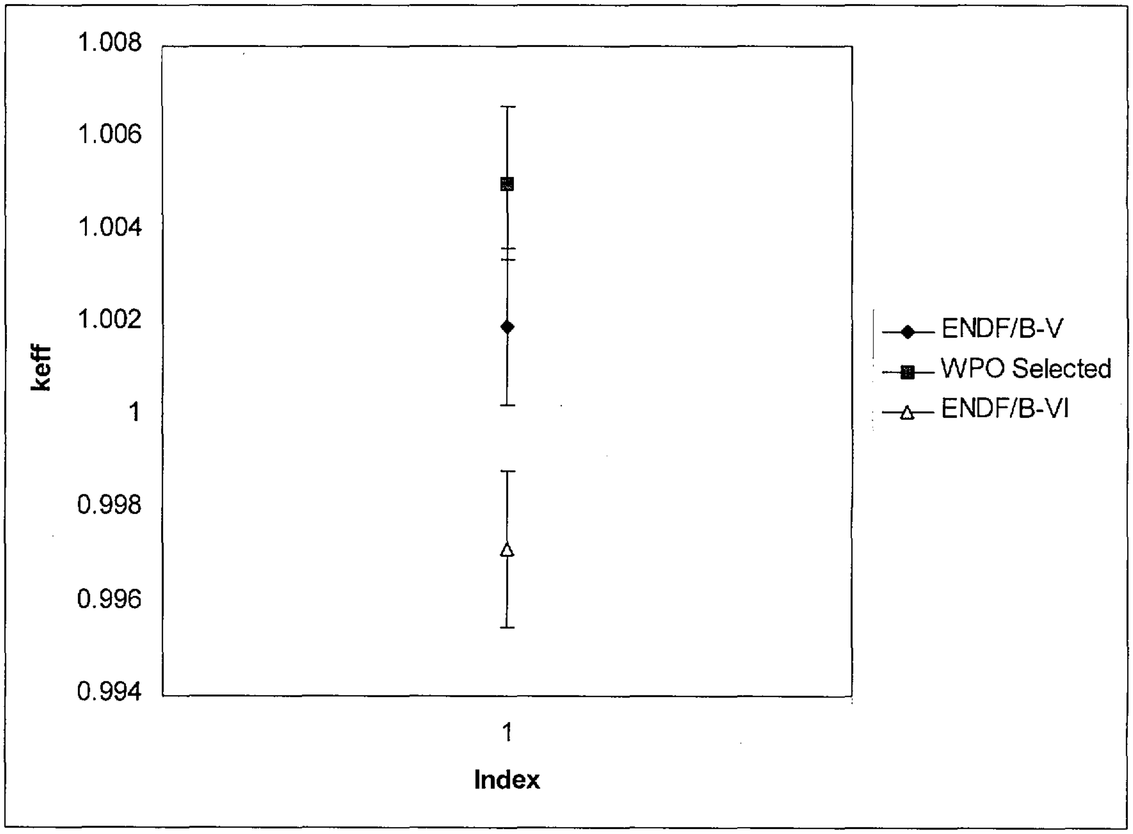

Figure 4.1-9. Low Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Thermal)

Table 4.1-10. Low Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Intermediate)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	core2	1.00058	0.00159	0.19988	1.00029	0.00147	0.19859	0.99441	0.00156	0.19779
2	core3	1.00019	0.00148	0.18078	0.99991	0.00147	0.18029	0.99389	0.00152	0.17831
3	core4	0.9948	0.0015	0.17908	0.99429	0.00157	0.18206	0.99094	0.00149	0.17962
4	core5	0.99445	0.00153	0.16919	0.99226	0.00149	0.16804	0.99118	0.00157	0.16845
5	core6	0.99556	0.00152	0.17216	0.9924	0.00152	0.17097	0.98847	0.00155	0.17213
6	core7	0.99463	0.00151	0.15963	0.99357	0.00153	0.16113	0.99004	0.00155	0.15992
7	core8	0.98895	0.00149	0.16496	0.99294	0.00158	0.16116	0.99107	0.00154	0.16043
8	core9	0.99298	0.00144	0.15528	0.9939	0.00152	0.15725	0.98803	0.00152	0.15449
9	core10	0.99511	0.00148	0.16036	0.99523	0.00147	0.1597	0.99204	0.00146	0.15896
10	core11	0.99699	0.00148	0.17893	0.99789	0.00153	0.17801	0.99693	0.00149	0.17822
11	core12	0.99549	0.00151	0.16671	0.99658	0.00155	0.1677	0.99442	0.00148	0.16679
12	core13	0.99933	0.00151	0.18075	0.99594	0.00159	0.17945	0.99702	0.00154	0.17772
13	core15	0.99107	0.00157	0.18348	0.98476	0.0015	0.18301	0.98498	0.00151	0.18146
14	core16	0.99041	0.0015	0.16952	0.98967	0.00154	0.1703	0.98719	0.00162	0.16862
15	core17	0.99365	0.00151	0.18187	0.99534	0.00148	0.18122	0.98957	0.00153	0.18184
16	core18	0.9947	0.0015	0.16855	0.9938	0.0016	0.16945	0.98602	0.00155	0.16933
17	core19	0.99383	0.00153	0.18354	0.99637	0.00146	0.18256	0.98955	0.00159	0.18185
18	core20	0.99392	0.00151	0.16933	0.99247	0.00157	0.17024	0.98777	0.00151	0.17006
19	core21	0.9916	0.0014	0.16225	0.99318	0.00146	0.16143	0.99135	0.00152	0.16011
20	exp1	1.00084	0.00088	0.12095	0.99958	0.00087	0.12037	0.99657	0.00093	0.12075
21	exp2	0.99842	0.00088	0.12469	0.99762	0.00086	0.12291	0.99351	0.00097	0.12334
22	exp3	0.99898	0.00089	0.12172	1.00103	0.00091	0.12005	0.99613	0.00095	0.12083
23	exp4	1.00104	0.00087	0.12003	1.00241	0.00091	0.12057	0.99582	0.0009	0.12186
24	exp5	1.00037	0.00107	0.27968	0.9969	0.00103	0.28386	0.99295	0.00104	0.28018
25	exp6	0.99675	0.00103	0.17662	0.99772	0.00105	0.17775	0.9944	0.00111	0.17643
26	exp7	0.99724	0.00111	0.1784	0.99821	0.0011	0.17822	0.99144	0.00105	0.1773
27	exp8	1.00719	0.0011	0.17735	1.007	0.00102	0.17807	1.0033	0.00105	0.17769
28	exp9	1.00827	0.00099	0.22171	1.00687	0.00109	0.22393	1.00496	0.00105	0.22242
29	exp10	1.0066	0.00174	0.2239	1.00499	0.00173	0.2225	1.00454	0.00182	0.22149
30	exp11	1.00358	0.00157	0.26643	1.00046	0.00177	0.26803	0.99995	0.00162	0.26652
31	exp12	1.00546	0.00108	0.19461	1.00113	0.00105	0.19577	1.00189	0.00111	0.19235
32	exp13	1.00371	0.00113	0.19421	1.00363	0.00109	0.19365	0.99936	0.00109	0.1936
33	exp14	0.99593	0.00099	0.20945	0.99534	0.00099	0.20881	0.98998	0.00099	0.20806
34	exp15	1.00074	0.00087	0.10984	1.00105	0.00088	0.10995	0.99603	0.00089	0.10902
35	exp17	1.00218	0.00186	0.15637	0.99815	0.00173	0.15534	0.99216	0.00173	0.15603
36	ugd1	1.00033	0.00143	0.20132	0.99717	0.0014	0.20069	0.99688	0.00149	0.19672
37	ugd20	1.00322	0.00153	0.20698	1.00179	0.00159	0.20624	0.9971	0.00155	0.20429
38	ugd2	0.99945	0.00145	0.19828	0.99892	0.0015	0.20018	0.9947	0.00151	0.19858
39	ugd3	1.00054	0.00147	0.19948	0.99846	0.00145	0.19736	0.99449	0.00151	0.19988
40	ugd4	1.00193	0.0015	0.19985	0.99911	0.00144	0.19744	1.00031	0.00149	0.1993
41	ugd5	0.99955	0.00154	0.19752	0.99958	0.00147	0.20041	0.99686	0.0015	0.19842
42	ugd6	0.99996	0.00152	0.19775	1.00229	0.00149	0.19897	0.9952	0.00154	0.19894
43	ugd7	1.0041	0.00148	0.19675	0.99807	0.00144	0.19896	0.99287	0.00147	0.19687
44	ugd8	0.99929	0.00154	0.19756	0.99904	0.00151	0.19942	0.99722	0.00147	0.19833
45	ugd9	1.00135	0.00156	0.19873	1.00042	0.00159	0.19963	0.99743	0.00148	0.19651
46	ugd10	0.9979	0.00144	0.2011	1.00115	0.00145	0.20029	0.99696	0.00151	0.19695
47	ugd12	0.9994	0.00161	0.20965	1.00178	0.00146	0.20785	1.00134	0.00149	0.2088
48	ugd13	1.00049	0.00155	0.20841	0.99887	0.00166	0.20937	1.00037	0.00148	0.2066
49	ugd14	1.00066	0.00156	0.20416	1.00069	0.00144	0.20585	0.99761	0.00153	0.20711
50	ugd15	1.00158	0.00151	0.2056	0.99927	0.00147	0.20333	0.9983	0.00156	0.20342
51	ugd16	1.00335	0.00151	0.20648	0.99904	0.0015	0.20947	0.99903	0.00158	0.20601
52	ugd17	0.99912	0.00151	0.20341	0.99909	0.00153	0.20572	0.998	0.00149	0.2024
53	ugd18	0.99876	0.0015	0.20851	0.99741	0.00156	0.21013	0.99819	0.00155	0.20862
54	ugd19	1.00133	0.00153	0.21011	0.99908	0.00155	0.20757	1.00098	0.00151	0.20506
55	Case_1	0.99436	0.00167	0.1229	0.9987	0.00191	0.1239	0.99248	0.00153	0.1227
56	Case_2	0.99445	0.00158	0.1223	0.99557	0.00185	0.1208	0.9923	0.00161	0.121

Table 4.1-10. Low Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Intermediate)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
57	Case_3	0.99982	0.00159	0.12	0.9957	0.00175	0.122	0.99637	0.00172	0.1221
58	Case_4	0.99313	0.00161	0.1222	0.99907	0.00144	0.1208	0.98949	0.00156	0.1207
59	Case_5	0.9931	0.00169	0.1204	0.99313	0.00157	0.1218	0.99429	0.00172	0.1187
60	Case_6	0.99831	0.00158	0.1221	0.99882	0.00143	0.1208	0.99161	0.00158	0.1219
61	Case_7	0.99261	0.00138	0.1211	0.99677	0.00143	0.1186	0.99578	0.00155	0.1176
62	Case_8	0.99888	0.00151	0.1209	1.00246	0.00162	0.1201	0.99597	0.00149	0.1214
63	subc2p8h	1.00887	0.0032	0.4085	1.0048	0.00294	0.4085	1.01057	0.00335	0.4005
64	subc3p1h	1.0335	0.00287	0.3417	1.03902	0.00248	0.3516	1.04014	0.0031	0.3447
65	subc3p4h	1.01929	0.00238	0.3153	1.02088	0.00277	0.3145	1.01971	0.00266	0.3165
66	ssr83	0.99299	0.00074	0.18197	0.9928	0.00069	0.1821	0.98821	0.00073	0.18115
67	ssr48	0.9939	0.00071	0.15568	0.99396	0.00074	0.1564	0.99136	0.00072	0.15464
68	lct27-1	1.0157	0.0005	0.1025	1.0139	0.0005	0.1024	1.0102	0.0006	0.1023

NOTE: Index numbers 1 through 19 carry TBV-1357, index numbers 20 through 23 carry TBV-1362, index numbers 24 through 26 carry TBV-1363 and 1364, index numbers 27 through 30 carry TBV-1365, index numbers 31 and 32 carry TBV-1361, index numbers 33 and 34 carry TBV-1368, index numbers 36 through 54 carry TBV-1358, index numbers 63 through 65 carry TBV-1360, and index numbers 66 and 67 carry TBV-1359 and 1368

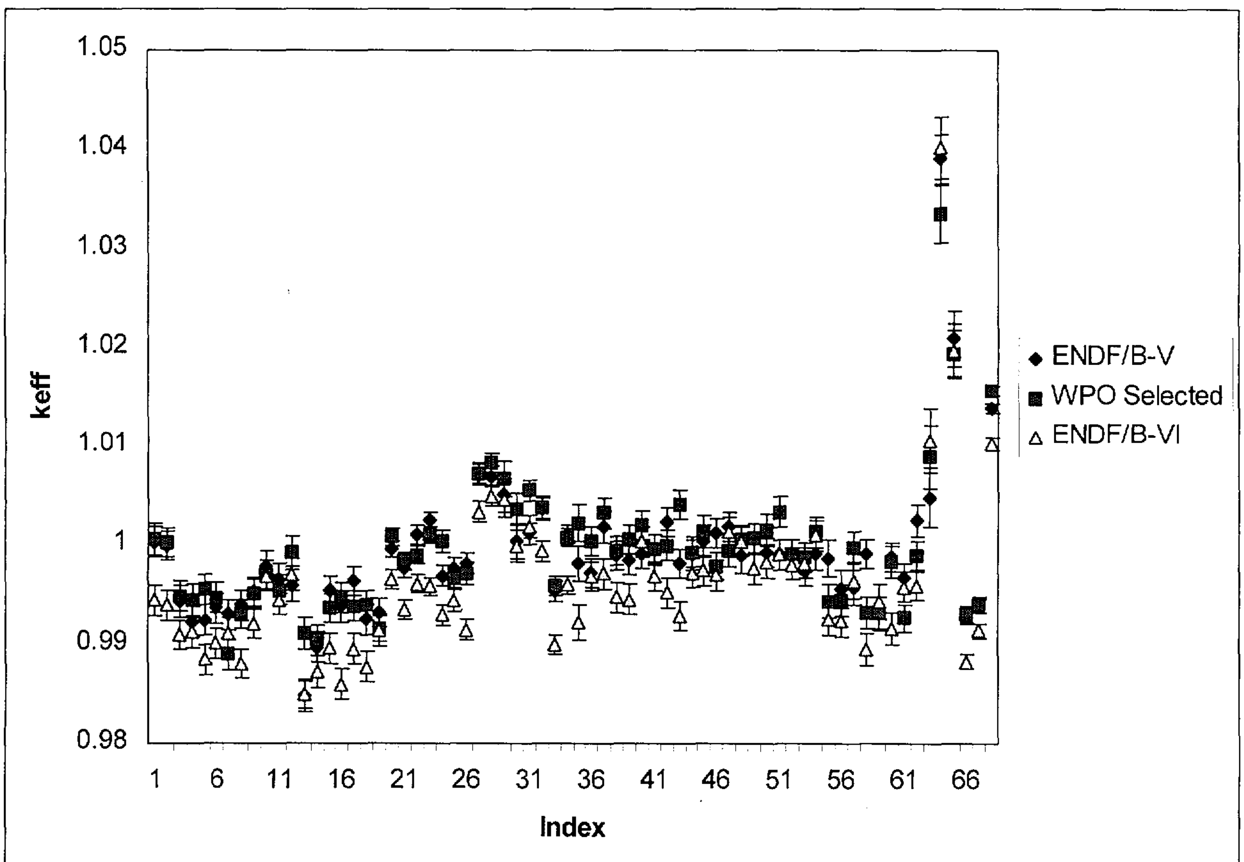

Figure 4.1-10. Low Enriched Uranium Oxide Fuel Pin Lattice Critical Experiments (Intermediate)

4.2 HOMOGENEOUS CRITICALS

This section tabulates the MCNP k_{eff} results for the LCEs from CRWMS M&O (1999b), CRWMS M&O (1999c), and CRWMS M&O (1999d) according to experimental similarities. Tables 4.2-1 through 4.2-12 and Figures 4.2-1 through 4.1-12 present the results for the LCEs according to the following distinct experimental classifications:

- Table and Figure 4.2-1: Homogeneous critical experiments using mixed Pu nitrate and U nitrate solutions (Thermal Systems)
- Table and Figure 4.2-2: Homogeneous critical experiments using Pu nitrate solutions (Thermal Systems)
- Table and Figure 4.2-3: Pu metal fast experiments (Fast Systems)
- Table and Figure 4.2-4: Homogeneous critical experiments using HEU nitrate solutions (Thermal Systems)
- Table and Figure 4.2-5: HEU metal fast experiments (Fast Systems)
- Table and Figure 4.2-6: Homogeneous critical experiments using IEU nitrate solutions (Thermal Systems)
- Table and Figure 4.2-7: Homogeneous critical experiments using IEU nitrate solutions (Intermediate Systems)
- Table and Figure 4.2-8: IEU metal fast experiments (Fast Systems)
- Table and Figure 4.2-9: Homogeneous critical experiments using LEU nitrate solutions (Thermal Systems)
- Table and Figure 4.2-10: Homogeneous critical experiments using LEU nitrate solutions (Intermediate Systems)
- Table and Figure 4.2-11: Homogeneous critical experiments using ^{233}U fuel (Thermal Systems)
- Table and Figure 4.2-12: Homogeneous critical experiments using ^{233}U fuel (Fast Systems)

The column identified as AENCF contains the average energy of the neutron causing fission. It is a measure of the energy spectrum of the neutrons and has units of MeV.

It should be noted that unaccepted data were used in the development of the results presented in Table 4.2-9 for the LEUJA cases. These values carry TBV-1370.

Table 4.2-1. Homogeneous Critical Experiments Using Mixed Plutonium and Natural Uranium Nitrate Solutions (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	pnl3187	0.99821	0.00116	0.04158	0.99762	0.00116	0.04174	0.98843	0.00105	0.04181
2	pnl3391	0.99318	0.00112	0.04075	0.99425	0.00116	0.04105	0.98609	0.00104	0.0413
3	pnl3492	0.99619	0.00113	0.04386	0.99754	0.00116	0.04313	0.98923	0.00119	0.04361
4	pnl3593	0.99694	0.00121	0.04614	0.99727	0.00111	0.04586	0.99095	0.00108	0.04649
5	pnl3694	1.00275	0.00113	0.04483	1.00255	0.00118	0.04452	0.99364	0.00116	0.04427
6	pnl3795	1.00302	0.00117	0.03965	1.00166	0.00116	0.04	0.99703	0.00119	0.03983
7	pnl3808	1.00178	0.00095	0.02059	1.00195	0.00113	0.0213	0.9928	0.00107	0.02061
8	pnl3896	1.00263	0.0011	0.02357	1.00235	0.00124	0.02319	0.99527	0.00113	0.02353
9	pnl3897	1.00323	0.00125	0.01447	1.00449	0.00105	0.0142	0.99639	0.00104	0.01426
10	pnl3898	1.00297	0.00118	0.02973	1.00287	0.00104	0.02994	0.99728	0.00116	0.03
11	pnl3999	1.00707	0.00108	0.02933	1.00919	0.00107	0.02959	1.00133	0.00117	0.02906
12	pnl5300	1.0067	0.00105	0.02917	1.008	0.0011	0.02884	1.00106	0.00102	0.02891
13	pnl1158	1.00686	0.00067	0.00393	1.00691	0.00065	0.00384	1.00301	0.00065	0.00386
14	pnl1159	1.00558	0.00064	0.0038	1.00735	0.00062	0.0037	1.00244	0.00069	0.0039
15	pnl1161	1.00751	0.00066	0.00597	1.0079	0.00067	0.00607	1.00033	0.0006	0.00605
16	awre1	1.01511	0.0012	0.03133	1.01469	0.00103	0.03153	1.00516	0.00127	0.03133
17	awre2	1.01167	0.00117	0.03206	1.01568	0.00115	0.03149	1.00467	0.00115	0.03199
18	awre3	1.01028	0.00114	0.03183	1.01203	0.0012	0.03197	1.00181	0.00117	0.03253
19	awre4	1.00486	0.00111	0.03228	1.0051	0.00118	0.03189	0.9984	0.0011	0.03207
20	awre5	1.00875	0.00101	0.01062	1.00847	0.001	0.01043	1.00019	0.00107	0.01028
21	awre6	1.01337	0.00108	0.01053	1.01067	0.00103	0.01035	1.00588	0.00107	0.01041
22	awre7	1.0064	0.00102	0.01089	1.00796	0.00099	0.0105	0.99924	0.00101	0.01046
23	awre8	1.01255	0.00091	0.00684	1.01284	0.00083	0.00686	1.00379	0.00079	0.00682
24	awre9	1.00977	0.00088	0.00684	1.0094	0.00092	0.00661	1.00207	0.00094	0.00692
25	awre10	1.00839	0.00081	0.00648	1.01024	0.00084	0.00662	1.00221	0.00088	0.00652
26	pnl1577	0.99645	0.00128	0.05956	0.99578	0.00122	0.05887	0.98992	0.00125	0.05858
27	pnl1678	0.99976	0.00115	0.05069	0.99735	0.00115	0.05044	0.98991	0.00125	0.05119
28	pnl1783	0.99976	0.00115	0.05386	0.99922	0.00118	0.05337	0.99258	0.00124	0.0533
29	pnl1868	1.00247	0.00119	0.03416	1.0039	0.00127	0.03434	0.99364	0.00127	0.0345
30	pnl1969	0.99967	0.00111	0.0336	1.00003	0.0012	0.0334	0.98934	0.00128	0.03334
31	pnl2070	0.99925	0.00115	0.03743	0.99956	0.00143	0.0377	0.99312	0.00133	0.03853
32	pnl2565	1.00363	0.00112	0.01295	1.00148	0.00123	0.01294	0.99786	0.00119	0.01298
33	pnl2666	1.00337	0.00105	0.0116	1.00179	0.00112	0.01169	0.99694	0.00109	0.01169
34	pnl2767	1.00629	0.00113	0.01197	1.00607	0.00113	0.0123	0.99664	0.00113	0.01216

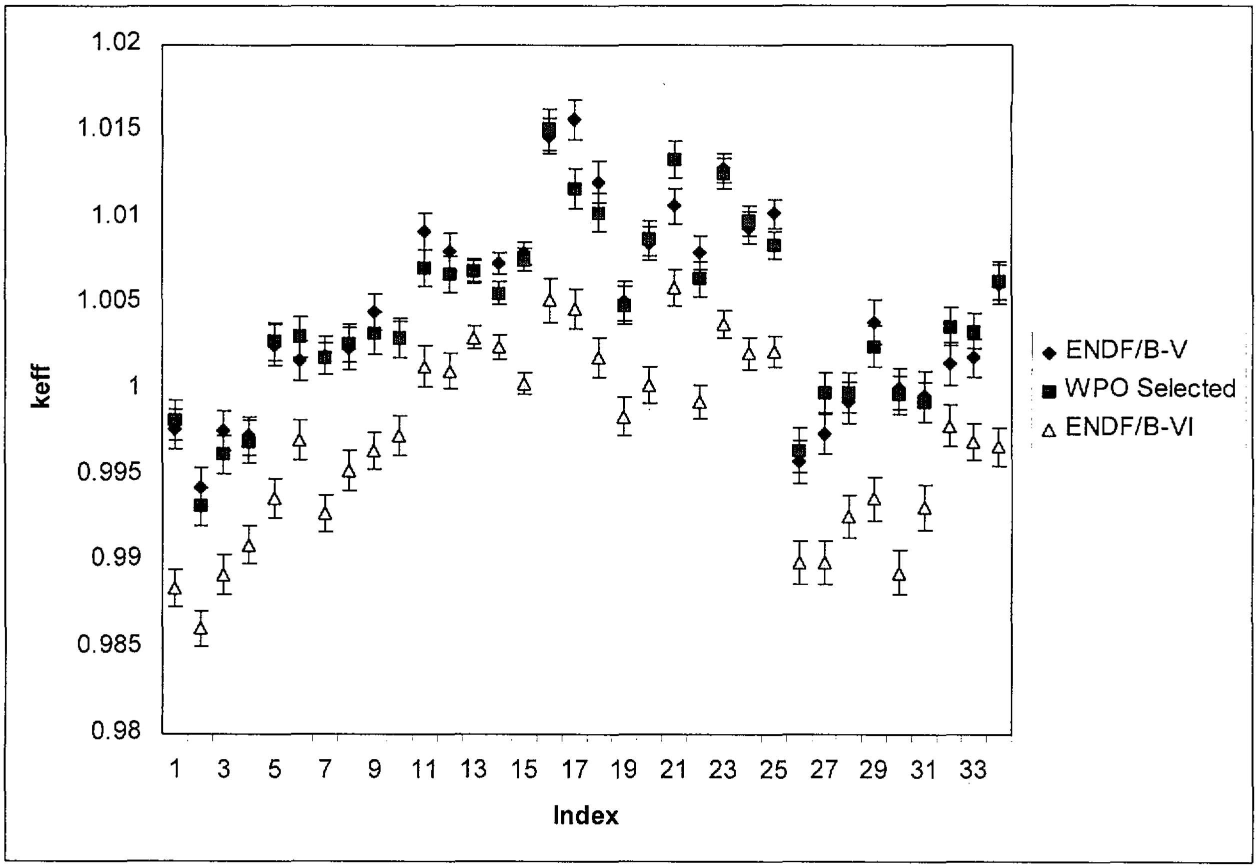

Figure 4.2-1. Homogeneous Critical Experiments Using Mixed Plutonium and Natural Uranium Nitrate Solutions (Thermal)

Table 4.2-2. Homogeneous Critical Experiments Using Plutonium Nitrate Solutions (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	pust1t1	1.00995	0.00102	0.01252	1.01064	0.00101	0.0126	1.00058	0.00102	0.01301
2	pust1t2	1.01109	0.001	0.01702	1.01278	0.001	0.01722	1.00408	0.00104	0.01715
3	pust1t3	1.01396	0.00094	0.02159	1.01447	0.00105	0.02086	1.00692	0.00106	0.02172
4	pust1t4	1.00643	0.00104	0.02397	1.00766	0.00105	0.02374	0.99921	0.00107	0.02405
5	pust1t5	1.01014	0.00101	0.02479	1.01107	0.00099	0.02464	1.00222	0.00103	0.02514
6	pust1t6	1.00831	0.00104	0.04809	1.01003	0.00104	0.04883	1.00585	0.00105	0.0481
7	pu003-1	1.00962	0.00091	0.00623	1.0089	0.0009	0.00634	1.00208	0.00095	0.00631
8	pu003-2	1.00885	0.00091	0.00651	1.00764	0.00091	0.00655	1.00029	0.00091	0.00664
9	pu003-3	1.01228	0.00092	0.00693	1.0116	0.00095	0.00696	1.00324	0.00097	0.00711
10	pu003-4	1.00965	0.00094	0.0072	1.01162	0.00097	0.00723	1.00401	0.00092	0.00731
11	pu003-5	1.01393	0.00092	0.00785	1.01162	0.00091	0.00772	1.00371	0.0009	0.00736
12	pu003-6	1.01214	0.00091	0.00845	1.01275	0.00097	0.00858	1.00425	0.00094	0.00863
13	pu003-7	1.01369	0.00093	0.00678	1.01438	0.00092	0.00669	1.00545	0.00092	0.00667
14	pu003-8	1.01175	0.00095	0.00703	1.01194	0.00093	0.00681	1.00196	0.00096	0.007
15	pu004-1	1.01134	0.00088	0.00524	1.01168	0.00086	0.00515	1.00302	0.00087	0.00521
16	pu004-2	1.00448	0.00082	0.00541	1.00671	0.00085	0.00534	0.99646	0.00095	0.00517
17	pu004-3	1.00916	0.00087	0.00538	1.0087	0.00091	0.00555	0.9994	0.00085	0.00542
18	pu004-4	1.00712	0.00086	0.00561	1.00559	0.0009	0.00563	0.99678	0.00094	0.00569
19	pu004-5	1.00753	0.00091	0.00543	1.00684	0.00088	0.00544	1.00082	0.00087	0.00538
20	pu004-6	1.00862	0.00087	0.00564	1.00867	0.00089	0.00572	1.00092	0.00088	0.00549
21	pu004-7	1.01248	0.0009	0.0056	1.01179	0.00086	0.00582	1.00513	0.00087	0.00582
22	pu004-8	1.00778	0.00086	0.0062	1.00786	0.00088	0.00593	0.99815	0.00089	0.00599
23	pu004-9	1.00965	0.00089	0.00619	1.0091	0.00087	0.00615	0.99749	0.00089	0.00643
24	pu005-1	1.0086	0.00088	0.00571	1.00871	0.00089	0.00568	1.00322	0.00087	0.00583
25	pu005-2	1.00908	0.00088	0.00589	1.01075	0.00091	0.00579	0.99995	0.00088	0.00587
26	pu005-3	1.01116	0.00091	0.0062	1.01227	0.00087	0.00622	1.00236	0.00092	0.00614
27	pu005-4	1.01197	0.00093	0.00664	1.01183	0.00088	0.00678	1.00279	0.00088	0.00665
28	pu005-5	1.01367	0.0009	0.00723	1.01254	0.00088	0.00725	1.00389	0.00093	0.00758
29	pu005-6	1.0102	0.00095	0.00766	1.01279	0.00092	0.0078	1.00338	0.00092	0.00771
30	pu005-7	1.01073	0.00094	0.00838	1.01031	0.00094	0.00837	1.00194	0.00093	0.00845
31	pu005-8	1.00799	0.00091	0.00593	1.00559	0.0009	0.00608	0.99915	0.00089	0.006
32	pu005-9	1.01023	0.00089	0.00631	1.00929	0.00086	0.00638	0.99937	0.00088	0.00618
33	pu007-2	1.01024	0.00102	0.04021	1.01358	0.00109	0.04098	1.00561	0.00106	0.0412
34	pu007-3	1.00591	0.00111	0.03928	1.00535	0.00104	0.03969	0.99948	0.00105	0.03932
35	pu007-5	1.01502	0.00106	0.01764	1.01438	0.00102	0.01754	1.00665	0.00104	0.01758
36	pu007-6	1.00873	0.00101	0.01799	1.00808	0.00106	0.01773	1.00134	0.001	0.01757
37	pu007-7	1.01053	0.00103	0.01783	1.01039	0.00105	0.0177	1.00333	0.00101	0.01747
38	pu007-8	1.00254	0.00103	0.0181	1.00512	0.00102	0.01796	0.99316	0.00105	0.01783
39	pu007-9	1.00327	0.00106	0.01815	1.00267	0.00108	0.01795	0.99427	0.00106	0.01774
40	pu04-10	1.00987	0.00092	0.00715	1.00979	0.00092	0.0072	1.00196	0.00093	0.00734
41	pu04-11	1.0095	0.00092	0.00805	1.00799	0.00096	0.00849	0.99716	0.00092	0.00836
42	pu04-12	1.01108	0.00087	0.00594	1.01028	0.00092	0.00573	1.0028	0.00087	0.00572
43	pu04-13	1.00856	0.00091	0.00579	1.00634	0.00089	0.00574	0.99942	0.00091	0.00585
44	pu07-10	1.00706	0.00104	0.01653	1.00643	0.00101	0.01711	0.99653	0.00106	0.0162
45	pu10091	1.02337	0.00101	0.01675	1.02188	0.00105	0.01658	1.01378	0.00102	0.01672
46	pu10092	1.02091	0.00097	0.01299	1.01932	0.00103	0.0127	1.01053	0.00097	0.01272
47	pu10093	1.01316	0.00097	0.00994	1.01428	0.00099	0.0093	1.00474	0.00096	0.00969
48	pu10111	1.01879	0.00099	0.01001	1.01833	0.00095	0.01026	1.01211	0.00102	0.00985
49	pu10112	1.01543	0.00098	0.00873	1.01691	0.00096	0.00888	1.00964	0.00097	0.00862
50	pu10113	1.01615	0.00092	0.00852	1.01378	0.00097	0.00867	1.00701	0.00097	0.00876
51	pu10114	1.00903	0.00091	0.0079	1.00854	0.00093	0.00798	0.99799	0.00096	0.0081
52	pu10115	1.01069	0.00093	0.00755	1.01003	0.0009	0.00753	1.00213	0.00093	0.00734
53	pu10116	1.01992	0.00101	0.01114	1.01922	0.001	0.01109	1.01027	0.00102	0.01128
54	pu10117	1.01146	0.00092	0.00879	1.00973	0.001	0.00901	1.0005	0.00094	0.0092
55	pu10121	1.0156	0.00097	0.00896	1.01726	0.00096	0.00892	1.00876	0.00094	0.00896
56	pu10122	1.01616	0.00095	0.00776	1.0153	0.00098	0.00795	1.0067	0.00096	0.00776

Table 4.2-2. Homogeneous Critical Experiments Using Plutonium Nitrate Solutions (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
57	pu10123	1.02352	0.00094	0.00691	1.02207	0.00095	0.00671	1.01591	0.00093	0.00699
58	pu10124	1.01642	0.00087	0.0061	1.01695	0.00089	0.00594	1.00717	0.0009	0.00599
59	pu11161	1.01661	0.00103	0.00738	1.01669	0.00101	0.00737	1.0111	0.001	0.00744
60	pu11162	1.02377	0.00101	0.00777	1.02064	0.00103	0.00759	1.01343	0.00098	0.00773
61	pu11163	1.02224	0.00101	0.00827	1.02545	0.001	0.00829	1.01656	0.00104	0.00794
62	pu11164	1.01688	0.00105	0.00845	1.01704	0.00101	0.00851	1.00895	0.00101	0.00808
63	pu11165	1.01338	0.00104	0.00973	1.01318	0.00104	0.00955	1.00576	0.00102	0.00988
64	pu11181	1.00169	0.00089	0.00505	1.00317	0.00088	0.00501	0.99468	0.00091	0.00503
65	pu11182	1.0068	0.00088	0.00549	1.00894	0.00091	0.00527	1.00073	0.00088	0.0051
66	pu11183	1.00336	0.00097	0.00514	1.0051	0.00091	0.00528	0.99776	0.00092	0.00512
67	pu11184	1.00285	0.00088	0.00547	1.00198	0.00088	0.00525	0.99428	0.00087	0.00545
68	pu11185	1.01131	0.00093	0.00593	1.00964	0.00095	0.00588	1.0044	0.00089	0.00574
69	pu11186	1.00796	0.00097	0.00633	1.00717	0.00094	0.00634	1.00011	0.00093	0.0062
70	pu11187	1.00792	0.00088	0.00548	1.00729	0.00091	0.0054	0.99943	0.00087	0.00533
71	pust9-1	1.01886	0.00088	0.00257	1.01992	0.00082	0.00294	1.01103	0.00074	0.00259
72	pust9-2	1.0239	0.00089	0.00266	1.02331	0.00083	0.00262	1.01723	0.00078	0.00259
73	pust9-3	1.02176	0.00089	0.00246	1.0209	0.00083	0.00251	1.01755	0.00076	0.00245

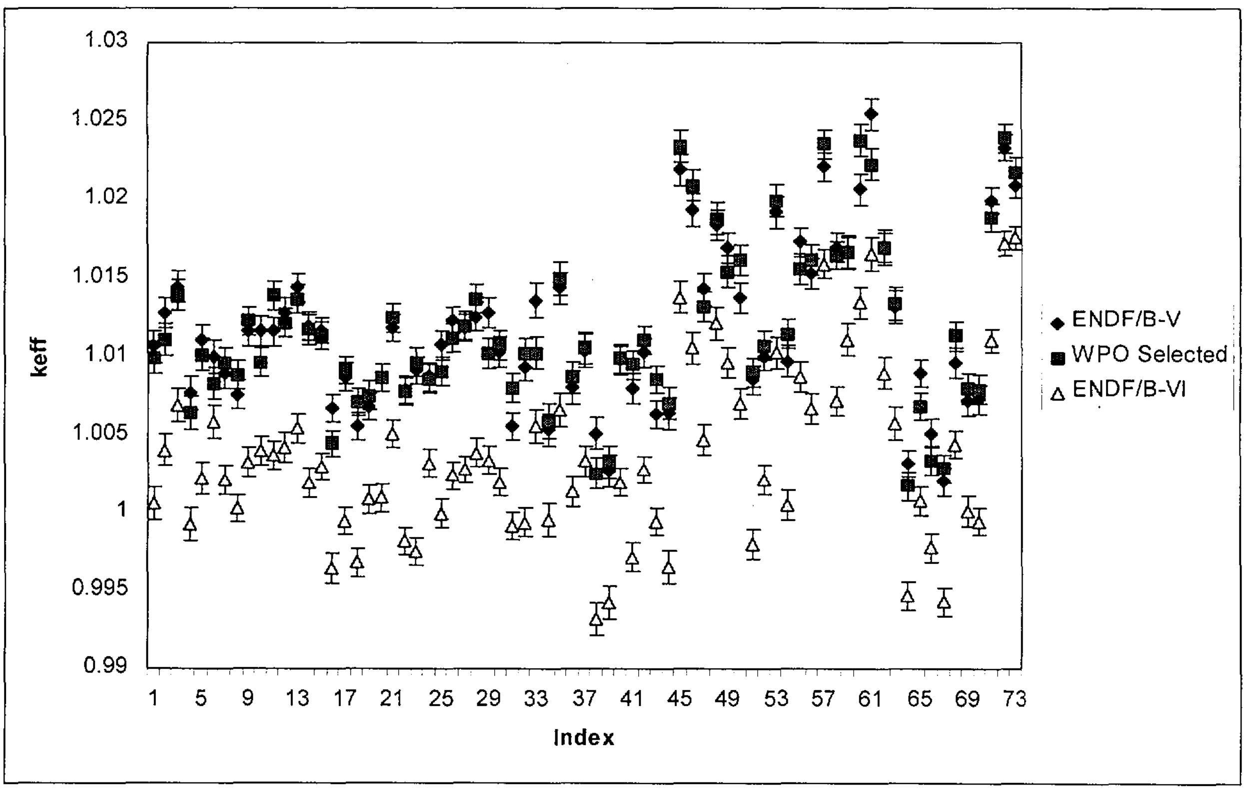

Figure 4.2-2. Homogeneous Critical Experiments Using Plutonium Nitrate Solutions (Thermal)

Table 4.2-3. Plutonium Metal Critical Experiments (Fast)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	PMF20	0.99968	0.00065	1.8886	1.00085	0.00064	1.8895	0.99878	0.00062	1.8960
2	PMF22	0.99459	0.00055	1.8932	0.99627	0.00057	1.8914	0.99707	0.00051	1.9004
3	PMF23	0.99681	0.00058	1.8011	0.99742	0.00059	1.7988	0.99869	0.00061	1.8070
4	PMF24	0.99756	0.00066	1.7421	0.99781	0.00065	1.7418	1.00122	0.00064	1.7414
5	PMF25	0.99601	0.00057	1.8316	0.99835	0.00062	1.8293	0.9971	0.00064	1.8431
6	PMF26	0.99551	0.00064	1.7318	1.00167	0.00059	1.7250	0.99754	0.00057	1.7407
7	PMF27	1.00113	0.00076	1.4768	1.00178	0.00078	1.4726	1.00258	0.00073	1.4768
8	PMF28	0.99739	0.00065	1.7166	1.00324	0.00063	1.7100	0.99891	0.0006	1.7229
9	PMF29	0.9933	0.00055	1.9188	0.99309	0.00056	1.9154	0.99463	0.00058	1.9254
10	PMF30	1.00129	0.00063	1.8152	1.00078	0.00066	1.8138	1.00176	0.00057	1.8213
11	PMF31	1.00104	0.0007	1.6059	1.00208	0.00073	1.6057	1.00305	0.00074	1.6082
12	PMF32	0.99667	0.00061	1.8192	0.99938	0.0006	1.8113	0.9964	0.00065	1.8263

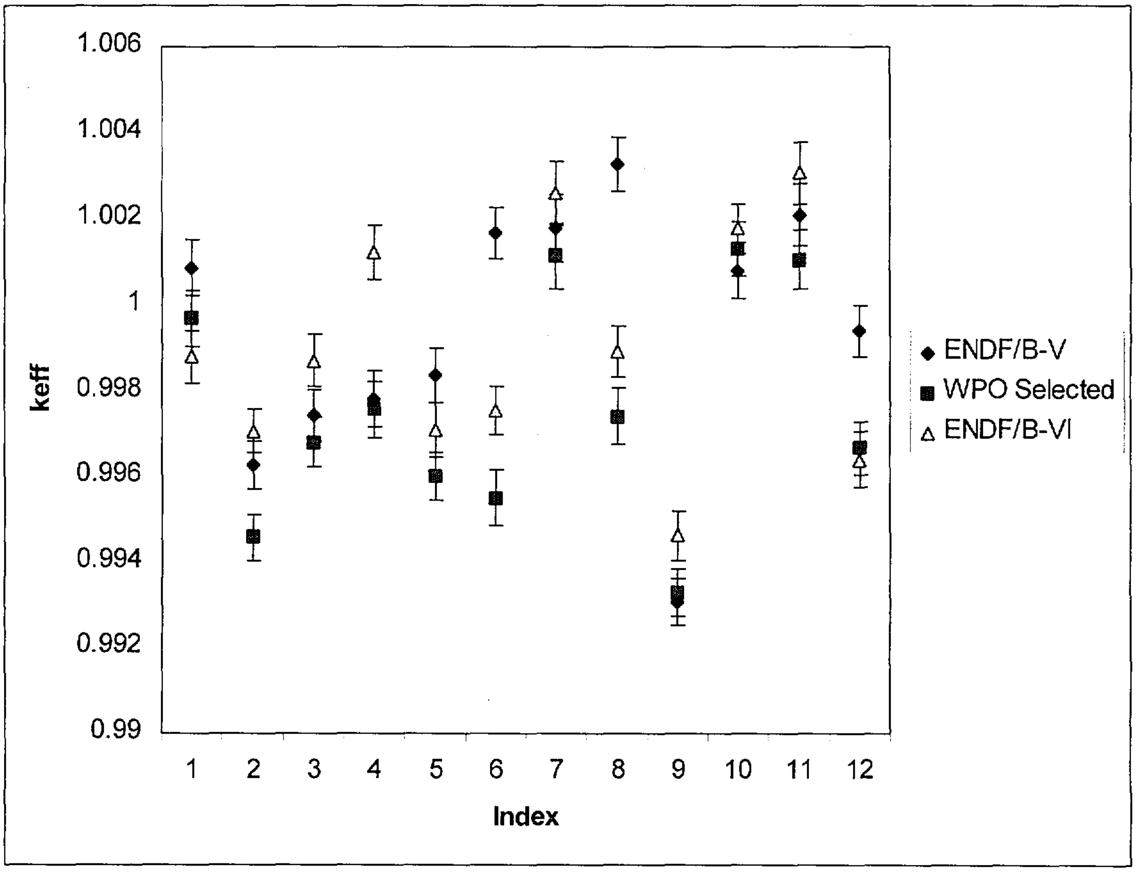

Figure 4.2-3. Plutonium Metal Critical Experiments (Fast)

Table 4.2-4. Homogenous Critical Experiments Using High Enriched Uranium Nitrate Solutions (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	hest1-1	1.00187	0.00144	0.01576	1.00241	0.00131	0.01582	0.99683	0.00138	0.01567
2	hest110	0.99468	0.00178	0.00757	0.99468	0.00178	0.00757	0.98954	0.0018	0.00758
3	hest1-2	0.99948	0.00208	0.03857	0.99816	0.00209	0.03873	0.99891	0.00208	0.03779
4	hest1-3	1.00453	0.00199	0.01546	1.00453	0.00199	0.01546	1.00161	0.00194	0.01565
5	hest131	1.00138	0.00053	0.00265	1.00135	0.0005	0.00267	0.99705	0.00055	0.0026
6	hest132	1.0002	0.00057	0.00307	1.0002	0.00057	0.00307	0.99678	0.00056	0.00307
7	hest133	0.99521	0.0006	0.00361	0.99689	0.00065	0.00378	0.99282	0.0006	0.00354
8	hest134	0.99824	0.00066	0.00379	0.99777	0.00061	0.00374	0.99519	0.00062	0.00381
9	hest1-4	1.0013	0.00203	0.0405	1.0013	0.00203	0.0405	1.00107	0.00216	0.03945
10	hest141	0.99801	0.00121	0.00727	0.99854	0.00119	0.00716	0.99204	0.00126	0.00708
11	hest142	1.01252	0.00121	0.00737	1.01307	0.00115	0.0073	1.00965	0.00106	0.0071
12	hest143	1.02159	0.00102	0.00791	1.02137	0.00109	0.00775	1.01882	0.00109	0.00756
13	hest1-5	1.00361	0.00166	0.00651	1.00361	0.00166	0.00651	0.99849	0.00159	0.00619
14	hest151	1.0039	0.00136	0.01047	1.0058	0.00123	0.01056	0.99834	0.00128	0.01048
15	hest152	0.99502	0.00126	0.01009	0.99295	0.00125	0.00998	0.98944	0.00128	0.00992
16	hest153	1.00999	0.00127	0.01127	1.00874	0.0012	0.01127	1.00941	0.00121	0.01122
17	hest154	1.01658	0.00121	0.01077	1.01579	0.00115	0.0104	1.01185	0.00122	0.01032
18	hest155	1.01218	0.00104	0.0111	1.01277	0.00116	0.01133	1.01178	0.00112	0.01077
19	hest1-6	1.01038	0.00187	0.00678	1.01038	0.00187	0.00678	1.00073	0.00177	0.00712
20	hest161	0.99346	0.00131	0.01517	0.9922	0.00137	0.01513	0.98628	0.00129	0.01531
21	hest162	1.01032	0.00125	0.015	1.01098	0.00122	0.01539	1.01039	0.0013	0.01487
22	hest163	1.02295	0.00114	0.01597	1.02046	0.00119	0.01608	1.01723	0.00122	0.01593
23	hest1-7	1.0023	0.00201	0.01501	1.0023	0.00201	0.01501	0.99985	0.002	0.01526
24	hest171	0.9966	0.00132	0.0189	0.99352	0.00125	0.01892	0.9909	0.00118	0.01885
25	hest172	0.98362	0.00126	0.02095	0.98387	0.0015	0.02097	0.97993	0.00126	0.02095
26	hest173	0.98345	0.00139	0.02004	0.98507	0.00134	0.01969	0.97985	0.00141	0.01969
27	hest174	1.00204	0.00132	0.01939	1.0036	0.00131	0.01947	1.0005	0.00134	0.01877
28	hest175	1.01159	0.00124	0.01994	1.01001	0.00121	0.01973	1.00821	0.00124	0.01927
29	hest176	1.00677	0.00127	0.02209	1.00653	0.00109	0.02183	1.00388	0.00138	0.02132
30	hest177	1.01209	0.00105	0.02045	1.01111	0.00121	0.02087	1.00984	0.00129	0.02007
31	hest178	1.00635	0.00123	0.02216	1.00409	0.00112	0.0221	1.00346	0.00122	0.02201
32	hest1-8	1.00505	0.00213	0.0161	1.00505	0.00213	0.0161	0.99318	0.0021	0.01608
33	hest181	0.99334	0.00136	0.02844	0.99364	0.00143	0.02846	0.9897	0.00125	0.02783
34	hest182	0.99132	0.00135	0.03152	0.99331	0.00142	0.03121	0.99139	0.00122	0.03053
35	hest183	0.99434	0.00143	0.0299	0.99025	0.00136	0.02976	0.98874	0.00132	0.02927
36	hest184	1.00191	0.00118	0.02895	1.00177	0.00133	0.02919	0.99834	0.00125	0.02829
37	hest185	0.99568	0.00142	0.03266	0.99727	0.00141	0.03274	0.9937	0.00142	0.03222
38	hest186	0.99667	0.00136	0.03066	0.99699	0.00129	0.03086	0.99258	0.00135	0.03063
39	hest187	1.01176	0.00134	0.02968	1.01204	0.00118	0.02988	1.00895	0.00119	0.02915
40	hest188	1.01178	0.00128	0.03335	1.01266	0.00117	0.03286	1.01154	0.00133	0.03256
41	hest189	1.00973	0.00118	0.03088	1.00855	0.00114	0.03109	1.00848	0.00127	0.03024
42	hest1-9	0.99973	0.00212	0.04099	0.99973	0.00212	0.04099	0.99817	0.00193	0.03977
43	hest191	1.0028	0.00121	0.04262	1.00102	0.00129	0.04249	1.00146	0.00133	0.04157
44	hest192	1.00401	0.00127	0.0392	1.00497	0.00116	0.03928	1.0043	0.00132	0.03851
45	hest193	0.99925	0.00111	0.04147	1.00044	0.00122	0.04166	0.99996	0.0012	0.04099
46	hest2-1	1.00513	0.00146	0.01551	1.00548	0.00148	0.01558	1.00211	0.00144	0.01534
47	hest210	1.00549	0.00185	0.0066	1.00937	0.00202	0.00663	0.99938	0.00196	0.00674
48	hest211	1.00568	0.00224	0.01551	1.00875	0.00211	0.01595	1.00142	0.00216	0.01577
49	hest212	1.0087	0.00235	0.01436	1.0127	0.00209	0.01487	1.00489	0.00211	0.0148
50	hest213	1.00564	0.00234	0.03703	0.99869	0.00232	0.03676	1.00223	0.0025	0.03659
51	hest214	1.00965	0.00238	0.03355	1.01062	0.00238	0.03377	1.0051	0.00232	0.03363
52	hest2-2	1.01236	0.00218	0.01507	1.00773	0.00235	0.01516	1.00627	0.00237	0.01434
53	hest2-3	1.00368	0.00239	0.03587	1.00219	0.0022	0.0374	0.99713	0.00256	0.03639
54	hest2-4	1.01063	0.00219	0.0346	1.00809	0.00242	0.03541	1.00836	0.00213	0.03443
55	hest2-5	1.00601	0.00213	0.0158	1.01049	0.0023	0.01622	1.00531	0.00217	0.01588
56	hest2-6	1.01719	0.00228	0.01469	1.00968	0.00215	0.01496	1.00976	0.00233	0.01493

Table 4.2-4. Homogenous Critical Experiments Using High Enriched Uranium Nitrate Solutions (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
57	hest2-7	1.00383	0.00217	0.03659	1.00691	0.00224	0.03747	0.99994	0.00248	0.03548
58	hest2-8	1.00883	0.00244	0.03423	1.01131	0.00206	0.03511	1.00552	0.0026	0.03454
59	hest2-9	1.0027	0.00204	0.00707	1.00348	0.00209	0.00654	0.9992	0.00201	0.00658
60	hest310	1.00102	0.00243	0.03817	1.00102	0.00243	0.03817	0.99709	0.00238	0.0381
61	hest311	1.00606	0.00232	0.03566	1.00606	0.00232	0.03566	1.00244	0.00231	0.03476
62	hest312	1.0074	0.00204	0.00651	1.0074	0.00204	0.00651	1.00061	0.00233	0.00715
63	hest313	1.00045	0.00185	0.00654	1.00045	0.00185	0.00654	1.00074	0.00183	0.00634
64	hest314	1.00822	0.00205	0.00704	1.00822	0.00205	0.00704	0.99809	0.00207	0.0068
65	hest315	0.9962	0.00186	0.00718	0.99675	0.0015	0.00704	0.9936	0.00143	0.00712
66	hest316	1.00356	0.00241	0.01593	1.00356	0.00241	0.01593	0.99192	0.00238	0.01574
67	hest317	1.00604	0.00213	0.01498	1.00604	0.00213	0.01498	1.00364	0.00237	0.0149
68	hest318	1.00007	0.00225	0.03842	1.00007	0.00225	0.03842	0.99356	0.00225	0.03806
69	hest319	1.01306	0.00225	0.03414	1.01306	0.00225	0.03414	1.00837	0.00213	0.03313
70	hest710	1.01186	0.00111	0.0086	1.01265	0.00111	0.00867	1.01111	0.00117	0.00848
71	hest711	1.00896	0.00124	0.03513	1.00917	0.00132	0.03589	1.00973	0.00131	0.03483
72	hest712	1.00746	0.00106	0.00855	1.00743	0.00111	0.00846	1.00327	0.0011	0.00844
73	hest713	1.00897	0.0012	0.0352	1.01049	0.00129	0.03494	1.01233	0.0012	0.03392
74	hest714	1.00811	0.00126	0.03617	1.00917	0.00132	0.03556	1.00837	0.00115	0.03517
75	hest715	1.00388	0.00134	0.03574	1.00378	0.00134	0.03612	1.00836	0.00119	0.03536
76	hest716	1.00658	0.00122	0.03632	1.00937	0.00129	0.0362	1.00585	0.00128	0.03589
77	hest717	1.00756	0.00124	0.03583	1.00651	0.00133	0.0365	1.00785	0.00124	0.03556
78	hest813	1.00616	0.0019	0.03558	1.00331	0.002	0.03616	1.00096	0.00098	0.03511
79	heust31	1.00665	0.00211	0.00657	1.00638	0.00189	0.00676	0.99825	0.00196	0.00682
80	heust32	1.00723	0.00206	0.00677	1.00635	0.002	0.00688	0.99973	0.00215	0.00702
81	heust33	1.00774	0.0024	0.01628	1.00317	0.00235	0.0158	1.00172	0.00241	0.01538
82	heust34	1.00491	0.00208	0.01539	1.00447	0.00249	0.01541	1.0022	0.00221	0.01494
83	heust35	0.99683	0.0023	0.03804	1.00436	0.00221	0.03818	0.99927	0.00243	0.0367
84	heust36	1.00523	0.0021	0.03538	1.00427	0.00241	0.03573	1.00274	0.00239	0.03419
85	heust37	1.00585	0.00173	0.00685	1.00585	0.00173	0.00685	0.99568	0.00184	0.00651
86	heust38	1.01086	0.00202	0.01639	1.01086	0.00202	0.01639	1.00251	0.00227	0.01597
87	heust39	1.01063	0.00204	0.01512	1.01063	0.00204	0.01512	0.99982	0.00216	0.01501
88	heust71	1.01399	0.00115	0.00703	1.01081	0.00101	0.00702	1.01043	0.00117	0.00708
89	heust72	1.01391	0.00133	0.03607	1.01474	0.0013	0.03612	1.01458	0.00123	0.03545
90	heust73	1.00773	0.00101	0.00713	1.00692	0.00107	0.00687	1.00283	0.00103	0.00687
91	heust74	1.0125	0.00123	0.0351	1.01389	0.00125	0.0354	1.01277	0.00128	0.03466
92	heust75	1.00867	0.00101	0.0084	1.00713	0.00115	0.00836	1.00215	0.00111	0.00815
93	heust76	1.00684	0.00136	0.03767	1.00493	0.00146	0.03792	1.00446	0.00129	0.0372
94	heust77	1.00792	0.0011	0.00843	1.00635	0.00103	0.00829	1.00006	0.00117	0.00811
95	heust78	1.00333	0.00128	0.03825	1.00478	0.00135	0.03823	1.0032	0.00134	0.03757
96	heust79	1.00834	0.00114	0.00899	1.00519	0.00123	0.00883	1.00464	0.00129	0.00855
97	heust81	1.00193	0.00138	0.00666	1.00316	0.00134	0.00661	0.99928	0.0012	0.00651
98	heust83	0.9973	0.0019	0.00644	0.9973	0.0019	0.00644	0.99185	0.00116	0.00642
99	heust86	1.0089	0.00221	0.03785	1.00969	0.0023	0.03669	1.00546	0.00215	0.03642
100	heust89	1.00274	0.00129	0.00643	1.00373	0.00116	0.0066	0.9977	0.00115	0.0061
101	hst1810	1.02605	0.00121	0.03443	1.02526	0.00114	0.03464	1.02652	0.00132	0.03392
102	hst1811	1.0288	0.00115	0.03182	1.02814	0.00115	0.03203	1.02612	0.00102	0.03183
103	hst1812	1.01997	0.00093	0.03276	1.01861	0.00101	0.03353	1.02152	0.00115	0.03246
104	hst121	1.0035	0.0004	0.0027	1.0035	0.0004	0.0027	0.9993	0.0004	0.0027
105	hst321	0.9991	0.0003	0.0022	0.9996	0.0003	0.0021	0.9972	0.0003	0.0021

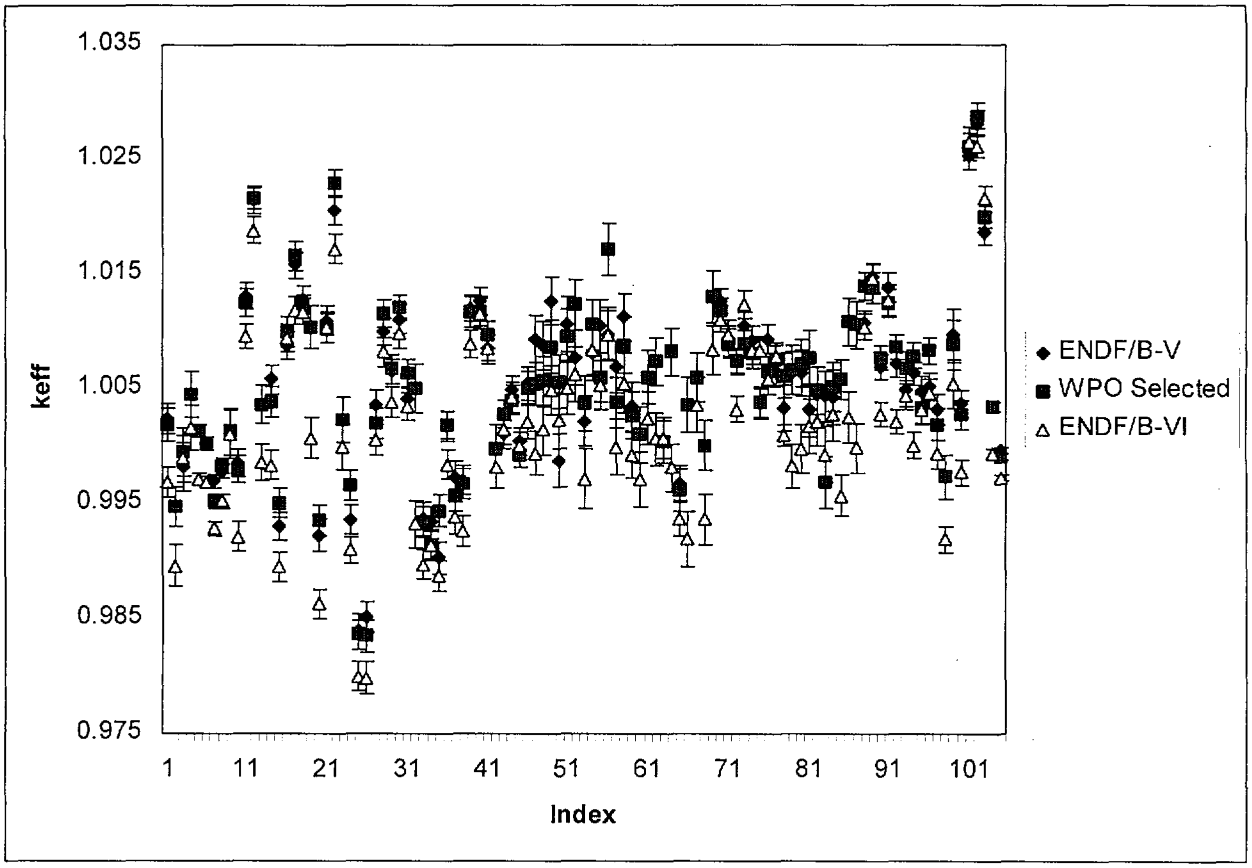

Figure 4.2-4. Homogenous Critical Experiments Using High Enriched Uranium Nitrate Solutions (Thermal)

Table 4.2-5. High Enriched Uranium Metal Critical Experiments (Fast)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	HMF1G	0.9986	0.0004	1.5681	0.9986	0.0004	1.5681	0.9970	0.0004	1.4893
2	HMF3Ni	1.0036	0.0005	1.3649	1.0145	0.0005	1.3555	1.0045	0.0005	1.2964
3	HMF8	0.99477	0.00059	1.5503	0.9951	0.00058	1.5501	0.99281	0.00059	1.4766
4	HMF11	0.99264	0.00078	1.1620	0.99348	0.00079	1.1607	0.99717	0.00081	1.1026
5	HMF12	0.99446	0.00044	1.5222	0.99399	0.00043	1.5237	0.9944	0.00043	1.4486
6	HMF13	0.9962	0.00061	1.4860	0.99997	0.00061	1.4838	0.99478	0.00063	1.4155
7	HMF14	0.99773	0.00058	1.5443	0.99831	0.00062	1.5465	0.99501	0.00063	1.4893
8	HMF15	0.9936	0.00062	1.5808	0.99263	0.00056	1.5813	0.99172	0.00061	1.5000
9	HMF18	0.9978	0.00056	1.5522	0.99805	0.00056	1.5515	0.99837	0.00058	1.4753
10	HMF19	1.00306	0.0006	1.4765	1.00277	0.0006	1.4793	1.00328	0.0006	1.4018
11	HMF20	0.99643	0.0006	1.4333	0.99574	0.00061	1.4341	0.99839	0.00069	1.3635
12	HMF21	0.99727	0.0006	1.4481	1.00192	0.0006	1.4431	0.99463	0.00061	1.3807
13	HMF22	0.99252	0.00064	1.5039	0.99272	0.0006	1.5049	0.99188	0.0006	1.4299
14	HMF24	0.99401	0.00108	1.2504	0.99521	0.00109	1.2553	0.99443	0.00109	1.1917
15	HMF28	1.0040	0.0005	1.5979	1.0040	0.0005	1.5979	1.0026	0.0005	1.5498

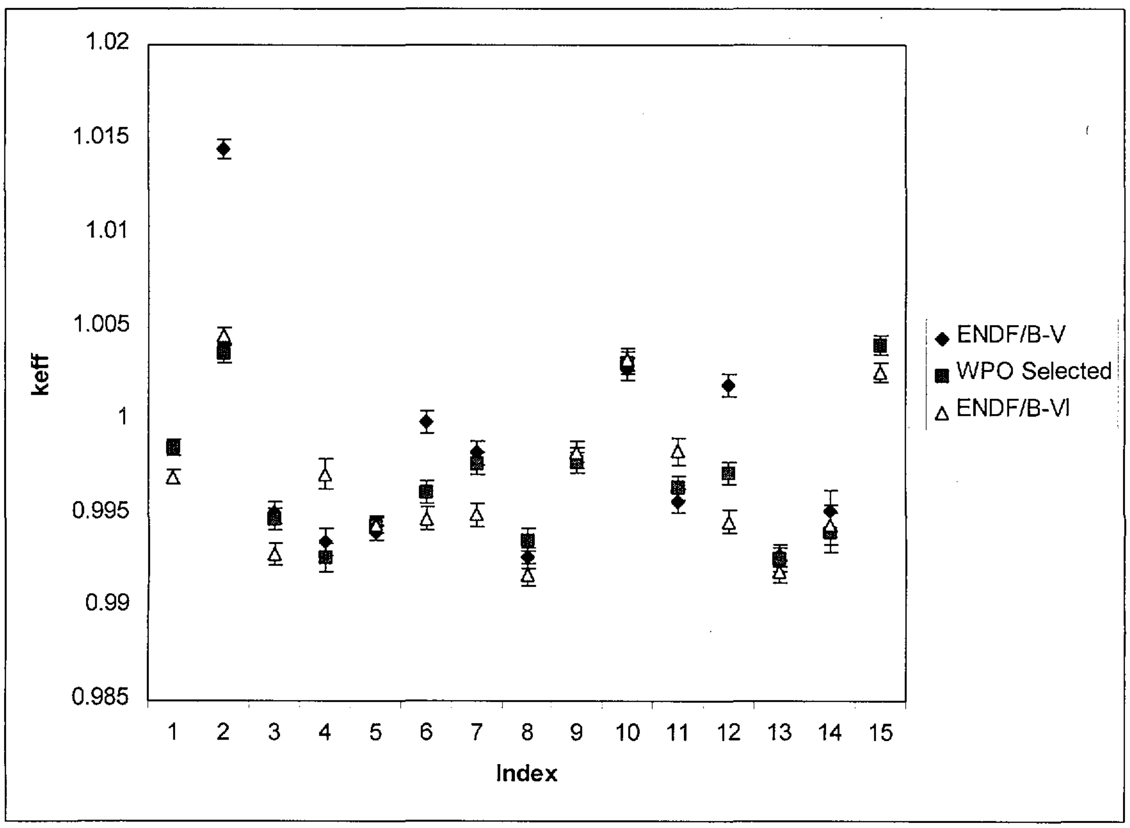

Figure 4.2-5. High Enriched Uranium Metal Critical Experiments (Fast)

Table 4.2-6. Homogenous Critical Experiments Using Intermediate Enriched Uranium Fuel (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	IECT104	0.99735	0.00105	0.07405	0.99735	0.00105	0.07405	1.00316	0.00102	0.07270
2	IECT105	1.00847	0.00091	0.04552	1.00847	0.00091	0.04552	1.00785	0.00082	0.04520
3	IECT113	0.99674	0.00103	0.07430	0.99674	0.00103	0.07430	1.00295	0.00096	0.07350
4	IECT114	0.99787	0.00094	0.07375	0.99787	0.00094	0.07375	1.00358	0.00098	0.07261
5	IECT115	0.99811	0.00099	0.07400	0.99811	0.00099	0.07400	1.00365	0.001	0.07309
6	IECT116	1.0021	0.00093	0.05547	1.0021	0.00093	0.05547	1.00414	0.00092	0.05435
7	IECT119	1.00255	0.00089	0.06026	1.0045	0.00101	0.06114	1.00479	0.00096	0.06006
8	IECT125	0.99874	0.0009	0.05992	0.99874	0.0009	0.05992	1.00527	0.00096	0.05891
9	IECT126	1.0057	0.00099	0.05695	1.00443	0.00095	0.05663	1.00838	0.00089	0.05559
10	IECT127	1.00328	0.00084	0.05616	1.0032	0.00086	0.05633	1.00637	0.00094	0.05607

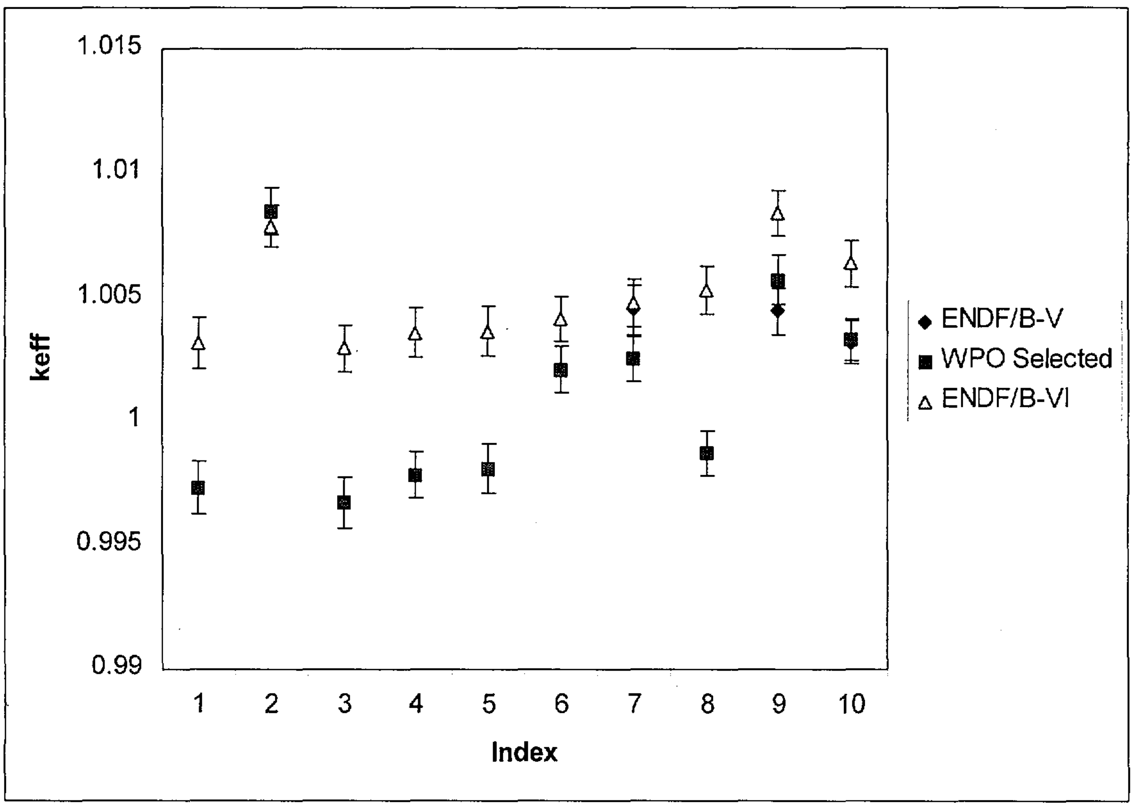

Figure 4.2-6. Homogenous Critical Experiments Using Intermediate Enriched Uranium Fuel (Thermal)

Table 4.2-7. Homogenous Critical Experiments Using Intermediate Enriched Uranium Fuel (Intermediate)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	IECT101	0.99735	0.00092	0.21679	0.99735	0.00092	0.2168	1.01062	0.00088	0.21085
2	IECT102	0.99601	0.00092	0.15817	0.99601	0.00092	0.15817	1.00748	0.00097	0.15427
3	IECT103	0.99305	0.00104	0.10412	0.99305	0.00104	0.10412	1.0024	0.00103	0.10243
4	IECT106	1.00026	0.00103	0.10793	1.00026	0.00103	0.10793	1.00871	0.00099	0.10467
5	IECT107	0.998	0.00101	0.11064	0.998	0.00101	0.11064	1.00645	0.00099	0.10997
6	IECT108	0.99604	0.00101	0.11867	0.99604	0.00101	0.11867	1.00358	0.00088	0.11611
7	IECT109	1.00043	0.00084	0.1679	1.00043	0.00084	0.1679	1.01287	0.00094	0.16326
8	IECT110	0.99672	0.00099	0.15756	0.99672	0.00099	0.15756	1.00845	0.00097	0.15359
9	IECT111	0.99579	0.00096	0.15732	0.99579	0.00096	0.15732	1.00818	0.00105	0.15255
10	IECT112	0.99642	0.00101	0.15568	0.99642	0.00101	0.15568	1.00601	0.00095	0.15347
11	IECT117	0.99503	0.00113	0.20838	0.99651	0.00102	0.20814	1.01029	0.00102	0.20117
12	IECT118	0.99852	0.00104	0.13376	0.99756	0.00113	0.13428	1.00986	0.00106	0.13125
13	IECT120	1.0005	0.00094	0.15539	1.0005	0.00094	0.15539	1.01303	0.00101	0.15118
14	IECT121	0.99884	0.00087	0.21334	0.99884	0.00087	0.21334	1.01392	0.00088	0.20757
15	IECT122	0.999	0.00109	0.19772	0.999	0.00109	0.19772	1.01168	0.00102	0.19125
16	IECT123	0.99516	0.00107	0.12826	0.99516	0.00107	0.12826	1.00429	0.00111	0.12582
17	IECT124	1.00036	0.00105	0.13305	1.00036	0.00105	0.13305	1.0123	0.001	0.12898
18	IECT128	1.00637	0.00095	0.1583	1.00506	0.00091	0.15824	1.01497	0.00092	0.1536
19	IECT129	1.00472	0.001	0.15103	1.00123	0.00099	0.15184	1.01219	0.00098	0.1472
20	ist1-1[a]	0.98509	0.00111	0.015	0.98074	0.00108	0.0149	0.97915	0.00106	0.0146
21	ist1-2[a]	0.97695	0.0012	0.0209	0.97596	0.00121	0.0207	0.97348	0.00118	0.0208
22	ist1-3[a]	0.97306	0.00113	0.0207	0.97222	0.00119	0.0205	0.97079	0.00117	0.0208
23	ist1-4[a]	0.97124	0.00126	0.0273	0.97105	0.00124	0.0275	0.96757	0.00126	0.0271

NOTE: [a] Cases ist1-1 through ist1-4 were homogenous critical experiments using uranyl-sulphate solutions

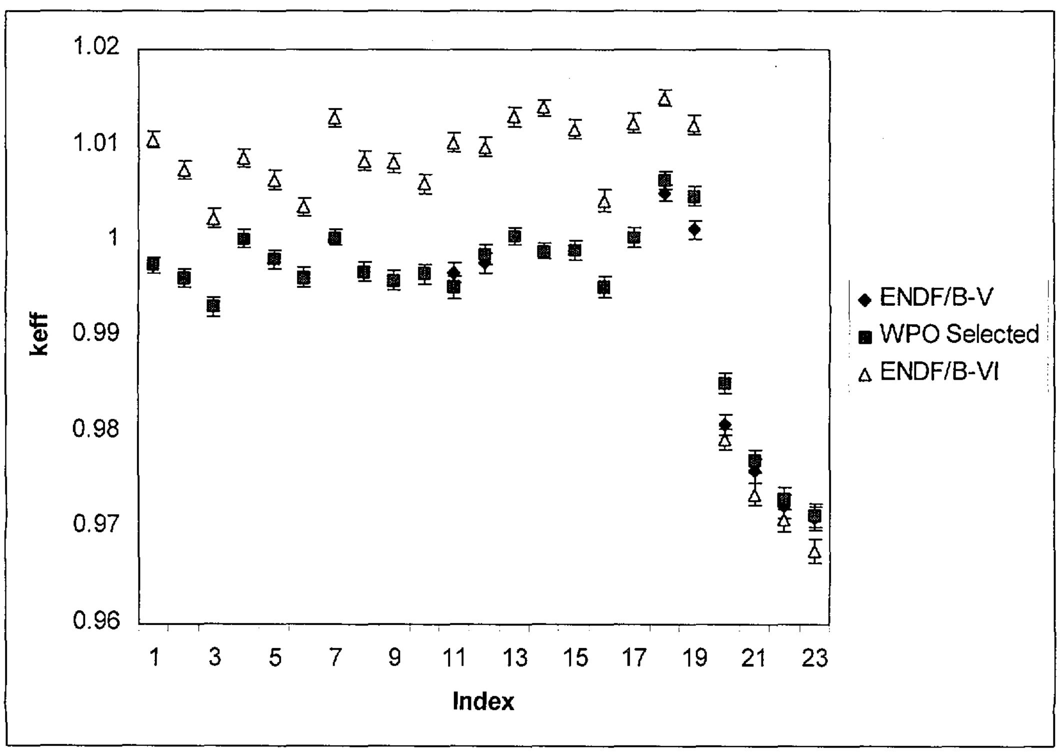

Figure 4.2-7. Homogenous Critical Experiments Using Intermediate Enriched Uranium Fuel (Intermediate)

Table 4.2-8. Intermediate Enriched Uranium Metal Critical Experiments (Fast)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	imf1-1	1.00183	0.00028	1.4395	1.00224	0.00029	1.4394	0.99714	0.00028	1.3910
2	imf1-2	1.00254	0.00028	1.4403	1.00225	0.00028	1.4398	0.99666	0.00027	1.3919
3	imf1-3	1.004	0.00027	1.3862	1.00401	0.00029	1.3860	0.9986	0.00028	1.3515
4	imf1-4	1.00459	0.00028	1.3848	1.00477	0.00028	1.3859	0.999	0.00026	1.3495
5	imf2-1	1.00594	0.0006	1.2784	1.00594	0.0006	1.2784	1.00327	0.00056	1.2650
6	imf3-1	1.00096	0.0008	1.3526	0.99925	0.00076	1.3502	0.99539	0.00076	1.3242
7	imf4-1	1.00884	0.00078	1.3076	1.01048	0.00084	1.3071	1.00446	0.00081	1.2768
8	imf5-1	1.00632	0.00082	1.2872	1.00881	0.00079	1.2852	1.00053	0.00081	1.2597
9	imf6-1	0.99727	0.00083	1.2915	0.99597	0.00081	1.2892	0.99173	0.00083	1.2610
10	imf8-1	1.00845	0.00079	1.3639	1.0087	0.00084	1.3650	1.0038	0.00081	1.3337

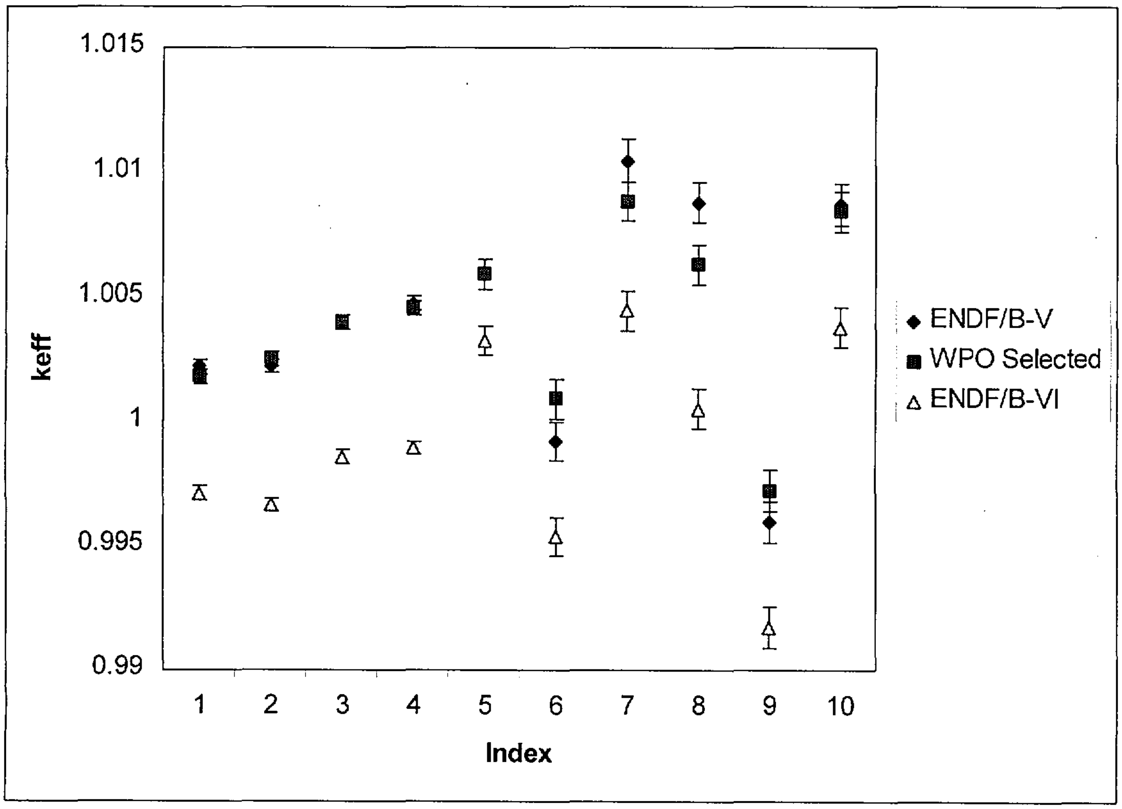

Figure 4.2-8. Intermediate Enriched Uranium Metal Critical Experiments (Fast)

Table 4.2-9. Homogenous Critical Experiments Using Low Enriched Uranium Nitrate Solutions (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_{eff}	σ	AENCF	k_{eff}	σ	AENCF	k_{eff}	σ	AENCF
1	leuja01	1.00425	0.00085	0.01896	1.00354	0.00076	0.01899	0.99826	0.00091	0.01905
2	leuja14	0.99755	0.00094	0.02001	0.99809	0.00094	0.01991	0.9941	0.0009	0.01939
3	leuja29	1.00377	0.00082	0.01806	1.00452	0.00087	0.01811	0.99854	0.00081	0.0176
4	leuja30	0.99885	0.00086	0.01881	1.00084	0.00083	0.0187	0.99687	0.00091	0.01845
5	leuja32	1.00143	0.00086	0.01757	0.9992	0.00088	0.0177	0.99529	0.00088	0.01733
6	leuja33	0.99961	0.0009	0.01662	1.00169	0.00084	0.01681	0.99687	0.00077	0.01661
7	leuja34	1.0029	0.00079	0.0159	1.00277	0.00092	0.01607	1.00017	0.00069	0.01602
8	leuja36	1.00185	0.00084	0.01665	1.00081	0.0008	0.01651	0.99742	0.0008	0.0164
9	leuja46	1.00311	0.0008	0.01535	1.00433	0.00081	0.01556	0.99972	0.00073	0.01536
10	leuja49	0.99875	0.00078	0.01593	0.99861	0.00072	0.0159	0.9957	0.00084	0.01554
11	leuja51	1.00279	0.0007	0.01479	1.00337	0.00071	0.01493	0.99795	0.00078	0.01461
12	leuja54	1.00246	0.00072	0.0144	1.00101	0.00067	0.01417	0.99904	0.00064	0.01428
13	leust21	0.99892	0.00053	0.02487	0.99855	0.00058	0.02513	0.99532	0.00056	0.02492
14	leust22	0.99469	0.00061	0.02832	0.99659	0.00064	0.0283	0.991	0.00067	0.02802
15	leust23	1.00078	0.00057	0.02665	1.0009	0.0006	0.02684	0.99624	0.00058	0.02615
16	lst1-1	1.01069	0.00085	0.0523	1.01182	0.00101	0.05186	1.00714	0.00092	0.0514
17	lst3-1	0.99784	0.00069	0.0185	0.9993	0.0004	0.0186	0.99274	0.00067	0.0186
18	lst3-2	0.99898	0.00068	0.0165	0.99705	0.00038	0.0166	0.99335	0.00066	0.0165
19	lst3-3	1.00141	0.00066	0.0164	1.00151	0.00037	0.0164	0.99701	0.00065	0.0161
20	lst3-4	0.99406	0.00061	0.016	0.99536	0.00038	0.0162	0.98964	0.00063	0.0161
21	lst3-5	0.99799	0.00056	0.0131	0.99897	0.00031	0.0133	0.99468	0.00056	0.0128
22	lst3-6	0.99935	0.00053	0.013	0.99924	0.0003	0.0129	0.99507	0.00053	0.0126
23	lst3-7	0.99917	0.0005	0.0126	0.99716	0.0003	0.0127	0.99397	0.00051	0.0124
24	lst3-8	1.00116	0.00044	0.0115	1.00079	0.00025	0.0114	0.99637	0.00045	0.0113
25	lst3-9	0.99779	0.00048	0.0114	0.99732	0.00025	0.0114	0.99603	0.00041	0.0112

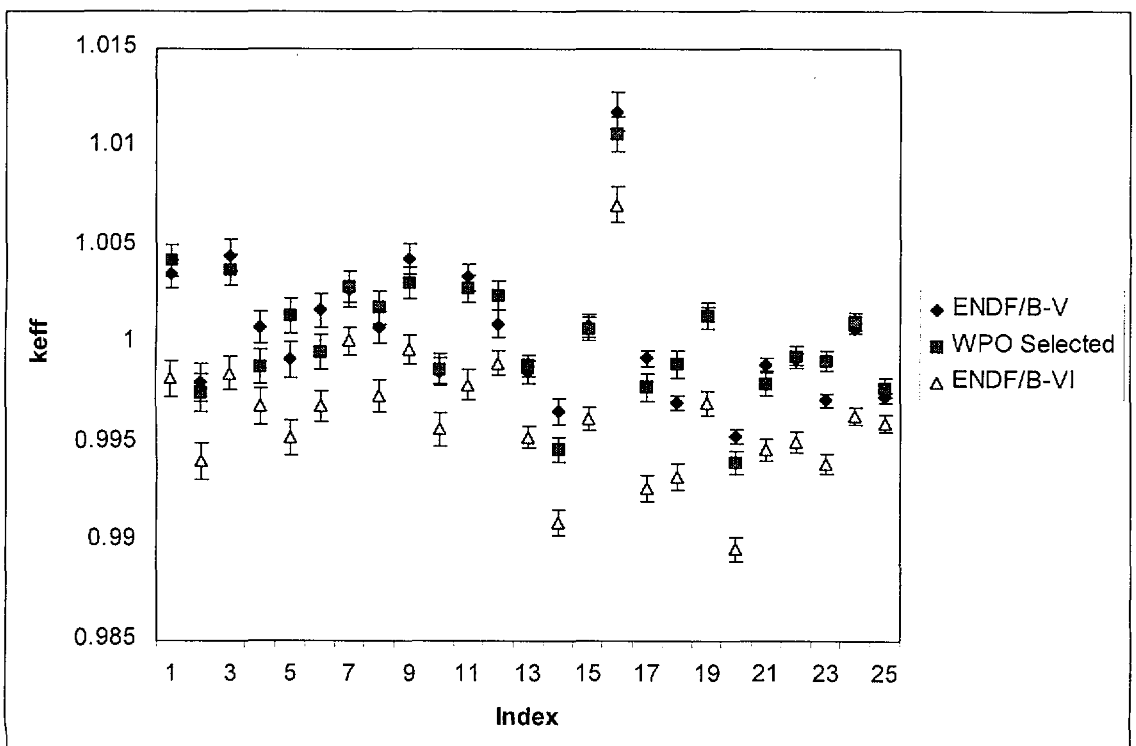

Figure 4.2-9. Homogenous Critical Experiments Using Low Enriched Uranium Nitrate Solutions (Thermal)

Table 4.2-10. Homogenous Critical Experiments Using Low Enriched Uranium Solutions (Intermediate)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_∞	σ	AENCF	k_∞	σ	AENCF	k_∞	σ	AENCF
1	sphu9a	0.99004	0.00249	0.2541	0.99004	0.00249	0.2541	0.98665	0.00293	0.2562
2	sphu9b	0.99269	0.00249	0.2163	0.99269	0.00249	0.2163	0.98518	0.00271	0.2155
3	sphu9c	0.97871	0.00256	0.1883	0.97871	0.00256	0.1883	0.97958	0.0019	0.1839
4	sphu9d	0.97914	0.00242	0.1737	0.97914	0.00242	0.1737	0.97132	0.00233	0.1740
5	sphu9e	0.96607	0.00163	0.1591	0.96607	0.00163	0.1591	0.96429	0.00221	0.1660
6	sphu9f	1.00952	0.00261	0.2511	1.00952	0.00261	0.2511	1.00009	0.00306	0.2543
7	sphu9g	1.0136	0.00246	0.1839	1.0136	0.00246	0.1839	1.00763	0.00235	0.1906
8	sphu9h	0.99713	0.00198	0.1651	0.99713	0.00198	0.1651	0.99364	0.00224	0.1642
9	sphu9i	1.03372	0.00274	0.2495	1.03372	0.00274	0.2495	1.03156	0.00233	0.2499
10	sphu9j	1.04207	0.00224	0.1783	1.04207	0.00224	0.1783	1.03178	0.0024	0.1760
11	sphu9k	1.02951	0.00216	0.1661	1.02951	0.00216	0.1661	1.02122	0.00206	0.1670
12	sphu9l	1.02281	0.0021	0.1549	1.02281	0.0021	0.1549	1.0191	0.00237	0.1545

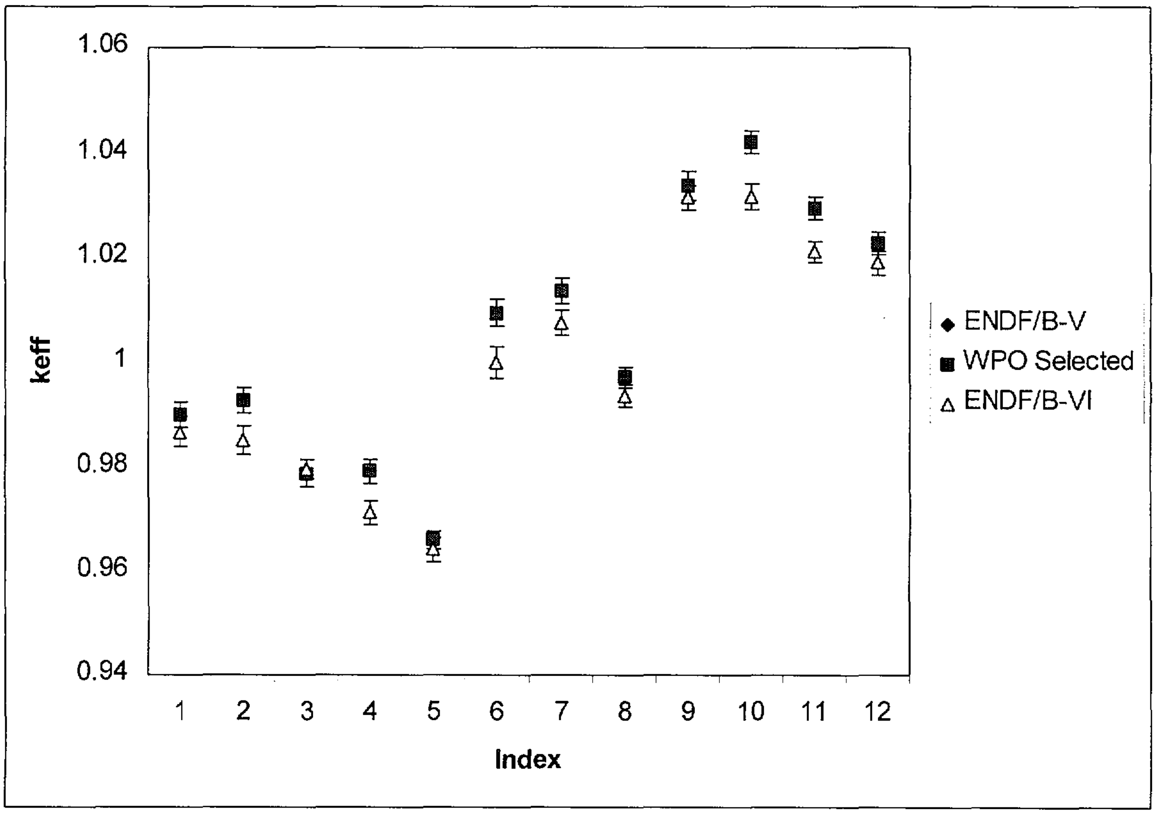

Figure 4.2-10. Homogenous Critical Experiments Using Low Enriched Uranium Solutions (Intermediate)

Table 4.2-11. Homogenous Critical Experiments Using Uranium-233 Fuel (Thermal)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_∞	σ	AENCF	k_∞	σ	AENCF	k_∞	σ	AENCF
1	u233s1	1.00153	0.00037	0.03738	1.00179	0.00039	0.00371	0.99824	0.00038	0.00367
2	u233s2	1.00029	0.00038	0.00390	1.00069	0.00038	0.00398	0.99773	0.00038	0.00404
3	u233s3	1.00045	0.00040	0.00402	1.00007	0.00039	0.00416	0.9969	0.00038	0.00413
4	u233s4	0.99951	0.00040	0.00432	0.99987	0.00039	0.00422	0.99667	0.0004	0.00423
5	u233s5	0.99856	0.00039	0.00435	0.99998	0.00041	0.00440	0.99524	0.00041	0.00437
6	u233s6	0.99826	0.00027	0.00301	0.99911	0.00025	0.00309	0.99534	0.00026	0.00311

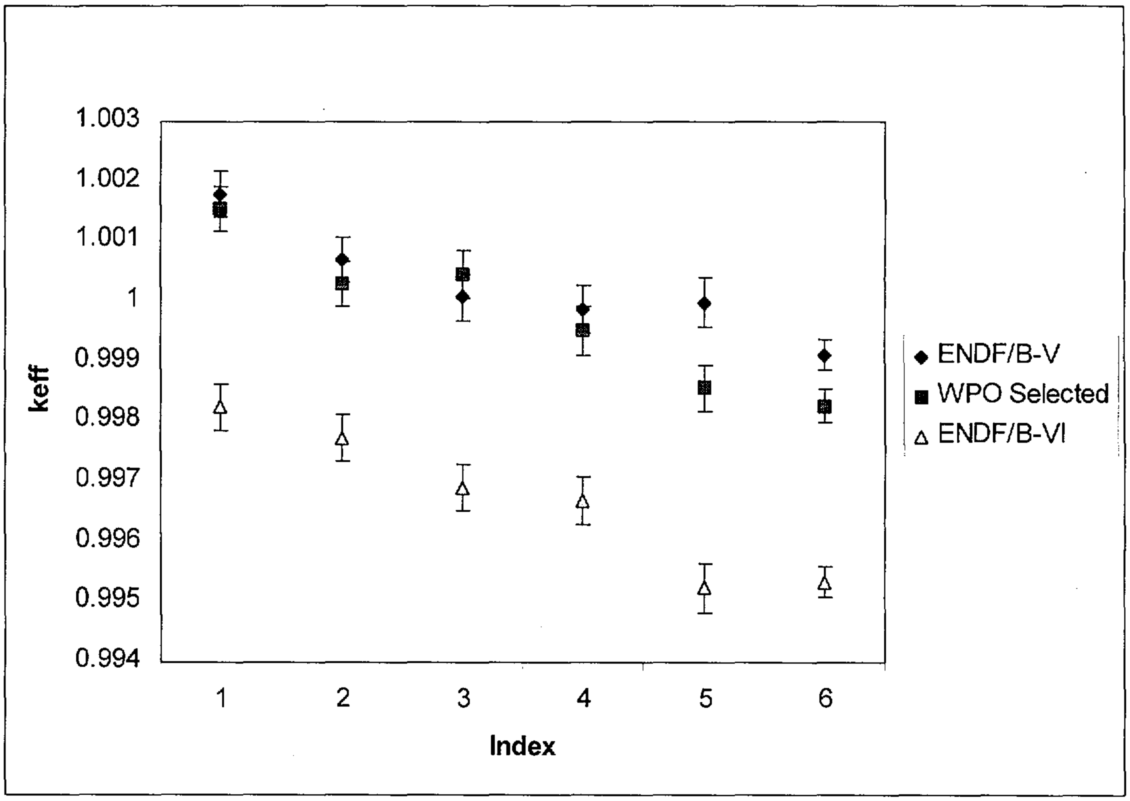

Figure 4.2-11. Homogenous Critical Experiments Using Uranium-233 Fuel (Thermal)

Table 4.2-12. Homogenous Critical Experiments Using Uranium-233 Fuel (Fast)

#	Case	WPO Selected			ENDF/B-V			ENDF/B-VI		
		k_∞	σ	AENCF	k_∞	σ	AENCF	k_∞	σ	AENCF
1	u2331a	0.99297	0.00038	1.77385	0.99297	0.00038	1.77385	0.99354	0.00038	1.77170
2	u2332a	0.99547	0.00038	1.73702	0.99547	0.00038	1.73702	0.99609	0.0004	1.72533
3	u2332b	0.99807	0.00039	1.70789	0.99807	0.00039	1.70789	0.99764	0.00039	1.68809
4	u2333a	0.99583	0.00041	1.74832	0.99583	0.00041	1.74832	0.99645	0.0004	1.74824
5	u2333b	0.99771	0.00041	1.76231	0.99771	0.00041	1.76231	0.99738	0.00042	1.76086
6	u2334a1	1.00380	0.00041	1.61187	1.00356	0.00041	1.61336	1.00278	0.00043	1.61505
7	u2334b1	1.00705	0.00042	1.51777	1.00637	0.00043	1.51775	1.00564	0.00044	1.52050
8	u2335a	0.99351	0.00043	1.61950	0.99351	0.00043	1.61950	0.99452	0.00042	1.61670
9	u2335b	0.99681	0.00045	1.51871	0.99681	0.00045	1.51871	0.99707	0.00045	1.51252
10	u2336a	1.00057	0.00045	1.77403	1.00057	0.00045	1.77403	1.00122	0.00047	1.77535

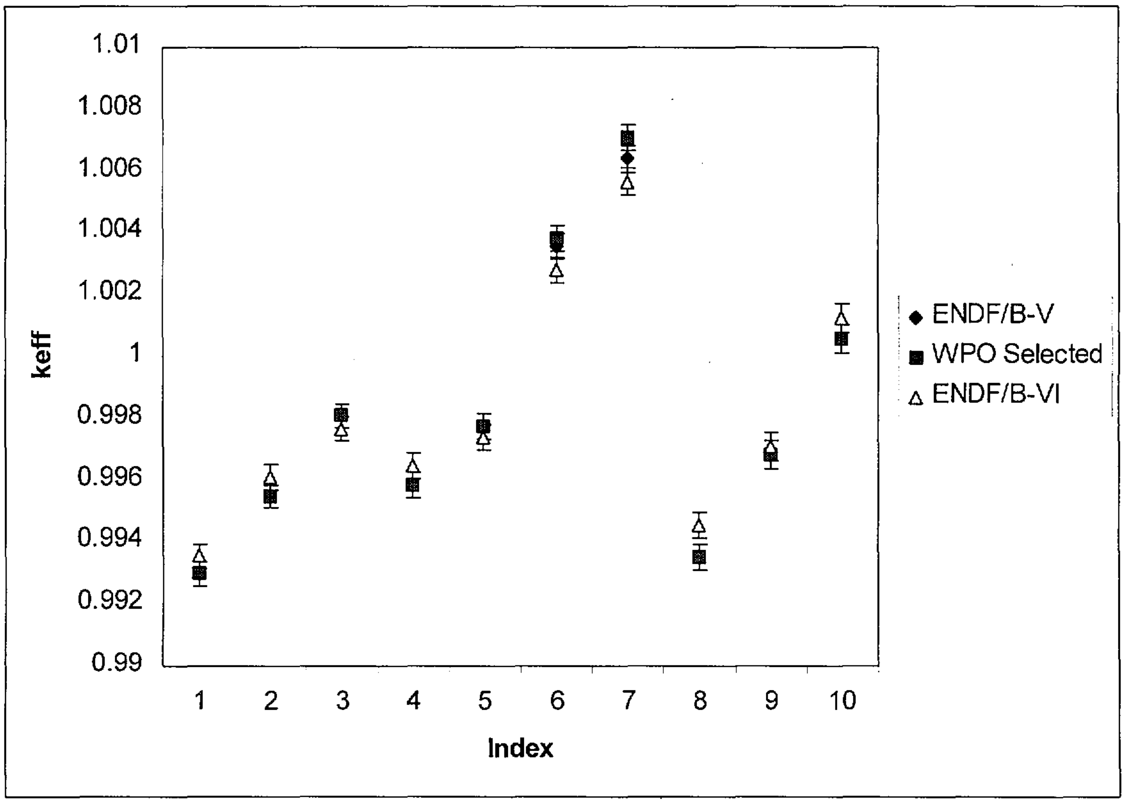

Figure 4.2-12. Homogenous Critical Experiments Using Uranium-233 Fuel (Fast)

5. CONCLUSIONS

The data reported herein not identified with TBVs are acceptable for quality affecting activities and for use in analyses affecting procurement, construction, or fabrication. TBV data specified in Section 4 will require the release of TBVs-1357 through 1371 prior to its use in quality affecting activities and for use in analyses affecting procurement, construction, or fabrication. Table 5-1 provides a summary listing of the average calculated values of k_{eff} for the LCEs with each library set. The values listed in Table 5-1 are based on unaccepted data, if the data from Table 5-1 are used as input into documents directly relied upon for radiological safety or waste isolation issues, then they are required to be identified and tracked as TBV in accordance with appropriate procedures.

Table 5-1. Average Values for k_{eff} from LCE Results

Type of System	Number of Cases	Average k_{eff} / Standard Deviation		
		WPO Selected	ENDF/B-V	ENDF/B-VI
Lattices				
Mixed oxide	18	1.0007 / 0.0044	1.0002 / 0.0049	0.9944 / 0.0038
HEU	77	1.0004 / 0.0076	0.9995 / 0.0076	0.9990 / 0.0071
IEU	2	1.0111 / 0.0029	1.0083 / 0.0013	1.0097 / 0.0032
LEU	65	0.9984 / 0.0043	0.9979 / 0.0039	0.9947 / 0.0045
Homogeneous Systems				
Mixed Pu and natural U nitrate	34	1.0043 / 0.0053	1.0045 / 0.0056	0.9970 / 0.0055
Pu nitrate	73	1.0117 / 0.0053	1.0117 / 0.0051	1.0036 / 0.0056
Pu metal fast	12	0.9976 / 0.0027	0.9994 / 0.0029	0.9990 / 0.0026
HEU nitrate	105	1.0055 / 0.0075	1.0054 / 0.0073	1.0020 / 0.0082
HEU metal fast	15	0.9971 / 0.0038	0.9985 / 0.0057	0.9967 / 0.0040
IEU	33	0.9965 / 0.0085	0.9962 / 0.0087	1.0036 / 0.0122
IEU metal fast	10	1.0041 / 0.0036	1.0042 / 0.0045	0.9991 / 0.0041
LEU nitrate	25	1.0004 / 0.0034	1.0005 / 0.0034	0.9963 / 0.0034
LEU	12	1.0046 / 0.0243	1.0046 / 0.0243	0.9993 / 0.0230
^{233}U	16	0.9988 / 0.0036	0.9989 / 0.0035	0.9977 / 0.0031

Based upon examinations of the average k_{eff} values presented in Table 5-1 it can be seen that the ENDF/B-V and WPO Selected library sets produce k_{eff} values that are closer to unity than the ENDF/B-VI library values for the lattice LCEs. For the homogenous systems, most of the ENDF/B-VI library results were closer to unity than the ENDF/B-V and WPO Selected library sets.

6. REFERENCES

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