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**INTEGRAL TESTING OF THORIUM AND
U233 DATA FOR THERMAL REACTORS
(AWBA Development Program)**

MASTER

JUNE 1979

CONTRACT EY-76-C-11-0014

**BETTIS ATOMIC POWER LABORATORY
WEST MIFFLIN, PENNSYLVANIA**

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INTEGRAL TESTING OF THORIUM AND U233
DATA FOR THERMAL REACTORS
(AWBA DEVELOPMENT PROGRAM)

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CONTRACT NO. EY-76-C-11-0014

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FOREWORD

The Shippingport Atomic Power Station located in Shippingport, Pennsylvania was the first large-scale, central-station nuclear power plant in the United States and the first plant of such size in the world operated solely to produce electric power. This program was started in 1953 to confirm the practical application of nuclear power for large-scale electric power generation. It has provided much of the technology being used for design and operation of the commercial, central-station nuclear power plants now in use.

Subsequent to development and successful operation of the Pressurized Water Reactor in the DOE-owned reactor plant at the Shippingport Atomic Power Station, the Atomic Energy Commission in 1965 undertook a research and development program to design and build a Light Water Breeder Reactor core for operation in the Shippingport Station.

The objective of the Light Water Breeder Reactor (LWBR) program has been to develop a technology that would significantly improve the utilization of the nation's nuclear fuel resources employing the well-established water reactor technology. To achieve this objective, work has been directed toward analysis, design, component tests, and fabrication of a water-cooled, thorium oxide fuel cycle breeder reactor for installation and operation at the Shippingport Station. The LWBR core started operation in the Shippingport Station in the Fall of 1977 and is expected to be operated for about 3 to 4 years. At the end of this period, the core will be removed and the spent fuel shipped to the Naval Reactors Expended Core Facility for a detailed examination to verify core performance including an evaluation of breeding characteristics.

In 1976, with fabrication of the Shippingport LWBR core nearing completion, the Energy Research and Development Administration established the Advanced Water Breeder Applications (AWBA) program to develop and disseminate technical information which would assist U. S. industry in evaluating the LWBR concept for commercial-scale applications. The program will explore some of the problems that would be faced by industry in adapting technology confirmed in the LWBR program. Information to be developed includes concepts for commercial-scale prebreeder cores which would produce uranium-233 for light water breeder cores while producing electric power, improvements for breeder cores based on the technology developed to fabricate and operate the Shippingport LWBR core, and other information and technology to aid in evaluating commercial-scale application of the LWBR concept.

All three development programs (Pressurized Water Reactor, Light Water Breeder Reactor, and Advanced Water Breeder Applications) have been administered by the Division of Naval Reactors with the goal of developing practical improvements in the utilization of nuclear fuel resources for generation of electrical energy using water-cooled nuclear reactors.

Technical information developed under the Shippingport, LWBR, and AWBA programs has been and will continue to be published in technical memoranda, one of which is this present report.

ABSTRACT

A survey is made of integral experiments useful for testing thorium and U233 nuclear data in thermal reactor applications. Emphasis is on homogeneous U233-H₂O criticals and simple, water-moderated U233-thorium and U235-thorium lattice experiments. Thorium-U233-graphite experiments are also discussed briefly. Although the available experiments provide a fairly consistent test of important nuclear data, their accuracy and scope leave much to be desired. In detailed Monte Carlo analyses, ENDF/B-IV data are found to perform reasonably well. Adequate (though partly fortuitous) agreement is found with integral measurements of thorium resonance capture in lattices. A new, harder fission spectrum for U233 can correct the principal discrepancy observed with ENDF/B-IV, a bias trend in K_{eff} attributed to an underprediction of leakage.

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Integral Testing of Thorium and U233
Data for Thermal Reactors*

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I. Introduction

This paper surveys U233 and thorium critical experiments of potential use as thermal reactor data-testing benchmarks. Two major questions are addressed: (1) To what extent do available integral experiments provide a consistent and adequately detailed test of the differential data? (2) How well do current differential data predict the results of these experiments?

We have selected a number of integral experiments which appear to be useful for data testing. Where analyses have been reported with recent data sets (such as ENDF/B-IV), the results are summarized. In addition, we have done some exploratory studies intended to identify promising integral experiments.

The experiments are separated into two main categories: homogeneous U233 criticals and U233-Th (and U235-Th) lattice experiments. Lattice criticals are supplemented by the BNL U233-ThO₂ exponentials and by experiments to measure effective resonance integrals of thorium and ThO₂ rods. In addition, references to a number of other experiments are listed, some of which may turn out to be useful.

* This work was done as part of the program to qualify Monte Carlo methods and nuclear data sets for Advanced Water Breeder Application (AWBA) studies and Light Water Breeder Reactor (LWBR) core-follow analyses.

For the purpose of testing nuclear data, integral experiments must be clearly interpretable and should embody selected configurations of interest (materials, resonance shielding conditions, flux spectra, etc.). Aside from K_{eff} , it is desirable that spectrum-sensitive integral parameters (reaction rate ratios) be measured in lattice experiments because these provide much more specific tests of important cross sections. Experiments should be carefully performed, with attention to systematic errors and determination of material compositions, and they should be thoroughly documented.

Homogeneous criticals can provide useful information about fissile materials - from calculated K_{eff} versus composition, hydrogen/uranium ratio and leakage, for example. The next step--introduction of a fertile material and spatial resonance shielding--is best obtained with simple, uniform critical lattices. Such experiments have seldom been performed, due to limitations of time, materials, and money. Of necessity, recourse has been had to exponential lattice experiments, critical sub-lattices with drivers, and single-rod shielded resonance integral experiments, all of which can provide useful information. Unfortunately, these have nearly always been more difficult to interpret than simple criticals. Useful information about the adequacy of nuclear data sets can be obtained from the complex mock-up experiments provided that they can be accurately modeled. Even then, interpretation of such experiments in terms of specific nuclear data inadequacies is rendered difficult by their complexity.

II. Homogeneous Aqueous U233 Critical Experiments

A large number of U233-fueled, aqueous homogeneous experiments have been carried out to determine the critical parameters for a variety of configurations, densities, poisons, and reflectors. Many of the configurations, which were studied for criticality-safety purposes, are not particularly useful for testing nuclear data for U233 and H₂O. This section is limited to consideration of those experiments which bear most directly on the question of nuclear data. A compilation of such critical experiments, ordered by increasing H/U233 atom density ratio, is shown in Table 1 of Section A. Salient critical parameters and reflector characteristics are provided, as well as notes on possible errors in the measurements. Section B summarizes recent analyses of selected homogeneous U233 criticals with ENDF/B-IV data. Results for U235 criticals are also shown for comparison.

A. Survey of Experiments

(1) In support of fuel processing plant nuclear safety, a large number of critical experiments in simple geometry were performed at Oak Ridge in the mid-1950's (Ref. 1). These consisted of aqueous uranyl nitrate or uranyl fluoride solutions enriched to 98.7% U233. Except for two unreflected assemblies, the cylindrical or spherical aluminum vessels in these experiments had effectively infinite homogeneous reflectors of water or of paraffin. Properties of these critical experiments are shown in Table 1. Only unreflected and H₂O-reflected systems are included, since it is judged that a paraffin reflector unduly complicates interpretation of a system for basic data testing purposes.

In some cases, due to limitations of vessel size and solution inventory, the critical condition was determined by extrapolation of source-neutron multiplication curves. In Table 1, the critical parameters deduced by this procedure are enclosed in parentheses and include an error which is dependent on the length of the extrapolation.

These experiments are easily analyzable with Monte Carlo techniques. However, the critical parameters are subject to the usual errors in uranium analysis, solution densities, and in the calibration of vessel capacities. Each of these effects was estimated to be $\pm 0.5\%$. In addition, for some of the experiments using uranyl fluoride there was a bias resulting from the use of a corrosion-inhibiting coating which consisted of 30 w/o chlorine. An approximately 2% increase in the critical mass due to absorption of neutrons by the chlorine was later deduced through an auxiliary experiment. Those experiments for which this bias possibly exists are noted in Table 1, but this does not significantly affect calculated K_{eff} .

(2) In other early work at ORNL (Ref. 2), critical experiments using U233 and U235 were performed in identical geometry to compare $\bar{\eta}(\text{U233})$, the number of fission neutrons produced per thermal neutron absorbed in U233, with the better-known $\bar{\eta}(\text{U235})$.

Aqueous solutions of uranyl oxyfluoride containing uranium enriched to 98.7% and 90%, for U233 and U235 respectively, were made critical in water-reflected spherical containers having nominal diameters of 26.4 and 32.0 cm. Solution temperatures ranged from 20°C to 100°C. The specific critical parameters for the U233 solutions appear in Table 1.

The aluminum vessels were 1.27 mm-thick and can be easily included in any calculation. The volumes of the critical solutions were determined to the nearest 5 cm³, which implies that the critical radii can be determined to within 0.01 cm. Chemical analyses of the solutions were reliable to within $\pm 0.5\%$, and temperatures were maintained to within $\pm 0.5^\circ\text{C}$. For the larger sphere, it was stated that a systematic error existed which caused the critical masses and concentrations to be 2% high.

(3) In Ref. 3, further investigations of the criticality of U-233 solutions in simple geometries were reported. For one series of experiments, the criticality of water-reflected and unreflected spherical and cylindrical assemblies was determined. Again the solutions were uranyl nitrate containing 97.45% U233. These experiments (which are included in Table 1) have H/U233 ratios ranging from 73.0 to 581, which implies high neutron leakage fractions. For a given H/U-233 ratio, very often the critical dimensions of a sphere and several cylindrical volumes were given, along with information on the solution vessels. Aluminum cylinders had a 1.5 mm-thick wall and a 1.27 cm-thick base. Spheres were also of aluminum with a 1.22 mm-thick wall. For the unreflected cylinders, corrections have been made for the aluminum base, so that the results describe cylindrical volumes having aluminum on the lateral surface only. For the water-reflected assemblies, there was no top reflector. Since no corrections have been made for the base plate, it would have to be accounted for in analyses of these assemblies. Despite the complications, these experiments are suitable for Monte Carlo analysis, and they can also be analyzed (as one-dimensional systems) by deterministic transport theory.

(4) In a program designed to establish the thermal-energy value of η for U^{233} and U^{235} , a series of experiments were performed by Gwin and Magnuson (Ref. 4) to establish the critical conditions of unreflected homogeneous aqueous uranyl nitrate solutions of U^{233} and U^{235} . In addition to the critical dimensions of cylinders and spheres, spatial neutron flux distributions were measured to define the buckling of these systems; the kinetic behavior was investigated to determine experimental nonleakage probabilities and neutron age in these solutions; and epithermal neutron energy distributions were measured with bare and cadmium covered fission foils to determine integral corrections for epithermal fissions and absorptions.

Analyses of the U^{233} - H_2O critical experiments and detailed consideration of systematic corrections have been reported in Refs. 5 and 6, and recent calculations employing ENDF/B-IV nuclear data and Monte Carlo methods have been reported in Refs. 7 and 8. Some results of the latter calculations appear later in this paper.

In the cylinders, solution heights were adjusted to achieve criticality. These critical heights included a 0.53-in. correction for the effects of the stainless steel baseplate of the 5-ft. diameter cylinder. In Ref. 6, the room return correction to the measured K_{eff} was estimated to be 14×10^{-4} , approximately twice that obtained in Ref. 5. For the spherical assemblies, K_{eff} was determined from small positive periods. Hence, a correction for delayed neutrons was necessary to obtain the fundamental mode eigenvalue. (This correction is discussed in Ref. 5.) The reactivity worths of the aluminum containers for the 27.24-in. diameter and 48.04-in. diameter spheres were estimated experimentally

to be approximately 7×10^{-4} . The solution fill and drain tubes at the top and bottom of the spheres had measured worths of 0.7×10^{-4} and 0.2×10^{-4} for the 27.24-in. and 48.04-in.-diameter spheres, respectively. For the 5-foot-diameter cylinders, the worth of the solution fill tube was not estimated experimentally, but was considered to be $< 0.1 \times 10^{-4}$. Because of the smallness of the various corrections, these assemblies are easily analyzable by one-dimensional transport theory and by three-dimensional Monte Carlo techniques.

Table 1

Summary of Homogeneous U233 Critical Experiments of Potential Use for Data Testing

	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
Moderator/Fissile Atom Ratio	73.0	122	154.0	194	194
Geometry	Cylinder	Cylinder	Cylinder	Cylinders	Sphere
Critical Dimensions	H=13.36 cm R=25.4 cm	H=13.51,15.14,24.69 cm R=24.80,19.05,12.15 cm	H=24.0 + 0.05 cm R=12.75 cm	H=13.42,17.22,22.86 cm R=19.05,12.15,10.15 cm	R=11.17 cm
Reflector	None	None	None	Water	Water
Critical Mass	8.8 kg of U233	5.59,3.52,2.53 kg of U233	2.02 + 0.05 kg of U233	1.726,1.112,0.952 kg of U233	0.749 kg of U233
Reference	3	3	1	3	3
Notes and Comments	Aluminum cylinder had a 1.5 mm-thick wall and 1.27 cm-thick bottom. Results were corrected for the 1.27 cm-thick base.	See note for No. 1	No reported aluminum vessel correction to the data.	Aluminum cylinders had 1.5 mm-thick wall and 1.27 cm-thick bottom. Results were not corrected for bottom plate. There was no reflector on top of any cylinder.	Spheres was of aluminum with a 1.22 mm thickness.
I.D.*		I1	I2		

* I.D. is that used in Ref. 4 and Ref. 8.

(Continued)

Table 1 (Cont'd)

	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>
Moderator/Fissile Atom Ratio	195	195	253	253	253
Geometry	Cylinders	Sphere	Cylinders	Sphere	Cylinders
Critical Dimensions	H=14.07,16.35,28.52 R=25.4,19.05,12.15	R=14.579 cm	H=17.60,33.40 R=19.05,12.65	R=15.078 cm	H=19.67,20.02 cm R=12.65,10.15 cm
Reflector	None	None	None	None	Water
Critical Mass	3.645,2.378,1.832 kg of U233	1.657 kg of U233	1.998,1.666 kg of U233	1.423 kg of U233	1.111,0.951 kg of U233
Reference	3	1	3	3	3
Notes and Comments	See note for No. 1	See note for No. 5	See note for No. 1	See note for No. 5	See note for No. 4

I.D.

(Continued)

Table 1 (Cont'd)

	<u>11</u>	<u>12</u>	<u>13</u>	<u>14</u>	<u>15</u>	<u>16</u>
Moderator/Fissile Atom Ratio	273	349	349	378	381	390
Geometry	Sphere	Cylinders	Sphere	Sphere	Sphere	Sphere
Critical Dimensions	R=11.847 cm	H=19.35, 43.69 cm R=19.05, 12.65 cm	R=15.821 cm	R=13.25 cm	R=15.95 cm	R=13.240 cm
Reflector	Water	None	None	Water	None	Water
Critical Mass	0.645 kg of U233	1.608, 1.598 kg of U233	1.208 kg of U233	0.638 kg of U233	1.14 kg of U233	0.623 kg of U233
Reference	3	3	3	2	1	2
Notes and Comments	See note for No. 5	See note for No. 1	See note for No. 5	Aluminum vessel was 1.27 mm-thick. Critical volume was 9.737 liters at 96.5° C.	See note for No. 3. There was 40-cm ³ void above the critical solution.	Aluminum vessel was 1.27 mm-thick. Critical volume was 9.723 l at 83.2° C.

-10-

I.D.

L3

(Continued)

Table 1 (Cont'd)

	<u>17</u>	<u>18</u>	<u>19</u>	<u>20</u>	<u>21</u>	<u>22</u>	<u>23</u>
Moderator/Fissile Atom Ratio	390	400.5	405	414.6	418.3	419	426
Geometry	Sphere	Sphere	Sphere	Sphere	Sphere	Sphere	Sphere
Critical Dimensions	R=13.3 cm	R=13.23 cm	R=13.3 cm	R=13.22 cm	R=13.21 cm	R=13.3 cm	R=13.3 cm
Reflector	Water	Water	Water	Water	Water	Water	Water
Critical Mass	0.61 kg of U233	0.611 kg of U233	0.60 kg of U233	0.596 kg of U233	0.591 kg of U233	0.59 kg of U233	0.59 + 0.06 kg of U233
Reference	1	2	1	2	2	1	1
Notes and Comments	See note for No. 3. There was 380 cm ³ void above the critical solution.	Aluminum vessel was 1.27 mm thick. Critical volume was 9.704 l at 65.5° C.	See note for No. 3.	Aluminum vessel was 1.27 mm-thick. Critical volume was 9.675 l at 39.5° C.	Aluminum sphere was 1.27 mm-thick. Critical volume was 9.666 l at 32.0° C.	See note for No. 3. There was a 40-cm ³ void above the critical solution presumably at room temperature.	See note for No. 3.

Table 1 (Cont'd)

	<u>24</u>	<u>25</u>	<u>26</u>	<u>27</u>	<u>28</u>	<u>29</u>	<u>30</u>
Moderator/Fissile Atom Ratio	514	522	548	548	581	581	602.8
Geometry	Cylinder	Cylinder	Cylinders	Sphere	Cylinder	Sphere	Sphere
Critical Dimensions	H=25.5 + .1 cm R=12.75 cm	H=25.9 + 0.1 cm R=12.75 cm	H=19.6, 31.53 cm F=19.05, 12.65 cm	R=14.579 cm	H=26.37 cm R=19.05 cm	R=18.378 cm	R=15.97 cm
Reflector	Water	Water	Water	Water	None	None	Water
Critical Mass	0.64 + 0.03 of U233	0.65 + 0.5 kg of U233	1.043, 0.738 kg of U233	0.605 kg of U233	1.306 kg of U233	1.130 kg of U233	0.703 kg of U233
Reference	1	1	3	3	3	3	2
Notes and Comments	See note for No. 3	See note for No. 3	See note for No. 4	See note for No. 5	See note for No. 1	See note for No. 5	Aluminum vessel was 1.27 mm-thk. Critical volume was 17.074 l at 99.5°C. Concentrations may be 2% high due to systematic error.

I.D.

L4

(Continued)

Table 1 (Cont'd)

	<u>31</u>	<u>32</u>	<u>33</u>	<u>34</u>	<u>35</u>
Moderator/Fissile Atom Ratio	643.1	663.1	1324	1368	1417
Geometry	Sphere	Sphere	Sphere	Sphere	Sphere
Critical Dimensions	R=15.96 cm	R=15.96 cm	R=34.595 cm	R=34.595 cm	R=34.595 cm
Reflector	Water	Water	None	None	None
Critical Mass	0.659 kg of U233	0.673 kg of U233	3.358 kg of U233	3.250 kg of U233	3.212 kg of U233
Reference	2	2	4	4	4
Notes and Comments	See note for No. 30. Critical volume was 17.020 μ at 26.3°C.	See note for No. 30. Critical volume was 17.042 μ at 56.0°C.	Aluminum sphere was 0.33 cm- thick with a measured worth of 0.00072.	See note for No. 33.	See note for No. 33.
I.D.			9	8	7

(Continued)

Table 1 (Cont'd)

	<u>36</u>	<u>37</u>	<u>38</u>	<u>39</u>	<u>40</u>	<u>41</u>	<u>42</u>
Moderator/Fissile Atom Ratio	1470	1533	1819	1900	1986	1996	2106
Geometry	Sphere	Sphere	Cylinder	Cylinder	Sphere	Cylinder	Cylinder
Critical Dimensions	R=34.595 cm	R=34.595 cm	H=50.851 cm R=77.368 cm	H=60.579 cm R=77.368 cm	R=61.0108 cm	H=79.045 cm R=77.368 cm	H=140.157 cm R=77.368 cm
Reflector	None	None	None	None	None	None	None
Critical Mass	3.098 kg of U233	2.972 kg of U233	13.500 kg of U233	15.404 kg of U233	12.310 kg of U233	19.120 kg of U233	32.128 kg of U233
Reference	4	4	4	4	4	4	4
Notes and Comments	See note for No. 33.	See note for No. 33.	Critical height includes a correction for the 1.346 cm-thick aluminum bottom structure.	See note for No. 38.	Aluminum sphere was 0.762 cm-thick with an estimated worth of 0.00069.	See note for No. 38.	See note for No. 38.
I.D.	6	5	17	18	11	19	20

B. Results of Analysis of Homogeneous U233 Critical Experiments

In a study done at Oak Ridge (Ref. 7) to compare Hanson-Roach and ENDF/B-IV data sets, McNeany and Jenkins reported that ENDF/B-IV produced errors of several percent in calculated neutron multiplication factors for homogeneous U233 assemblies with low hydrogen/uranium atom ratios (H/U). It was concluded that the ENDF/B-IV data file was not satisfactory for calculating criticality of such assemblies. Conley (Ref. 9) had previously observed similar behavior with a somewhat different data set.

More recently, fourteen U233 criticals (including several of those considered in the Oak Ridge study) were analyzed at Bettis, and a set of seventeen U235 criticals was also analyzed for comparison (Ref. 8). The purpose was to examine in detail the ability of ENDF/B-IV data to predict criticality of such systems over a wide range of H/U, with the intent of drawing specific conclusions about the data set. These systems spanned a wide range of spectrum hardness (with as much as 50% epithermal fission) and of leakage (from 1%-50%). The critical eigenvalue and detailed reaction rates were calculated, and eigenvalue sensitivities to certain major data changes were determined.

1. Calculational Methods

For the calculations in Ref. 8, nuclear data were from ENDF/B-IV, multigroup processed with the ETOMX and ETYTX programs (Bettis versions of ETYG (Ref. 10) and ETYT (Ref. 11), respectively). Thermal scattering kernels were generated with FLAN2 (a Bettis version of FLANGEII (Ref. 12)) from the ENDF/B $S(\alpha, \beta)$ files based on the Haywood kernel.

The criticals were analyzed with the Monte Carlo program RCP01 (Ref. 13) and with P7MG, an extension of P3MG (Ref. 14), a one-dimensional, multigroup P_L program. RCP01 was used for the overall study of criticality versus H/U ratio. Cross section sensitivities were derived from P7MG, which was also used for the analysis of the Gwin-Magnuson criticals to infer constraints on the 2200 m/s parameters for U233 and U235. The agreement between RCP01 and P7MG was checked for five specific experiments and found to be very good.

2. Criticality Versus Hydrogen/Uranium Ratio

Eigenvalues for the U233 criticals (Table 2) show generally good agreement with the Oak Ridge calculations. This suggests that the use of different analysis programs and nuclear data processing procedures has little effect on the calculated results. For comparison, the U235 results are shown in Table 3. For both U233 and U235, the ENDF/B-IV-calculated eigenvalues become increasingly high relative to experiment as the H/U ratio is decreased from well thermalized systems.

Although there remain questions of interpretation about individual experiments, these are judged not to be significant for the overall pattern. Errors in calculated eigenvalue can stem from energy-dependent errors of η which are emphasized as the spectrum hardens (for example, uranium capture and fission resonance integrals, and thermal capture and fission cross sections). The hydrogen thermal σ_A affects the very thermalized assemblies, and $\bar{\nu}$ affects all assemblies equally. The other major source of eigenvalue error is under-prediction of leakage, which is emphasized in the smaller cores (see Figures 1 and 2). The most significant factors which could affect leakage are considered to be:

Table 2 - Results for Homogeneous U233 Critical
Experiments Calculated with ENDF/B-IV Data

Experiment	H/U233	Calculated		Calculated K_{eff}	
		Total Leakage	δ^{23}	Ref. δ^a	Ref. δ^b
L1	119	.5149	.4493	1.0116 $\pm$.0034	
L2	154	.5081	.3466	1.0226 $\pm$.0027	1.027 $\pm$.007
L3	381	.4617	.1526	1.0110 $\pm$.0035	1.026 $\pm$.006
L4	581	.4081	.1094	1.0111 $\pm$.0034	
9	1324	.1742	.0635	1.0007 $\pm$.0037	
		.1736	.0655	1.0029	
8	1369	.1743	.0633	1.0035	
7	1417	.1751	.0610	1.0035	
6	1470	.1759	.0588	1.0037	
5	1533	.1776	.0548	1.0017 $\pm$.0016	1.005 $\pm$.003
		.1769	.0563	1.0032	
17	1820	.1094	.0505	.9990	
18	1901	.0880	.0491	1.0002	
11	1986	.0683	.0463	.9974 $\pm$.0031	.994
		.0686	.0477	.9981	
19	1997	.0649	.0476	.9991	
20	2106	.0402	.0461	.9956 $\pm$.0008	
		.0398	.0461	.9967	

^aUncertainty is two standard deviations for RCP01 results. Eigenvalues without uncertainties are from P7MG.

^bUncertainty is one standard deviation.

δ^{23} is the ratio of epithermal/thermal U233 fission.

Table 3 - Results for Homogeneous U235 Critical Experiments
Calculated with ENDF/B-IV Data

Experiment	H/U235	Calculated Results (Ref. 8)		
		Total Leakage	λ^{25}	K_{eff}^a
L5	27.1	.4380	.8175	1.0085 $\pm$.0036
L6	44.3	.4501	.4722	1.0117 $\pm$.0030
L7	76.1	.4661	.1962	1.0098 $\pm$.0051
4	972	.1669	.0306	.9964
3	1033	.1678	.0288	.9949
L8	1112	.2333	.0251	1.0085 $\pm$.0037
2	1177	.1678	.0255	1.0007 $\pm$.0035
		.1699	.0252	.9980
1	1378	.1729	.0215	.9983
L9	1393	.1703	.0214	1.0035 $\pm$.0035
12	1604	.1197	.0194	.9985
13	1634	.1126	.0192	.9976
14	1821	.0679	.0178	.9980
10	1835	.0663	.0179	.9936 $\pm$.0031
		.0670	.0177	.9957
15	1905	.0475	.0173	.9980
21	1955	.0388	.0170	.9934
16	1981	.0329	.0169	.9939 $\pm$.0008
22	2004	.0269	.0167	.9938
23	2052	.0141	.0165	.9949 $\pm$.0003
		.0134	.0165	.9957

^aUncertainty is two standard deviations for RCP01 results. Eigenvalues without uncertainties are from P7MG.

Calculated Eigenvalue Versus Total Leakage Fraction for Homogeneous
U233 Criticals

FIGURE 1

Calculated Eigenvalue Versus Total Leakage Fraction
for Homogeneous U235 Criticals

FIGURE 2

- (1) multigroup averaging and limited P_0 treatment in RCP01
- (2) oxygen scattering cross section and angular pattern
(MeV range)
- (3) fission spectrum of U233 and of U235.

Since the trend in calculated eigenvalue error is nearly the same for U233 and for U235 systems, it is tempting to seek a common source of difficulty (e.g., items (1) and (2) above), but this has not proved fruitful.

For U233 criticals, evidence on the separate effects of leakage and flux spectrum hardening is afforded by the boron-poisoned Gwin-Magnuson experiments. Figure 3 shows K_{eff} for these systems plotted against δ^{23} , the ratio of epithermal-to-thermal U233 fissions. If the bias drift in Figure 3 for the unpoisoned cores is attributed to the spectrum hardening which accompanies increased leakage, then additional hardening obtained solely by boron poisoning should continue the trend, but this does not occur. Although the range of hardening is limited, this evidence suggests that the more likely source of difficulty is leakage itself.

It seems unlikely that the trend is due to energy-dependent errors of ϵ_a within the thermal range (which are emphasized as the flux spectrum hardens) since substantial errors would be required. Furthermore, calculational approximations (including data processing) are deemed to be relatively unimportant. This is supported by the good agreement obtained with the Oak Ridge Analysis (Ref. 7) and also by a comparison of RCP01-calculated leakage with results obtained by H. Henryson with the VIM Monte Carlo Program (Ref. 15). RCP01 employed multigroup averaged scattering cross sections with a P_3 angular treatment.

Eigenvalues of Gwin-Magnuson U^{233} Criticals as a
Function of Epithermal/Thermal U^{233} Fission Ratio

FIGURE 3

VIM employed a strictly pointwise representation of the data and a detailed elastic scattering angular treatment (20 equal $\cos \theta$ intervals). The comparison, using ENDF/B-IV data, was made for a small cylinder of U235-H₂O in vacuum, with a spatially flat fission source. Results for total leakage (10 MeV - 1.12 eV) were:

$$\begin{array}{rcl} \text{RCP01:} & 0.5285 \pm 0.0006 \\ \text{VIM:} & 0.5267 \pm 0.0007 \\ \hline \text{RCP01-VIM:} & 0.0018 \pm 0.0009 \end{array}$$

Agreement is very good, allowing for statistics (one standard deviation), and if anything RCP01 slightly overestimates leakage relative to VIM.

The effect of a leakage underprediction on criticality can be characterized quantitatively in terms of an error in the age to thermal energies of fission neutrons by means of a simple linear relationship between δK_{eff} and $\ln(1-L_F(E_c))$

$$\delta K_{\text{eff}} = a + b \ln(1-L_F(E_c)), \quad (\text{Eqn. 1})$$

where $L_F(E_c)$ is the calculated neutron leakage to energy E_c and $b =$

$$\frac{\delta L^2(E_c)}{L^2(E_c)}$$

is the relative error in the square of the slowing down length (age)

to energy E_c . The constant a is included since it is possible that all the calculated results could deviate from unity by a practically constant value due to error in the assumed number of neutrons produced per fission.

With the calculated leakage fractions to cutoff energies of $E_c = 5.53$ keV (L_{F1}) and $E_c = 0.625$ eV (L_{F2}), weighted least-squares analyses have been made using Eqn. 1 (Ref. 8). The results for the fitted constants a and b are summarized in Table 4. Within statistics the trend in calculated eigenvalues is independent of fuel type, and the age to thermal energies is underpredicted by 3-4%. For U235, this is consistent with a direct calculation of the U235 fission neutron age to indium-resonance in water: ENDF/B-IV data yield 25.49 cm^2 , 3.4% below the measured value of $26.4 \pm 0.3 \text{ cm}^2$ (Refs. 16, 17).

3. Sensitivities of Calculated Eigenvalues to Nuclear Data

Sensitivities of calculated eigenvalues to several nuclear data changes were determined for selected U233 and U235 assemblies. These changes include the hydrogen thermal absorption cross section, the uranium capture and fission resonance integrals, the uranium inelastic cross sections, the ENDF/B version of oxygen, and representations of the fission neutron spectra.

A reference eigenvalue was computed for each assembly with PTMG and ENDF/B-IV data. Then for each change in data a δK_{eff} was determined relative to this reference value. The sensitivities of K_{eff} to the various parameters are summarized in Tables 5-8. The following points are noteworthy:

- 1) The uranium capture and fission resonance integrals were varied as shown in Table 5. In each case this was achieved by a uniform percentage change of the smooth multigroup cross sections. There are significant sensitivities for the U235 systems but not for U233. The relative importance of these sensitivities can be seen from the resonance integrals shown in Table 6. The ENDF/B-IV U235 has a resonance alpha which is 7 (+ 4)% below the best integral value,

Table 4 - Results of Studies Using Eqn. 1

<u>Correlation</u>	<u>U233 Assemblies</u>	
	<u>a</u>	<u>b</u>
$\delta K_{\text{eff}} = a + b \ln(1 - L_{F1})$	-0.005 ± 0.002	-0.048 ± 0.011
$\delta K_{\text{eff}} = a + b \ln(1 - L_{F2})$	-0.006 ± 0.001	-0.035 ± 0.009

	<u>U235 Assemblies</u>	
	<u>a</u>	<u>b</u>
$\delta K_{\text{eff}} = a + b \ln(1 - L_{F1})$	-0.007 ± 0.002	-0.037 ± 0.017
$\delta K_{\text{eff}} = a + b \ln(1 - L_{F2})$	-0.007 ± 0.002	-0.030 ± 0.012

resulting in a fission integral which is too large and a low capture integral. Each of these could account for about 0.5% of the eigenvalue difference between small and large cores.

- 2) Reducing the hydrogen 2200 m/sec absorption cross section from 0.332 b to 0.330 b increases the reactivity of the thermal assemblies by 0.3% with negligible effect on the high-leakage assemblies.
- 3) A 20% reduction of the inelastic cross sections for both fuel types produced negligible changes of reactivity.
- 4) Sensitivities to the ENDF/B version of oxygen are worth as much as 0.6% in K_{eff} for the high-leakage assemblies and are negligible for the large assemblies. The successive versions of oxygen (ENDF/B-I to ENDF/B-IV) show an increase in backscattering, which reduces leakage.
- 5) The fission neutron spectrum significantly affects fast neutron leakage. Sensitivity of eigenvalue to mean energy (for a Maxwellian shape) and to the difference between Watt and Maxwellian shapes (for the same mean energy) is shown in Tables 7 and 8. This matter is discussed further in Section II.B.4 below.

Table 5 - Sensitivities of Eigenvalues to Uranium
Capture and Fission Resonance Integrals

		Change in Eigenvalue					
<u>U233</u>	H/U233:	119.4	154	381	581	1533	2106
Smooth capture integral increased 6.0 b		-.0021	-.0017	-.0007	-.0005	-.0002	-.0001
Smooth fission integral increased 13.0 b		+.0020	+.0016	+.0009	+.0008	+.0005	+.0005
<u>U235</u>	H/U235:	27.1	44.3	76.1	1112	2052	
Smooth capture integral increased 6.0 b		-.0077	-.0053	-.0028	-.0003	-.0002	
Smooth fission integral increased 10.0 b		+.0080	+.0053	+.0027	+.0005	+.0004	

Table 6 - Uranium Resonance Capture and Fission Integrals

		Resonance Integral Above 0.625 eV (barns)	
		<u>Fission</u>	<u>Capture</u>
U233	ENDF/B-IV (Mat. 1260)	733	132
U233	BNL-325, 3 rd ed. (Ref. 18)	735 $\pm$ 13	138 $\pm$ 6
U235	ENDF/B-IV (Mat. 1261)	268	136
U235	BNL-325, 3 rd ed.	259 $\pm$ 5	142 $\pm$ 6

Table 7 - Sensitivities of U233 Critical Assemblies to Changes in Nuclear Data

<u>Data Component</u>	<u>Change in Data Component</u>	<u>Change in Eigenvalue, δK_{eff}</u>			
		<u>H/U233=154^a</u>	<u>381</u>	<u>1533</u>	<u>1986</u>
<u>Hydrogen Cross Sections</u>	Reduced thermal absorption cross section from 0.332 b to 0.330 b	+0.0004	+0.0009	+0.0026	+0.0030
<u>Uranium-233 Scattering Cross Sections</u>	Reduced the inelastic cross section by 20%.	-0.0004	-0.0001	0.0000	0.0000
	Inelastic cross sections reduced by 20% and the elastic cross section increased by an equivalent amount to maintain the same total cross section.	-0.0003	-0.0001	0.0000	0.0000
<u>Oxygen Cross Sections</u>	ENDF/B-IV $\rightarrow$ ENDF/B-III	-0.0041	-0.0031	-0.0008	-0.0001
	ENDF/B-IV $\rightarrow$ ENDF/B-I	-0.0062	-0.0044	-0.0012	-0.0002
<u>Fission Spectrum^b</u>	Maxwellian spectrum $\bar{E}=2.0$ MeV	+0.0036	+0.0029	+0.0009	+0.0003
	Maxwellian spectrum $\bar{E}=2.1$ MeV	-0.0134	-0.0109	-0.0049	-0.0024
	Watt spectrum ($\bar{E}=2.0$ MeV)	-0.0001	+0.0001	+0.0005	+0.0004

^aExperiment ID's are those of Ref. 8 (see Table 1).

^bAll δK_{eff} are relative to the ENDF/B-IV fission neutron spectrum for U233.

Table 8 - Sensitivities of U235 Critical Assemblies to Changes in Nuclear Data

<u>Data Component</u>	<u>Change in Data Component</u>	<u>Change in Eigenvalue, δK_{eff}</u>				
		<u>H/U235-27.1(L5)^a</u>	<u>44.3(L6)</u>	<u>1112(L8)</u>	<u>1177(2)</u>	<u>1835(10)</u>
<u>Hydrogen Cross Sections</u>	Reduced thermal absorption cross section from 0.332 b to 0.330 b	+0.0001	+0.0001	+0.0021	+0.0020	+0.0028
<u>Uranium-235 Scattering Cross Sections</u>	Reduced the inelastic cross section by 20%.	-0.0016	-0.0011	0.0000	0.0000	0.0000
	Inelastic cross section reduced by 20% and the elastic cross section increased by an equivalent amount.	-0.0009	-0.0006	0.0000	0.0000	0.0000
<u>Oxygen Cross Sections</u>	ENDF/B-IV $\rightarrow$ ENDF/E-III	-0.0033	-0.0034	-0.0013	-0.0008	-0.0001
	ENDF/B-IV $\rightarrow$ ENDF/E-I	-0.0049	-0.0050	-0.0018	-0.0012	-0.0002
<u>Fission Spectrum^b</u>	ENDF/B-III U235 fission neutron spectrum	+0.0049	+0.0051	+0.0027	+0.0020	+0.0009
	Maxwellian spectrum $\bar{E}=2.0$ MeV	-0.0022	-0.0023	-0.0012	-0.0009	-0.0004
	Maxwellian spectrum $\bar{E}=2.1$ MeV	-0.0157	-0.0165	-0.0087	-0.0067	-0.0031
	Watt spectrum ($\bar{E}=2.0$ MeV)	-0.0053	-0.0054	-0.0020	-0.0015	-0.0004

^aExperiment ID's are those of Ref. 8.

^bAll δK_{eff} are relative to the ENDF/B-IV fission neutron spectrum for U235.

4. Fission Spectra

Four recent differential experiments show that the prompt neutron spectrum for low-energy U235 fission is significantly harder than that of ENDF/B-IV, a Maxwellian with $\bar{E} = 1.985$ MeV. A description of the experiments and an analysis of them by J. M. Adams appeared in the Proceedings of the Specialists Meeting on Inelastic Scattering and Fission Neutron Spectra, which was held at Harwell in April 1975. Although it did not provide an exact fit to the data, the Watt model fit quite well and was clearly better than a Maxwellian. The Adams best-fit Watt spectrum had a prompt mean energy of 2.016 MeV.

A recent simultaneous least squares adjustment of data for U233 and U235 systems has provided best-fit fission spectra for both nuclides (Ref. 19). Among the data included in the analysis were thermal parameters (including Cf252 $\bar{\nu}$, thermal $\bar{\nu}$ ratios, η , σ_f , σ_c) for U233 and U235, K2 values* from the Gwin-Magnuson criticals, eigenvalues of small homogeneous criticals, neutron age, capture and fission resonance integral and differential prompt fission spectra. For U235, a good fit was obtained with a Watt shape ($\bar{E}_p = 2.027 \pm .004$ MeV). This result is based on four differential experiments, three of which were used by Adams. A more complex prompt spectrum model (two fully accelerated fragments plus a stationary source) produced an equally good fit to the differential data and a substantially higher mean energy of $2.073 \pm .005$ MeV. This is denoted the Standard model. For U233, based on the differential experiment of L. Green et al. (Ref. 20), a good fit was obtained for a Standard model with $\bar{E}_p = 2.082$ MeV. The Watt and Maxwellian models fail to provide adequate representation of these data.

*K2 is described in Section 5 below.

The effects of these alternative spectra on K_{eff} of the homogeneous criticals are shown in Figures 4 and 5. For U233 systems (Figure 4), the Standard model removes the trend in calculated K_{eff} versus leakage observed with straight ENDF/B-IV data (Figure 1).

For U235 systems (Figure 5), where the situation is complicated by the resonance absorption sensitivity, three combinations are illustrated. In part A of this figure, the effect of a 10b reduction of the ENDF/B-IV U235 fission resonance integral is shown by solid symbols for the three high-leakage criticals. (For the others, the effect is negligible.) Part B shows the effect of the reduced fission integral plus use of an equilibrium spectrum based on a Watt prompt spectrum with $\bar{E}_p = 2.027$ MeV. This combination removes most of the eigenvalue variation with leakage. Finally, part C shows the effect of an equilibrium spectrum based on the much harder Standard model. Both the Watt and Standard models can, with suitable choice of U235 resonance fission and capture integrals, eliminate the trend of calculated K_{eff} with leakage. In either case, there remains a defect of K_{eff} which is most readily removed by an $\sim 0.6\%$ increase of $\bar{v}$.

With ENDF/B-IV data, the calculated age to indium resonance (1.47 eV) for a U235 fission source in H_2O is well below experiment. The equilibrium spectrum based on the Watt prompt spectrum with $\bar{E}_p = 2.027$ MeV provides slightly better agreement. The harder Standard model spectrum, with $\bar{E}_p = 2.073$ MeV, leads to very good agreement as shown in Table 9.

Calculated Eigenvalue Versus Total Leakage Fraction for Homogeneous U^{233} Criticals,
Obtained with the Standard Fission Spectrum Model

FIGURE 4

Effects of Alternative Fission Spectra on Eigenvalues of U235 Criticals (Solid symbols include a 10 b reduction of the U234 fission resonance integral. See text Section III.B.4)

FIGURE 5

In another paper at the 1975 Harwell Specialists Meeting, A. Fabry reviewed current evidence on fission spectra derived from integral experiments. The spectrum mean energy derived from dosimetry measurements is now 2.00 - 2.05 MeV, and this is consistent with the mean energy implied by the fission age-to-indium resonance in H_2O . He also emphasized the importance of integral spectrum-average cross sections, particularly $\bar{\sigma}_f^{U235}$, $\bar{\sigma}_f^{U238}$ and their ratio.

Table 10 shows integral experimental values recommended by Fabry along with calculated values. The ENDF/B-IV calculation produces a low value of $\bar{\sigma}_f^{U238}$. Some improvement is obtained with the Watt shape, and very good agreement is shown by the Standard spectrum. It should be mentioned that $\bar{\sigma}_f^{U238}$ was not used in the fit so that this is an independent test of the spectrum.

Table 9 - Flux Age to 1.47 eV in H₂O (U235 Fission Source)

	<u>Age (cm²)</u>
Experiment (Ref. 16, 17)	26.4 ± 0.3
ENDF/B-IV Calculation	25.49
ENDF/B-IV (except Watt Model, E _p = 2.027 MeV)	25.88
ENDF/B-IV (except Std. Model, E _p = 2.073 MeV)	26.33

Table 10 - Fission Spectrum Averaged Cross Sections

	$\bar{\sigma}_f^{U235}$ (mb)	$\bar{\sigma}_f^{U238}$ (mb)	$\bar{\sigma}_f^{U235} / \bar{\sigma}_f^{U238}$
Integral Experiment (A. Fabry)	1220 ± 31	313.5 ± 10	$3.89 \pm .08$
ENDF/B-IV Calculation	1244	294	4.23
ENDF/B-IV (except Watt Spec., $E_p = 2.027$ MeV)	1242	304	4.08
ENDF/B-IV (except Std. Spec., $E_p = 2.073$ MeV)	1235	312	3.96

5. Thermal Criticality Parameter Constraint

A constraint on the thermal criticality parameters of U233 has been derived by Gwin (Ref. 21) from the Gwin-Magnuson homogeneous criticals near the zero-leakage limit. This is $K_2 = (\bar{\eta}_U - 1) \frac{\bar{\sigma}_U}{\bar{\sigma}_H} \frac{a}{a}$, relating the thermal-average eta for U233 and the U233 and hydrogen thermal absorption cross sections. Results are shown in Table 11 along with those for U235, for comparison. Also shown are K_2 values obtained in Ref. 8 and corresponding "g" factors. The uncertainty on K_2 is meant to account for all sources of error in the analysis (mainly fissile inventory, calculated epithermal multiplication and absorption, and calculated leakage).

It is essential to determine K_2 from well-thermalized systems near the zero-leakage limit ($FP=1.0$) since inferences about thermal eta, derived from analysis of critical systems, should not be influenced by errors of calculated criticality stemming from inability to calculate leakage or from uncertainties of calculated epithermal absorption rates. This is illustrated in Figure 6 which shows K_2 values calculated for the Gwin-Magnuson U233 Criticals with ENDF/B-IV and with artificially enhanced fast leakage*. Errors in calculated leakage can be caused by high-energy data inadequacies (as is evident for ENDF/B-IV) and calculational approximations, and by systematic errors in interpreting the experiments.

With the "g" factors of Table 11, ENDF/B-IV gives K_2 values of 2220 and 2137 for U233 and U235, respectively. These are 0.9% and 0.8% below the corresponding experimental values.

* In Figure 6, K_2 is plotted versus the quantity FP . F can be viewed as the product of fast multiplication factor and resonance escape probability (ϵ_p), and P is the nonleakage probability. At $FP = 1.0$, fast multiplication just balances the small fast absorption and leakage. The augmented-leakage points were obtained by artificially hardening the fission spectrum.

Table 11 - Thermal Criticality Parameter Constraint
K2 Derived from the Gwin-Magnuson Criticals

		<u>U233</u>	<u>U235</u>
(Ref. 21)	K2	2234	2148
(Ref. 8)	K2	2240 $\pm$ 11	2155 $\pm$ 11
	g_f^u	.99752	.97065
	g_a^u	1.00045	.97326

Note: $K2 \equiv (\bar{\eta}_u - 1) \frac{\bar{\sigma}_a^u}{\sigma_a^u}$

where $\bar{\sigma}_f^u = g_f^u \sigma_f^u$ (.0253 eV)

$\sigma_a^u = g_a^u \sigma_a^u$ (.0253 eV)

Thermal Criticality Parameter K2 Versus FP, Calculated with ENDF/B-IV and with Augmented Leakage for U233 Criticals

FIGURE 6

6. Summary of Results for Homogeneous Criticals

Homogeneous aqueous U233 and U235 critical assemblies, spanning a wide range of spectrum hardness (from 1% - 50% epithermal fission) and corresponding leakage (from 1% - 45%), were analyzed with ENDF/B-IV data and Monte Carlo methods. At the zero-leakage limit, these criticals determine an experimental constraint

$$K_2 = (\bar{\tau}_U - 1) \frac{\frac{\sigma_U}{\sigma_H} a}{\sigma_a}$$

on the thermal criticality parameters ($K_2 = 2240 \pm 11$ and 2155 ± 11 for U233 and U235, respectively).

For U233 systems, calculated K_{eff} ranged from 0.996 - 1.014, and a similar effect was observed for U235. Errors of interpretation and of calculational methods are not very significant. The spread in K_{eff} values is attributed primarily to deficiencies of ENDF/B-IV data, which can be of two types:

- (1) energy-dependent errors of $\bar{\tau}$
- (2) leakage underprediction.

Based on sensitivity studies and consideration of data uncertainties, the following items appear to be relatively unimportant: hydrogen and oxygen σ_s , and the fission and capture resonance integrals of U233. For U235, the fission and capture resonance integrals can each be worth approximately 0.5% at 45% leakage. For the well-thermalized systems, uncertainties of fissile thermal σ_f and σ_c are worth a few tenths percent, and the hydrogen σ_A is of similar importance. Normalization of $\bar{v}$ affects all systems equally.

The primary factor is the fission spectrum, and this is significantly model dependent. Recent differential experiments imply a spectrum that is harder than ENDF/B-IV and not Maxwellian. Although the Watt shape is better, a more complex model (such as the Standard model) is required to fit the differential data. For U233, the Standard Model with $\bar{E}_p = 2.082$ MeV (all other data from ENDF/B-IV) eliminates the K_{eff} trend with leakage.

For U235, the Standard model with $\bar{E}_p = 2.073$ MeV provides a good fit and eliminates the K_{eff} trend (with appropriate values of capture and fission integrals). A Watt shape with $\bar{E}_p = 2.027$ MeV provides as good a fit and also eliminates the K_{eff} trend (again with appropriate values of capture and fission integrals). In either case, further measures such as an $\sim 0.5\%$ increase of $\bar{v}$ are required to completely eliminate the K_{eff} bias. The Standard model also brings the calculated U235 fission neutron age in H_2O into good agreement with experiment, while the softer spectra do not. (For ENDF/B-IV, the age is $3.4 \pm 1.1\%$ below experiment.) The Standard model also gives excellent agreement with integral measurements of $\bar{\sigma}_f^{U238}$ and $\bar{\sigma}_f^{U235} / \bar{\sigma}_f^{U238}$ (Table 10).

III. Thorium-U233 and Thorium-U235 Lattice Experiments (H_2O and D_2O Moderated)

A. Survey of Experiments

This section summarizes a number of lattice experiments deemed potentially useful for data testing, as shown in Table 12. Very few such experiments are available with U233 fuel. Aside from the critical experiments in direct support of LWBR (the SB and BMU series and the detailed cell), there are only the BNL U233O₂-ThO₂ exponentials and the very hard spectrum Bettis ETA-II critical (which used the BNL fuel rods).

There are several U235-fueled ThO₂ criticals available. Among these are experiments run in the late 1950's at B&W and at ANL: the H₂O-moderated TUPE and CETR lattices, the H₂O-D₂O SSCR experiments, and the D₂O-moderated THUD lattices. One series of experiments, with 19-element cluster rods, was run at Chalk River in the center of a ZED-2 lattice of CANDU-type rods.

As a supplement to the lattice critical experiments, measurements of shielded resonance capture in single rods are useful for testing thorium resonance capture data. These and the BNL exponentials are the only departures taken from the restriction to critical experiments.

These experiments have been selected as the most promising candidates for thermal reactor data testing. Some of them - the TUPE's, three THUD's, the BNL exponentials, one BMU and the detailed cell - have been analyzed with Monte Carlo methods and ENDF/B-IV data. These studies are described in Section B. Aside from K_{eff} , the following integral parameters have been computed where there are measured values available:

- ρ^{02} = ratio of epithermal/thermal capture in thorium.
- $\delta^{23}(\delta^{25})$ = ratio of epithermal/thermal fission in U233 (or U235).
- δ^{02} = ratio of fission in thorium to fission in U233 (or U235).
- CR* = ratio of capture in thorium to fission in U233 (or U235) -
the modified conversion ratio.

Table 12

Summary of Th-U233 and Th-U235 Exponential and
Critical Experiments of Potential Use for Data Testing

Experiment	Effective resonance capture integrals	TUPE criticals	CETR criticals
Laboratory	Bettis, B&W, A. B. Atomenergi	B&W	B&W
References	22, 23, 24, 25, 26 (1960-1966)	27 (1960)	28 (1960)
Fuel	Thorium and ThO ₂ rods	U235O ₂ -ThO ₂ rods, aluminum clad, .660 cm diam. x 152 cm Th/U235 = 15/1 and 25/1	U235O ₂ -ThO ₂ rods, steel clad .660 cm diam. x 122 cm Th/U235 = 15/1 and 25/1
Moderator	--	H ₂ O, partial height	H ₂ O
Temperature	300° K, 1000° K, 1200° K	Ambient	Ambient
Lattices	(additional measurements of temperature dependence are cited in Ref. 21).	Uniform square array of rods, cylindrical boundary. Five lattices with mod./fuel volume ratio from 1.4 to 5.8.	Uniform square array of rods, cylindrical boundary. Seven lattices with mod./fuel volume ratio from 1.3 to 9.5
Parameters Measured	Activation measurements for single rods with $\sqrt{S/M}$ ranging from 0.30-0.90 cm gm ⁻² relative to the dilute resonance integral.	Critical size, material B ² , radial and axial flux shapes, reflector savings, thermal flux depression, cadmium ratio of Th232 capture and U235 fission	Critical size, material B ² , radial and axial flux shapes, reflector savings, thermal flux depression, cadmium ratio of Th232 capture and U235 fission, differential H ₂ O worth, differential boron poison worth.
Comments	--	Good for criticality. Analysis of U235 and Th232 cadmium ratios requires detailed study of experimental effects.	Good for criticality. Analysis of cadmium ratios requires detailed study of experimental effects. Fuel same as Ref. 26 with different clad. (Continued)

Table 12 (Cont'd)

Experiment	SSCR criticals	THUD criticals	ZED-2 19-element-rod lattices
Laboratory	D2W	ANL	CRNL
References	29 (1961)	30 (1961)	31, 32 (1968)
Fuel	U235O ₂ -ThO ₂ rods, .660 cm diam. x 152 cm, aluminum clad, Th/U235 atom ratio 15/1	U235O ₂ -ThO ₂ rods, .660 cm diam. x 152 cm, aluminum clad, Th/U235 atom ratio 15/1 and 25/1	19-element clusters of ThO ₂ containing 1.5 w/o UO ₂ (93 atom % U235)
Moderator	D ₂ O-H ₂ O (60.4 mole % D ₂ O)	D ₂ O (99.4%-97.0% purity), partial moderator height	D ₂ C (coolant: air, D ₂ O, H ₂ O, HB ⁴ O)
Temperature	Ambient	Ambient	Ambient
Lattices	Uniform square array of rods, cylindrical boundary, mod./fuel volume ratio 1.4.	Uniform triangular array of rods, cylindrical boundary. 45 lattices on four pitches, from 1.9 cm to 5.7 cm.	Triangular array. Seven cluster-rods (center and first hex) in lattice of 19-element UO ₂ CANDU-type rods. Pitches of 22 cm and 28 cm (B ² also at pitches of 24 cm and 32 cm).
Parameters Measured	Critical size, reflector savings, radial and axial flux shapes, material B ² , thermal flux depression, cadmium ratios of Th232 capture and U235 fission.	Critical size, reflector savings, differential D ₂ O worth, cadmium ratio of Th232 capture, thermal flux depression, fast fission factor.	Material B ² ; neutron fine structure; relative U233, U235 and Pu239 fission ratios; δ^{02} , CR*.
Comments	Good for criticality. Analysis of cadmium ratios requires detailed study of experimental effects. Fuel rods same as Ref. 27.	D ₂ O purity determination in question. Good for criticality. Analysis of Th232 cadmium ratio and fast fission factor requires detailed study of experimental effects. Fuel rods same as Ref. 27.	-- (Continued)

Table 12 (Cont'd)

Experiment	BNL exponentials	ETA-I Critical	ETA-II Critical
Laboratory	BNL	Bettis	Bettis
References	33, 34 (1966, 1970)	35, 36 (1974)	35 (1974)
Fuel	U2330 ₂ -ThO ₂ rods, 1.09 cm diam x 107 cm, Zircaloy-2 clad, Th/U233 atom ratio 32.5/1	U2350 ₂ -ThO ₂ rods, .660 cm diam x 152 cm, aluminum clad, Th/U235 atom ratio 15/1	U2330 ₂ -ThO ₂ rods, 1.09 cm diam x 107 cm, Zircaloy-2 clad, Th/U233 atom ratio 32.5/1
Moderator	H ₂ O and D ₂ O	D ₂ O (99% purity) fully reflected	D ₂ O (98.4% purity)
Temperature	Ambient	Ambient	Ambient
Lattices	Uniform triangular array of rods, cylindrical boundary. 21 H ₂ O-moderated lattices with mod./fuel volume ratio from 1.0 to 9.3 and varying concentrations of H ₃ BO ₃ . Eight D ₂ O-moderated lattices with mod./fuel volume ratio from 2.2 to 11.5	Inner: 2304 rods in square array of 3x3 rod clusters; cylindrical boundary 61 cm diam. Mod./fuel vol. ratio = 2.02 Driver: H ₂ O moderated lattice of TRX slightly-enriched UO ₂ rods.	Inner: 864 rods in square array of 2x2 rod clusters, cylindrical boundary, ~56 cm diam., mod./fuel vol. ratio = 1.70. Driver: H ₂ O moderated lattice of TRX slightly-enriched UO ₂ rods.
Parameters Measured	Material B ² , reflector savings, thermal flux depression, ρ^{02} , δ^{02} .	δ^{02} , CR*, ρ^{02} , δ^{25} ; Th232 capture Doppler	δ^{02} , CR*, ρ^{02} , δ^{23}
Comments	--	Central parameters in asymptotic spectrum; K _{eff} depends significantly on the driver and requires full core analysis since coupling to the driver not measured. Very hard spectrum (ρ^{02} =10.5). Fuel rods same as B&W 15/1 (Ref. 27).	Central parameters in asymptotic spectrum; K _{eff} depends significantly on the driver and requires full core analysis since coupling to the driver not measured. Very hard spectrum (ρ^{02} =8.89). Fuel rods same as BNL (Ref. 33). (Continued)

Table 12 (Cont'd)

Experiment	LWBR SB criticals	LWBR BMU criticals	LWBR detailed cell
Laboratory	Bettis	Bettis	Bettis
References	37 (1967)	38 (1975)	39 (1974)
Fuel	Zircaloy-2 clad, 38 cm-long rods. Seed: 0.54 cm diam, 26 w/o U2330 ₂ -ZrO ₂ and 26 w/o U2350 ₂ -ZrO ₂ . Blanket: 1.25 cm diam, ThO ₂ and 1 w/o U2330 ₂ -ThO ₂ .	Axially zoned U2330 ₂ -ThO ₂ and ThO ₂ in Zircaloy-4 clad. Seed: 0.52 cm diam rods of four types. Blanket: 1.4 cm diam rods of four types.	Twelve types of U2330 ₂ -ThO ₂ and ThO ₂ seed and blanket rods in Zircaloy-4.
Moderator	H ₂ O	H ₂ O	H ₂ O
Temperature	Ambient (two of the wet-blanket assemblies were run also at 480°F)	Ambient (except BMU-1C at 410°F)	Ambient (except Core III b at 477°F)
Lattices	Small seed-blanket assemblies. Five wet-blanket lattices were square arrays with rectangular boundaries. Three dry-blanket lattices were tri- angular arrays with hexagonal boundaries. <div style="text-align: right; margin-right: 50px;"><u>Mod./Fuel Volume Ratio</u></div> <div style="display: flex; justify-content: flex-end;"> <div style="text-align: left; margin-right: 20px;"> U233 Seed U235 Seed Dry Blanket Wet Blanket </div> <div style="text-align: right;"> 27, 78 26, 76 0.21 1.97 </div> </div>	Multiply-zoned seed-blanket assemblies. Three BMU-1 lattices were triangular arrays with hexagonal boundary (single module). Five BMU-2 lattices were 1-, 3-, and 4-module assemblies.	Triangular array with hexagonal, 270-cm- high multiply-zoned seed and blanket. Seven configurations with movable seed at different heights.
Parameters Measured	Critical size. Radial and axial flux shapes (U235 fiss, U233 fiss, Th cap, Dy) ρ^{02} , δ^{25} , δ^{23} , thermal disadvantage factors, fast advantage factors in selected locations.	Critical size; radial and axial flux shapes; region power fractions; ρ^{02} and δ^{23} in selected locations; central worths of structural materials; Cf252 neutron importance.	Critical size; region power fractions; axial and radial flux shapes; δ^{23} , ρ^{02} , δ^{02} , at selected lo- cations.
Comments	Good for criticality. Interpretation of parameters requires detailed study of experimental effects. Lack of asymptotic region and small size complicate analysis.	Good for criticality. In- terpretation of integral parameters difficult due to geometrical complexity.	Good for criticality. Interpretation of integral parameters difficult due to geometrical complexity.

B. Results of Analysis of Th-U233 and Th-U235 Lattice Integral Experiments

This section surveys results of current analyses of Th-U233 and Th-U235 lattice experiments (mostly with ENDF/B-IV data) supplemented by previously published results from other analyses with closely related data sets.

1. Resonance Integrals for Thorium Capture

Measurements of the dilute resonance capture integral of thorium have usually been made by the cadmium ratio method relative to the dilute resonance integral of gold. In this method, resonance and thermal activation ratios are compared by irradiating dilute gold and thorium detector foils, bare and cadmium-covered, in a pseudo 1/E flux spectrum. Normalization depends on the thorium and gold thermal capture cross sections and the gold dilute resonance integral:

$$I_{\text{Th232}} = \frac{(A_{\text{res}}/A_{\text{ther}})_{\text{Th232}}}{(A_{\text{res}}/A_{\text{ther}})_{\text{Au}}} \times \frac{\sigma_{\text{Th232}}}{\sigma_{\text{Au}}} I_{\text{Au}} \quad (\text{Eqn. 2})$$

The measured resonance and thermal specific activations (A_{res} and A_{ther}) of Eqn. 2 must be suitably corrected for cadmium-cutoff energy, detector foil shielding and deviation of the neutron flux spectrum from 1/E behavior.

Grenèche (Ref. 40) has recently reviewed the available integral experiments. On the basis of fourteen measurements he obtains a dilute resonance integral

$$I_{\text{Th232}} = 85.8 \pm 0.9b \text{ (experimental)} \\ \pm 1.6b \text{ (normalization).}$$

This represents all capture above 0.5 eV and is normalized as follows:

$$\tau_{\text{Au}} = 98.8 \pm 0.3b$$

$$\tau_{\text{Th232}} = 7.40 \pm 0.08b$$

$$I_{\text{Au}} = 1560 \pm 22b$$

The BNL 325 (Ref. 18) value of the dilute Th232 resonance capture integral is $85 \pm 3b$.

Complementing lattice experiments as a test of shielded resonance capture are measurements of effective resonance integrals for thorium metal and ThO_2 rods of various diameters. The experimental methods, problems of interpretation and results have been summarized by Hellstrand (Ref. 22).

For thorium, the experiments were done by activation techniques in which a cadmium-covered rod was irradiated in a pseudo-1/E flux spectrum. Thorium captures were sampled by counting the capture activity of a ground-up portion of the ThO_2 rod or (for a metal rod) of a thorium metal detector foil. Normalization was explicitly to the dilute resonance integral of thorium:

$$I' (S/M, T) = \left(\frac{A'_{\text{res}} (S/M, T)}{A_{\text{res}}} \right)_{\text{Th232}} \times I_{\text{Th232}} \quad (\text{Eqn. 3})$$

where I' and A'_{res} are the effective resonance integral and specific thorium resonance capture activation for a rod of surface/mass ratio $S/M \text{ cm}^2/\text{gm}$ at temperature T . In some experiments, normalization was directly to the gold resonance integral via Eqn. 2.

In comparison to lattice measurements, the thorium activation counting techniques are similar, but there is the additional complication of determining the actual flux spectrum impinging on the rod. In this sense an effective resonance integral measurement is more difficult to interpret than a lattice experiment, although in principle it is more specific (i.e., not subject to interference from competing reactions).

On the basis of two experiments (Ref. 23, 25) for metal and three for oxide rods (Ref. 24, 25 and 26), Hellstrand (Ref. 22) recommends the following room-temperature effective resonance integrals:

$$\begin{array}{ll} \text{Thorium metal} & I' = 5.05 + 15.5 \sqrt{S/M} \, b \\ \text{ThO}_2 & I' = 5.85 + 16.2 \sqrt{S/M} \, b \end{array} \quad (\text{Eqn. 4})$$

Here, S/M is the surface-to-mass ratio (cm^2/gm) of metal or of oxide. These expressions apply over the range $0.30 < \sqrt{S/M} < 0.90$, with an assigned uncertainty of $\pm 4.5\%$, and include all capture above 0.5 eV.

A connection with lattice measurements of thorium resonance capture can be established via the LWBR thorium capture data evaluation (Ref. 41). The LWBR data set produced calculated effective resonance integrals for ThO_2 rods which are well represented by the expression:

$$I' = 5.74 + 15.54 \sqrt{S/M} \, b \, (300^\circ\text{K})$$

The calculated effective resonance integral is approximately 3% below the Hellstrand recommendation. Analysis of several lattices with the LWBR thorium data have shown (Ref. 42) that it produces approximately 3% more resonance capture than does ENDF/B-IV. Hence, it is concluded that ENDF/B-IV thorium produces effective resonance capture integrals approximately 6% below the Hellstrand recommendation.

2. TUPE Lattices (Th-U235-H₂O)

The TUPE (Thorium-Uranium-Physics-Experiment) criticals were operated at Babcock and Wilcox in the late 1950's (Ref. 27). There were five H₂O-moderated simple lattices constructed from U235O₂-ThO₂ rods having Th/U235 atom ratios of 15/1 and 25/1. The lattices were square arrays at four pitches, with moderator/fuel volume ratios ranging from 1.38 to 5.79.

The five uniform lattices have been analyzed with ENDF/B-IV data and the RCP01 Monte Carlo program. The cores were described explicitly in three dimensions, including the lattice plates, core support structure and tank, with a black absorbing boundary placed outside the tank.

Table 13 and Figure 7 present eigenvalues obtained in the full-core analyses with ENDF/B-IV data. These show relatively good agreement with experiment, although there is a distinct downward trend in the tighter lattices.

Material bucklings, inferred from careful measurements of radial and axial activation shapes, were reported in Ref. 27. For comparison with the full-core Monte Carlo results, eigenvalues have been calculated from a zero-leakage cell Monte Carlo, with leakage correction obtained from MUFT7 with the measured material buckling. Fast advantage factors and thermal disadvantage factors were obtained from the RCP01 cell region fluxes for the fast (10 MeV - 820 KeV) and thermal (0 - 0.625 eV) groups. The MUFT7 calculation used a resonance absorption treatment based on integral transport theory, with the same detailed cross section profile as used in RCP01. This produced reaction rates in good agreement with the cell RCP01 over the range 0.625 eV - 5530 eV in which it was applied. An alternative calculation was done with the

Table 13 - Calculated Criticality of TUPE Lattices
(Full Core RCP01)

<u>Lattice Ident. (Ref. 27)</u>	<u>Mod/Fuel Volume Ratio</u>	<u>Number of Rods</u>	<u>Moderator Height (cm)</u>	<u>Calculated Net Leakage</u>	<u>Calculated $K_{eff} \pm \sigma^*$</u>
15A	1.379	1108	127.21	0.313	$1.0002 \pm .0014$
15B	1.642	880	133.44	0.325	$.9947 \pm .0013$
15D	2.945	514	135.05	0.317	$1.0056 \pm .0014$
25D	3.636	1176	140.50	0.206	$1.0075 \pm .0013$
25 $\sqrt{2}$ B	5.794	1146	135.27	0.165	$1.0060 \pm .0013$

* Monte Carlo statistical error at one standard deviation.

Calculated K_{eff} Versus Moderator/Fuel Volume Ratio for
TUFE Lattices

FIGURE 7

P7MG program. This used the small measured axial buckling ($\sim 0.0004 \text{ cm}^{-2}$) and did an explicit radial traverse of the homogenized core out through the reflector. The calculation was otherwise similar to MUFT7.

Results are shown in Table 14 and in Figure 8. Two points are noteworthy:

- (1) The P7MG leakage-corrected eigenvalues are somewhat higher than the full-core RCP01 but in generally reasonable agreement, in view of the approximations (e.g., homogenization and few-group flux weighting) in P7MG.
- (2) For the 15/1-fueled lattices, use of the measured B^2 values in MUFT7 produces eigenvalues significantly below those calculated with the explicit radial traverse by P7MG or by the full-core RCP01. Since MUFT7 and P7MG contain basically similar approximations, except for the explicit spatial treatment in P7MG, this suggests that the measured bucklings are too large. It is clear from the P7MG results that this is due to nonasymptotic spectra in the smaller cores (Ref. 27) - that is, the calculated radial flux shapes agree reasonably well with measurement, but these are not the asymptotic shapes. For the 25/1-fueled lattices, which have smaller bucklings, agreement is satisfactory.

Table 14 - Comparison of K_{eff} from MUFT7 and P7MG
With the Full-Core RCP01 Results

Lattice Ident.	Measured Material B^2 (cm^{-2}) $\times 10^4$	K_{eff}^{+0*}		(MUFT7)	(P7MG)
		MUFT7	P7MG	(Full Core RCP01)	(Full Core RCP01)
15A	85.31 $\pm$.26	.9756 $\pm$.0014	1.0044 $\pm$.0014	.9754 $\pm$.0020	1.0042 $\pm$.0020
15B	94.25 $\pm$.29	.9758 $\pm$.0018	1.0010 $\pm$.0018	.9810 $\pm$.0022	1.0063 $\pm$.0022
15D	114.20 $\pm$.70	.9883 $\pm$.0019	1.0137 $\pm$.0019	.9828 $\pm$.0024	1.0081 $\pm$.0024
25D	64.01 $\pm$.46	1.0046 $\pm$.0033	1.0117 $\pm$.0033	.9971 $\pm$.0035	1.0042 $\pm$.0035
25 $\sqrt{2}$ B	53.55 $\pm$.11	1.0060 $\pm$.0029	1.0129 $\pm$.0029	1.0000 $\pm$.0032	1.0069 $\pm$.0032

* Both normalized to the same cell RCP01. Uncertainties stem from Monte Carlo statistics at one standard deviation. The MUFT7 calculations used measured material B^2 .

Comparison of K_{eff} Values for TUPE Lattices Obtained by Different Methods

FIGURE 8

Two additional studies of TUPE lattices with ENDF/B-IV data have been reported:

At GE-San Jose (Ref. 43), lattices 15A and 15B were analyzed with BWR methods. These methods consisted of multigroup mixed one- and two-dimensional transport theory and transport-corrected diffusion theory techniques. The epithermal spectrum was computed in 68 groups with determination of spatial weighting of the fuel, clad and moderator, and evaluation of the two-dimensional Dancoff interaction. Resolved resonances were evaluated with the intermediate resonance approximation. The space-dependent thermal spectrum was computed by use of the 30 group integral transport theory.

At Combustion Engineering (Ref. 44), lattice 15B was analyzed with the same methods used for the BNL exponentials. In a CEPAC point calculation, the leakage was accounted for by a total B^2 obtained from the measured core dimensions and the radial and axial reflector savings. Also, a PDQ spatial calculation was done with the core explicitly modeled in two dimensions in the horizontal plane. Four-energy-group diffusion parameters were generated in CEPAC for use in PDQ.

Eigenvalue results are compared in Table 15. These show good agreement for the spatially dependent calculations. For the point calculations with measured B_m^2 , the CEPAC result is not as low as obtained with MUFT7.

Cadmium ratios of Th232 capture and U235 fission activations were measured with thorium metal foils and U235-Al foils respectively (Ref. 27). Statistics, from reproducibility, appear to be good for U235 ($\sim 1\%$ in k^{25}); for thorium capture, uncertainties of σ^{02} range from 2% (for core 25V2B) to

nearly 6% (for core 15B). Interpretation of these measurements would require detailed study of experimental effects - mainly cadmium cutoff energy and flux perturbation by the detector foils and cadmium sleeves. No attempt has been made to analyze these measurements, and such an effort is judged to be unpromising.

This conclusion is supported by the calculated ρ^{02} values shown in Table 16. The GE-San Jose and Bettis results agree well but are far below experiment. Since it seems unlikely that the calculations are in error by anywhere near this amount (in view of the good eigenvalues), much of the differences must be attributed to problems with the measurements or their interpretation. This suggests that useful information about thorium resonance capture is not likely to be gained by analysis of these measurements. A similar conclusion was reached in Ref. 44.

Table 15 - Comparison of ENDF/B-IV Calculated
Eigenvalues for TUPE Lattices

<u>Laboratory</u>		<u>Method</u>	<u>TUPE 15A</u>	<u>TUPE 15B</u>
A)	GE-San Jose (Ref. 43)	BWR Methods	1.006	1.005
	Combustion Engineering (Ref. 44)	PDQ	---	1.0023
	Bettis (Ref. 42)	RCP01/P7MG	1.0044 $\pm$.0014	1.0010 $\pm$.0018
B)	Combustion Engineering	CEPAK		.9873
	Bettis	RCP01/MUFT7		.9758 $\pm$.0018
C)	Bettis	Full core RCP01	1.0002 $\pm$.0014	.9947 $\pm$.0013

Table 16 - Comparison of Calculated and
Measured Values of ρ^{02} for TUPE Cores

	<u>15A</u>	<u>15B</u>	<u>Cutoff Energy</u>
Experiment:	$2.70 \pm .09$	$2.09 \pm .12$	
Calculation:			
GE-San Jose (Ref. 43)	1.65	1.42	0.4 eV
Bettis RCP01/P7MG (Ref. 42)	1.69	1.43	0.4 eV
	1.56	1.33	0.625 eV

ρ^{02} = epithermal thorium capture/thermal thorium capture.

3. THUD Lattices (Th-U235-D₂O)

The THUD (Thorium-Uranium-Deuterium) lattice critical experiments were run at Argonne National Laboratory in the late 1950's (Ref. 30). There were 45 uniform single-rod lattices with two types of fuel: U²³⁵O₂-ThO₂ rods with Th/U²³⁵ atom ratios of 15/1 and 25/1 (the same rods used in the TUPE criticals run at Babcock and Wilcox). The lattices were triangular arrays at four pitches (of 2, 3, 4, and 6 X 0.953 cm). At each pitch, several lattices were run differing in number of fuel rods and moderator critical height.

The D₂O purity, initially better than 99%, decreased by 2% over the 2½ year duration of the program. At the end, an independent analysis obtained a purity 1.6% below the last ANL determination, and this discrepancy was not explained.

To test overall consistency of interpretation, we chose to analyze a specific lattice for which measurements were made at the beginning, middle, and end of the program (designated core 91, 233, and 456, respectively). This lattice had 745 fuel rods with Th/U²³⁵ of 25/1 at the 2 X 0.953 cm pitch. Core 456 was also analyzed with a D₂O purity 1.6% below the ANL determination. Composition data for the fuel were those used in analysis of the TUPE lattices (obtained from Ref. 27).

The lattice was described explicitly in three dimensions by the RCP01 Monte Carlo program, including corner posts, grid sheets, cross-shaped control rod guides and the tank.

Table 17 shows eigenvalues obtained in the full core analyses with ENDF/B-IV data. Within statistics, the three cores have consistent eigenvalues, averaging $1.0010 \pm .0013$ (σ). For core 456, a 1.6% reduction of D_2O purity reduced the calculated eigenvalue by 0.22 ± 0.31 percent.

In order to do sensitivity studies, core 456 was analyzed with P7MG, which performed an explicit radial traverse of the homogenized lattice out through the reflector. Zero-leakage cell reaction rates were normalized to a Monte Carlo cell calculation.

For THUD 456, P7MG showed a small (0.17%) decrease in eigenvalue due to the 1.6% reduction of D_2O purity, in good agreement with RCP01. Epithermal/thermal reaction ratios were reduced by 9%, however. Thus, the D_2O purity ambiguity is quite significant for such quantities as ρ^{02} (ratio of epithermal/thermal thorium captures) and δ^{25} (ratio of epithermal/thermal U235 fissions) although it has little effect on eigenvalue.

There is another ambiguity, which has to do with the composition of the 25/1 fuel. Different chemical analyses obtained values for the Th/U atom ratio of 25.33 and 24.65, as shown in Tables XLII and XLV of Ref. 30. Although the former value seems to be preferred and has been used by B&W (Ref. 27 and private communication from Warren Wittkopf), there remains a significant uncertainty in this regard. K_{eff} results quoted here are for Th/U = 25.33. Use of the 24.65 value would raise K_{eff} by 0.75% for these THUD lattices. It would raise K_{eff} for the $25\sqrt{2}$ B and 25D TUPE lattices by 0.9%, worsening agreement with the 15/1-fueled TUPE lattices.

Table 17 - Calculated Criticality of THUD Lattices
(Full Core RCP01)

<u>Lattice Ident.</u>	<u>Date of Run</u>	<u>D2O Purity (%)</u>	<u>Moderator Height (cm)</u>	<u>Calculated Net Leakage</u>	<u>Calculated $K_{eff} \pm \sigma^*$</u>
91	3/58	98.5	132.84	.269	$1.0024 \pm .0023$
233	11/58	97.9	135.08	.268	$1.0018 \pm .0024$
456	12/59	97.2	139.09	.267	$.9987 \pm .0021$
				Average	$1.0010 \pm .0013$
456	12/59	95.6(?)	139.09	.271	$.9965 \pm .0023$

* One standard deviation.

It is noteworthy that neutron leakage out of the reactor tank was calculated to be ~10%. This suggests that questions of room return could be significant for these lattices.

There were a limited number of integral parameter measurements for these cores - specifically, Th232 capture cadmium ratios obtained with metal foils. Interpretation of these measurements would require detailed study of experimental effects such as cadmium cutoff. Since, in most cases, only one irradiation was made and uncertainties are not quoted, it is judged that such an undertaking is not warranted.

4. BNL Exponential Lattices (Th-U233-H₂O and D₂O)

A series of exponential experiments were carried out at the Brookhaven National Laboratory (Ref. 33, 34) in the early 1960's. They were fueled with U233-ThO₂ elements arranged in uniform triangular pitch lattices in light and heavy water. Measurements were made of the material buckling B_m^2 , reflector savings, dysprosium disadvantage factor, ratio of epicadmium to subcadmium capture in thorium (ρ^{02}), and the ratio of Th232 fission to U233 fission (κ_{02}^{23}). Much of the basic data for the BNL experiments has been compiled by Price (Ref. 33), and the most recent survey of the data is presented in the paper by Windsor et al. (Ref. 34).

Recent calculations of the light water lattices with moderator-to-fuel volume ratios ranging from 0.997:1 to 3.004:1 have been reported by Sehgal et al. (Ref. 45). This analysis, which used ENDF/B-IV cross sections in RABBLE/HAMMER cell calculations, produced values of K_{eff} shown in the third column of Table 18. Other calculations were reported by Shapiro et al. (Ref. 44) for lattices with volume ratios ranging from 0.997:1 to 4.2722:1. The analytic model used in this study was based on the CEPAC code and used ENDF/B-IV data and L factors for Th232 and U233 resonance reaction rates which were corrected for resonance interference between Th232 and U233. These results are shown in the fourth column of Table 18. They typically are 0.5 to 1% higher than the corresponding results of Sehgal and exhibit the same upward trend with increasing moderator-to-fuel volume ratio (Figure 9).

More recent studies of the light water lattices have yielded differing results. Weiss and Malaviya (Ref. 46) have analyzed all eight unpoisoned light water lattices using the HAMMER computer code and ENDF/B-IV cross sections. Resonance self-shielding was accounted for with the Nordheim approximation. As shown in column 5 of Table 18, criticality was under-predicted by an amount varying from 0.1 to 1.3%. However, the trend in the calculated K_{eff} versus moderator-to-fuel volume ratio is opposite to that seen in the studies in Refs. 44, 45. The causes for this difference were not discussed in Ref. 46.

Table 18 - Calculated Values of K_{eff} for the BNL U233-Th232-H₂O Exponential Experiments

$\frac{V(\text{Moderator})}{V(\text{Fuel})}$	$B_m^2 (m^{-2})$	Calculated K_{eff}			
		(Ref. 45)	(Ref. 44)	(Ref. 46)	(Ref. 47)
0.997	75.88	0.9823	0.9918	0.9994	1.005
1.384	86.06	0.9895	0.9957	0.9915	1.005
1.7134	89.34	0.9962	1.0013	0.9907	1.006
2.1943	90.35	1.0005	1.0054	0.9889	1.002
3.0043	85.54	1.0037	1.0087	0.9871	0.994
4.2722	69.8	--	1.0117	0.9886	--
6.8449	32.2	--	--	0.9928	--
9.2747	-1.22	--	--	0.9970	--

Bhatt et al. (Ref. 47) have reported an analysis of those BNL exponential lattices with $(M/F) \leq 3.0043$. These calculations used BWR lattice physics methods similar to those applied to the TUPE lattices (Table 15). The calculated critical eigenvalues, column six of Table 18, are overpredicted by ~0.5% for moderator-to-fuel ratios below 2. For $(M/F) = 3.004$, the bias changes sign.

This comparison of K_{eff} values depicts significant method differences, and possibly problems of interpretation, which affect the use of these experiments for nuclear data testing purposes. In an attempt to clear up some of the discrepancies, we have analyzed the unpoisoned light water lattices and all of the heavy water lattices. For the light water lattices three methods to compute K_{eff} were employed. In increasing order of sophistication, they were:

- 1) Single lattice cell RCP01 Monte Carlo calculations corrected for neutron leakage using the MUFT7 program with measured values of the material buckling. Resonance absorption in MUFT7 was handled by a heterogeneous collision probabilistic method. Computed reaction rates in the resonance range and other energy ranges for the unbuckled MUFT calculations generally were in good agreement with the corresponding Monte Carlo cell results.
- 2) Similar to method 1 except that the MUFT7 calculation was replaced by P7MG. This was a one-dimensional (radial), homogenized transport calculation, with a small negative transverse buckling given by $B_z^2 \equiv \frac{1}{\ell_z^2}$, where ℓ_z was the measured relaxation length. (Resonance absorption in P7MG was treated in a manner similar to MUFT7.)

Comparison of Eigenvalues Based on ENDF/B-IV for the
BNL U233-ThO₂-H₂O Lattices

FIGURE 9

- 3) Two-dimensional, explicit representation of the full lattice with the RCP01 Monte Carlo program. The final K_{eff} values included a small correction for axial in-leakage which was determined using P7MG and the measured B_z^2 .

Our results for the light water lattices using method 1 (Table 19, column 1) are in good agreement with those of Ref. 45 and once again exhibit an upward trend with M/F volume ratio for values of M/F below 3.0. The results for lattices with $M/F \geq 3.0$ show greater consistency, with the exception of the 9.27:1.0 lattice which has a negative material buckling. The Method 2 (RCP Cell + P7MG) results appear to be nominally 0.5% higher than the corresponding Method 1 values of K_{eff} and appear to form a more consistent set. They also exhibit a slight rising trend in K_{eff} for M/F values below 3.0. With perhaps the exception of the 1.38:1.0 lattice, the 2D Monte Carlo calculations (Method 3) yield the most consistent set of K_{eff} values (Figure 10).

It appears that the greater overall consistency of interpretation that is achieved with the 2D Monte Carlo results compared with those based on Methods 1 and 2 may be due to the elimination of approximations involved in the homogenization of the lattices. Since the tight lattices have the smallest fueled radii, it is supposed that errors in effective resonance cross sections that occur near the core-reflector interface can significantly affect calculated results. Note that for the larger lattices ($M/F > 3.0$) the rising trend in K_{eff} vs M/F does not persist for Methods 1 and 2.

Table 19 - Comparison of Several Methods for Computing K_{eff} for
the BNL U233-Th232-H₂O Exponential Experiments

<u>V(Moderator)</u> <u>V(Fuel)</u>	Calculated K_{eff}^*		
	Cell RCP01+MUFT7 (Method 1)	Cell RCP01+P7MG** (Method 2)	2D Full Core RCP01+ (Method 3)
0.997	0.9835 $\pm$ 0.0021	0.9993	0.9986 $\pm$ 0.0020
1.384	0.9926 $\pm$ 0.0018	0.9984	0.9907 $\pm$ 0.0022
1.7134	0.9944 $\pm$ 0.0018	1.0038	1.0077 $\pm$ 0.0028
2.1943	1.0025 $\pm$ 0.0021	1.0104	1.0040 $\pm$ 0.0019
3.0043	1.0068 $\pm$ 0.0022	1.0106	1.0028 $\pm$ 0.0024
4.2722	1.0066 $\pm$ 0.0023	1.0074	1.0055 $\pm$ 0.0025
6.8449	1.0091 $\pm$ 0.0026	1.0049	1.0015 $\pm$ 0.0025
9.2747	0.9976 $\pm$ 0.0029	1.0026	0.9969 $\pm$ 0.0023

* Monte Carlo statistical errors are one standard deviation.

** Statistical errors (not shown) are the same as for Method 1.

Comparison of Three Methods to Calculate K_{eff}
 for the BNL U233-ThO₂-H₂O Lattices. ENDF/3-IV
 Data Were Used Throughout
 (See text Section III.B.4)

FIGURE 10

The BNL U233-thorium-D₂O exponential lattices were also analyzed, in a manner similar to that for the light water lattices. Results are shown in Table 20 and Figure 11. With Method 1 (RCP01 Cell + MUFT7), the calculated values of K_{eff} form a nearly consistent set, but criticality is nominally underpredicted by 1.5%. Results based on Method 2 (Cell RCP01 + P7MG) and Method 3 (2D full core RCP + P7MG), with the exception of the M/F = 120.5 lattice, also underpredict criticality by 1-2%. It is not known why K_{eff} calculated for the 120.5:1.0 lattice using Methods 2 and 3 is 10% low. It is probably symptomatic of the fact, recognized by the experimenters, that all the D₂O-moderated lattices were so small as to present significant questions of interpretation. For M/F $\leq$ 55.2, the full core Monte Carlo results depict a slight downward trend in K_{eff} vs the M/F volume ratio.

Also shown in Table 20 are the values of K_{eff} from the Ref. 47 analysis of these D₂O experiments. They are in disagreement with our calculations in that they nominally overpredict K_{eff} by ~1.0%. There is also evidence of a slight downward trend in K_{eff} versus increasing M/F volume ratio. The source of the disagreement is not known.

Another key comparison of results can be made for the predicted episcadmium-to-subcadmium rates of Th232 capture (ρ_c^{02}). Calculated values of ρ_c^{02} are compared with the measurement for the light water lattice in Table 21. Generally, the calculation of ρ_c^{02} are in good agreement with each other, but consistently underpredict the measured values by (5-8)% $\pm$ 4%.

Table 20 - Comparison of Several Methods for Computing K_{eff} for the
BNL U233-Th232-D₂O Exponential Experiments

V(Moderator) V(Fuel)	Calculated K_{eff}^*			(Ref. 47)
	Cell RCP01+MUFT7 (Method 1)	Cell RCP01+P7MG** (Method 2)	2D Full Core RCP01+P7MG (Method 3)	
3.005	0.9857 $\pm$ 0.0032	--	--	1.0186
11.716	0.9806 $\pm$ 0.0020	0.9911	0.9906 $\pm$ 0.0029	1.0201
16.051	0.9821 $\pm$ 0.0020	--	--	1.0215
29.113	0.9822 $\pm$ 0.0019	0.9868	0.9846 $\pm$ 0.0028	1.0139
37.828	0.9890 $\pm$ 0.0020	--	--	1.0184
55.226	0.9792 $\pm$ 0.0020	0.9905	0.9795 $\pm$ 0.0025	1.0053
81.212	0.9913 $\pm$ 0.0027	--	--	1.0117
120.491	0.9892 $\pm$ 0.0033	0.8963	0.9085 $\pm$ 0.0028	1.0006

* Monte Carlo statistical errors are one standard deviation.

** Statistical errors (not shown) are the same as for Method 1.

Calculated K_{eff} (Method 1) vs. M/F Volume Ratio for the BNL U233-ThO₂-D₂O Lattices

FIGURE 11

The predicted values for ρ^{02} for the D_2O lattices are compared with experiment in Table 22. As in the case of the light water lattices, ρ^{02} tends to be underpredicted, but only slightly, with an average bias of $-2\% \pm 4\%$. There appear to be possible problems with the interpretation of the ρ^{02} values for the 81.31:1.0 and 120.5:1.0 lattices.

The analyses of the BNL exponential experiments based on ENDF/B-IV do not yield a wholly consistent set of results. The results obtained for the H_2O lattices show K_{eff} to be overpredicted slightly by the most consistent method of interpretation (explicit 2D Monte Carlo). On the other hand, ρ^{02} is consistently underpredicted by $\sim 8\%$. For the D_2O lattices, the K_{eff} values are underpredicted by $\sim 1.5\%$ while the values for ρ^{02} are slightly low by $\sim 2\%$. A comparison of these results with other thorium capture measurements will be made in Section III.7.

Table 21 - Comparison of Calculated and Measured Values
of ρ^{02} for the Unpoisoned BNL U233-Th232-H₂O Lattices

$\frac{V(\text{Moderator})}{V(\text{Fuel})}$	$\frac{B_m^2}{(\text{m}^{-2})}$	Measured ρ^{02}	Calculated Results			
			Ref. (45)	Ref. (44)	Ref. (47)	This Work
0.997	75.88	1.380 ± 0.042	1.308	1.307	1.283	1.271
1.384	86.06	0.928 ± 0.038	0.901	0.906	0.904	0.865
1.713	89.34	0.754 ± 0.024	0.716	0.720	0.729	0.698
2.194	90.35	0.607 ± 0.026	0.553	0.561	0.575	0.543
3.004	85.54	0.435 ± 0.013	0.406	0.411	0.431	0.396
4.2722	69.8	--	--	--	--	--
6.8449	32.2	0.218 ± 0.008	--	--	--	0.201
9.2747	-1.22	0.170 ± 0.007	--	--	--	0.158

Table 22 - Comparison of Calculated and Measured Values
of ρ^{02} for the BNL U233-Th232-D₂O Lattices

$\frac{V(\text{Moderator})}{V(\text{Fuel})}$	Measured ρ^{02}	Calculated ρ^{02}	$\frac{\rho^{02}(\text{calc.})}{\rho^{02}(\text{meas.})}$
3.005	4.71 ± 0.18^a	4.377	0.929 ± 0.038^b
11.716	0.794 ± 0.028^a	0.759	0.956 ± 0.035
16.051	0.565 ± 0.017^a	0.555	0.982 ± 0.030
29.113	0.297 ± 0.011	0.297	1.000 ± 0.037
37.828	0.234 ± 0.008	0.230	0.983 ± 0.034
55.226	0.166 ± 0.006	0.164	0.987 ± 0.036
81.312	0.104 ± 0.005	0.111	1.066 ± 0.048
120.491	0.089 ± 0.005	0.079	0.888 ± 0.056

^aWeighted average of ρ^{02} determined by the cadmium ratio and thermal activation methods. All other values are based on the Cd ratio method.

^bErrors are of experimental origin only.

5. ETA-I and ETA-II Lattices (Th-U235-D₂O and Th-U233-D₂O)

The ETA-I and ETA-II critical experiments were carried out at Bettis in 1969 and 1970. These were D₂O-moderated tight lattices designed to test thorium and U233 cross sections in very hard neutron flux spectra. For ETA-I, the 26" diameter test lattice contained 2304 U235O₂-ThO₂ fuel rods, with Th/U235 atom ratio of 15/1. (These were the same rods that had been used ten years earlier in the TUPE and THUD experiments.) For ETA-II, there were 864 U233O₂-ThO₂ rods which had been used previously in the BNL exponential lattices. In both experiments it was necessary to use an H₂O-moderated driver lattice of TRX UO₂ fuel rods in order to achieve criticality. At core center, where integral parameters were measured, the flux spectrum was nearly asymptotic, despite the strong spectrum shift between the test and driver lattices. Parameters were measured by activation techniques, as follows (see Ref. 35 for more detail):

- $\delta^{23}(\delta^{25})$ a) cadmium ratio; U233(U235)-Al detector foil.
b) thermal subtraction method, relative to Dyl64-Al detector normalized in reflector.

ρ^{02} Thermal subtraction method, relative to Dyl64-Al detector foil, normalized in reflector. Ge(Li) count of 312 keV γ from fuel and ThO₂ pellets.

- CR* a) From same thorium and U233 (U235) activations.
b) Indirect, from ρ^{02} and $\delta^{23}(\delta^{25})$ as a check.

δ^{02} Fuel and ThO₂ detector pellets. Ge(Li) counting of selected fission product γ 's.

Interpretation of the parameters requires an explicit radial analysis of the full core to obtain leakage corrections. Since spatial activation shapes were not measured, K_{eff} depends significantly on the calculated leakage to the driver and hence on the driver itself. Other integral parameters are relatively insensitive to the driver, however. Due to the tightness of the inner lattice, interaction between thorium and uranium is strongly emphasized as compared to a more thermalized lattice.

Integral parameters at core center were calculated for a cell by RCP01, with leakage corrections from P7MG, which performed a homogenized multigroup P_0 radial traverse. Results from the original analyses are compared with experiment in Tables 23 and 24. For both lattices agreement with experiment is reasonably good, with ρ^{02} approximately 3% high. These analyses used the LWR thorium cross sections (see Ref. 41 for the thorium capture cross section evaluation) which differ in several significant respects from the ENDF/B-IV file. The most notable differences affecting integral parameters are a thermal σ_A of 7.34b (versus 7.40b in ENDF/B-IV) and a fission cross section about 10% higher than ENDF/B-IV. The U235 deck had a fission resonance integral of 259b above 0.625 eV, compared to 268b for ENDF/B-IV material 1261. Its thermal fission cross section was 0.8% below ENDF/B-IV. Other important data (for U233, deuterium and oxygen) were not significantly different from the ENDF/B-IV set.

The ETA lattices have been recently reanalyzed with the same calculational methods and consistent ENDF/B-IV data. These integral parameter results are also shown in Tables 23 and 24. The major differences from the calculated parameters of Ref. 35 are:

Table 23 - Comparison of Measured and Calculated
Integral Parameters for ETA-I (U235-Th-D₂O)

Integral Parameter	Experiment	Calculation ⁺	
		Original (Ref. 35)	ENDF/B-IV
ρ ⁰²	10.54 ± .15	10.92 ± .06	10.9 ± .06
β ²⁵	1.74 ± .02	1.78 ± .01	1.86 ± .01
λ ⁰²	.0166 ± .0009*	.0155 ± .0002	.0141 ± .0001
CR*	.867 ± .009	.882 ± .003	.857 ± .003

* Measured and calculated for the central rod of the nine-rod cell. Other parameter results are for the entire cell.

⁺ Uncertainties on calculated results are from Monte Carlo statistics.

Table 24 - Comparison of Measured and Calculated
Integral Parameters for ETA-II (U233-Th-D₂O)

Integral Parameter	Experiment	Calculation	
		Original (Ref. 35)	ENDF/B-IV
ρ 02	$8.89 \pm .15$	$9.16 \pm .05$	$8.71 \pm .05$
β 23	$2.96 \pm .08$	$2.99 \pm .02$	$2.90 \pm .02$
β 02	$.0181 \pm .0014$	$.0185 \pm .0001$	$.0165 \pm .0002$
CR*	$1.047 \pm .02$	$1.050 \pm .003$	$1.034 \pm .004$

- (1) Slightly (1.5%) reduced Th232 resonance capture (reflected in ρ^{02} and CR*).
- (2) 4.5% higher δ^{25} , of which 3.5% can be attributed to the higher U235 resonance fission integral of ENDF/B-IV. (This also lowers calculated CR* by about 1% and raises ρ^{02} by 1%).
- (3) Approximately 10% lower δ^{02} due mostly to the difference of σ_f^{Th232} .

6. LWBR Criticals (Th-U233-H₂O)

A number of critical experiments in direct support of LWBR were carried out at Bettis during the late 1960's (Ref. 37,38,39). An initial SB series of small seed-blanket assemblies was followed by the BMU experiments, which were larger, multiply-zoned seed-blanket assemblies. There was, finally, the detailed-cell critical assembly, which mocked up the moving seed-blanket arrangement of the LWBR module.

One configuration of the detailed cell (Ref. 39) and one BMU core (Ref. 38) have been analyzed with RCP01 and ENDF/B-IV data. These cores were represented explicitly in three dimensions, including all fuel rods and lattice structural components. There was some homogenization of peripheral structure.

The BMU-1B core consisted of a symmetric hexagonal seed with 810 28-inch-long fuel rods. These were approximately 0.25-inch diameter on a 0.32-inch triangular pitch. The upper 14-inches of each rod was 5 w/o U233O₂-ThO₂, the remainder was 9 w/o. The seed rods rested on an equal number of 14-inch-long ThO₂ rods. The center seven lattice positions were occupied by a Zircaloy pipe. The seed was enclosed in a double-walled hexagonal Zircaloy can, and the 1/8-inch water gap between the walls provided channels for six hafnium control blades. The surrounding blanket contained 811 42-inch-long ThO₂ rods and 60 2 w/o U233O₂-ThO₂ rods

which were of approximately 0.625-inch diameter, on a 0.70-inch triangular pitch. Figure 12 shows the RCP01 representation of the core loading, and Figure 13 shows a schematic view of the axial description.

The detailed cell was a single seed and blanket core, approximately 105 inches high, with a movable central seed region. The seed was a symmetric hexagon containing 15 rows of 0.305-inch diameter fuel rods on a 0.368-inch triangular pitch. An inner blanket region surrounded the seed, separated from it by a double-walled hexagonal Zircaloy can. The 1/8-inch water gap between the walls provided channels for six hafnium control blades. The inner blanket was a nonsymmetric hexagon comprising ordinary and power-flattening regions. The outer blanket comprised a driver region with 2 w/o $U^{233}O_2$ - ThO_2 rods and a reflector region of ThO_2 rods. Figure 14 shows the top view of the core, and Figure 15 shows the complex axial zoning in the seed and inner blanket regions. In the configuration analyzed, the seed was 2 inches below the axial reference position shown in Figure 15.

Eigenvalues obtained with RCP01 and ENDF/B-IV data are shown in Table 25. These are consistent with results obtained for other lattices, as is shown in the next section. Analysis of the measured parameters is rendered difficult by the complexity of these cores, and no attempt has been made to calculate them.

Table 25 - Eigenvalues of LWBR Criticals Calculated With
RCP01 and ENDF/B-IV Data

Core	$K_{eff} \pm \sigma$
Detailed cell (-2 inches)	.9940 $\pm$.0012
BMU-1B	.9963 $\pm$.0014

RCP01 Planar Representation of the LWBK BMU-1B Lattice

FIGURE 12

DIMENSIONS IN CM

RCP01 Axial Representation of the LWBR
BMU-1B Lattice (Schematic)

FIGURE 13

Planar View of the LWBR Detailed Cell Lattice

FIGURE 14

NOTE: w/o DENOTES WEIGHT PERCENT OF URANIUM / TOTAL FUEL. SEED IS SHOWN IN AXIAL - ZERO POSITION.

Axial Arrangement of the LWBR Detailed Cell
Showing Fuel Zoning

FIGURE 15

7. Consistency of Calculated Results for Uranium-Thorium Lattices

In this section analyses of thorium-uranium lattice experiments have been presented. These results are based on ENDF/B-IV data and methods similar to those described in Section II on homogeneous systems. Calculations of K_{eff} for homogeneous criticals revealed a significant underprediction of neutron leakage which warranted a hardening of both the U233 and U235 fission neutron spectra. Analysis of lattices introduces two new complications, namely the presence of a fertile nuclide, thorium, and spatial self-shielding. It is useful now to examine the U233 and U235 lattice K_{eff} results for consistency with those for the corresponding homogeneous criticals and to assess the overall consistency of calculations of integral thorium capture measurements.

The effect of the Standard model U233 fission neutron (see Section II.4) spectrum on K_{eff} for the homogeneous U233 criticals, the BNL H₂O moderated exponential experiments, and the two LWBR criticals considered in this work is depicted in Figures 16 and 17. Figure 16 shows K_{eff} vs total neutron leakage fraction based on ENDF/B-IV. The BNL exponentials and the LWBR cores fit nicely into the pattern that was established for the homogeneous criticals. These same K_{eff} results are shown in Figure 17 when the harder Standard model U233 fission neutron spectrum is employed. Much of the rising trend in K_{eff} vs leakage is eliminated, with the result that most of these experiments, which are H₂O moderated, form a reasonably consistent set. However, the BNL D₂O moderated exponentials are not consistent with this picture since K_{eff} is already underpredicted by 1.5% using ENDF/B-IV, for reasons that are not precisely known. It is expected that a harder U233 fission neutron spectrum would worsen this underprediction.

K_{eff} versus neutron leakage fraction is plotted for comparison in Figure 18 for the U235 systems. It is apparent that the results for the TUPE lattices are reasonably consistent with the rising trend in K_{eff} that was established for the homogeneous systems (Figure 2). A new harder U235 fission spectrum (see Section II) will help eliminate this trend in K_{eff} .

Overall, the studies of K_{eff} for U233 (and U235) homogeneous and heterogeneous systems using ENDF/B-IV supports the conclusion that leakage is underpredicted. This effect is principally associated with fission spectra for U233 and U235 that are too soft. Newer versions of these fission spectra being proposed for ENDF/B-V are harder and should help remove the leakage bias in K_{eff} calculations.

It is apparent that there are only a very limited number of integral Th232 capture measurements potentially useful for data testing purposes. For room temperature ThO_2 rods, these are:

- (1) Effective resonance integrals of ThO_2 rods.
- (2) $\rho^{(12)}$ measurements in the Th-U233 BNL exponentials (H_2O and D_2O moderated).
- (3) ρ^{02} measurements in the D_2O -moderated, hard-spectrum ETA-I (Th-U235) and ETA-II (Th-U233) lattices.

The consistency of these measurements can be seen from Figure 19, which shows experiment/ENDF/B-IV-calculation ratios. The experiments range from 10% high to 3% low, with experimental uncertainties ranging from $\pm 2\%$ to $\pm 5\%$. In addition, there are several questions of interpretation which should be mentioned:

- (1) The measured ThO_2 rod resonance integrals are directly proportional to the dilute resonance integral I . Integral measurements of I are directly proportional to the thorium thermal capture cross section. (Normalization at present is to $\sigma_a = 7.40\text{b}$, consistent with ENDF/B-IV.)
- (2) ρ^{02} measurements do not depend on any such normalization. However, calculated ρ^{02} depends inversely on the thorium thermal σ_a . Without a firm value of σ_a it is difficult to draw conclusions about resonance capture.
- (3) In the ETA cores, ρ^{02} calculations are somewhat sensitive to data other than $\text{Th}232$ capture itself (through leakage corrections and competing absorptions). For example, the ENDF/B-IV-calculated δ^{25} is 7% above experiment. A 10b reduction of the $\text{U}235$ fission integral can remove about half of this difference. Such a change would lower calculated ρ^{02} by 1% and raise calculated CR^* by about the same amount.

Altogether, these experiments, assuming that the ENDF/B-IV $\sigma_a^{\text{Th}232} = 7.40\text{b}$ is fixed, favor an amount of thorium resonance capture slightly above - or at least, not much below - ENDF/B-IV.

K_{eff} Versus Neutron Leakage Fraction for U233 Systems

FIGURE 15

K_{eff} Versus Neutron Leakage Fraction for ^{233}U Systems,
Obtained with the Standard Fission Spectrum Model

FIGURE 17

K_{eff} Versus Neutron Leakage Fraction for U235 Systems

FIGURE 18

Comparison of Thorium Capture Predictions
Based on ENDF/B-IV Data

FIGURE 19

IV. U233-Thorium Graphite Moderated Experiments

In Sections II and III, the analyses of U233-thorium experiments using consistent ENDF/B-IV data and methods have been devoted entirely to water moderated systems (both H_2O and D_2O). There are a number of graphite moderated U233-thorium experiments which will now be described. These include the High Temperature Lattice Test Reactor (HTLTR) experiments and a series of critical experiments related to the measurement of reactivity worths of materials used in High Temperature Gas Cooled Reactors (HTGR). These latter measurements included many U233 and thorium samples. Unlike most experiments discussed previously in this review, these graphite-moderated systems have not been reanalyzed by the authors. However, some analytical results will be presented. In the case of the HTLTR experiments, the analysis was based wholly on ENDF/B-IV.

A. High Temperature Lattice Test Reactor Experiments

The High Temperature Lattice Test Reactor was operated at the Battelle Pacific Northwest Laboratory to study uranium-graphite lattices at temperatures up to $1000^{\circ}C$. Its purpose was to provide information on the nuclear behavior of High Temperature Gas Cooled Reactors and Molten Salt Breeder Reactors (MSBR's) in the temperature range from 500 to $1000^{\circ}C$.

Only a small portion of the test lattice was required for these experiments. Typical measurements included temperature-dependent neutron multiplication factors, total temperature coefficients, Doppler coefficients, and reactivity worths of special poison materials at elevated temperatures.

Five lattices were operated in the HTLTR series of experiments. The first (Ref. 48), which was fueled with a $U^{235}C_2-ThO_2-C$ fuel mixture with $C/Th \sim 200$ and $C/U^{235} \sim 6000$, served to qualify many of the experimental and analytical techniques. The second (Ref. 49), third (Ref. 50), and fifth (Ref. 51) lattices were fueled with $U^{233}O_2-ThO_2-C$ and are of special interest for this survey. In lattice #5, the fuel components were similar to those in the other U^{233} lattices, but the compositions and lattice dimensions were changed to approximate the neutronic characteristics of an MSBR core. Physical properties of these lattices can be found in Table 26 and in the references noted there.

The principal result of the HTLTR $U^{233}O_2-ThO_2$ experiments is $K_{\infty}(T)$, where T is the lattice temperature. These results are summarized in Tables 27-29 for the U^{233} lattices. Also shown are calculated results to be discussed shortly. The determination of $K_{\infty}(T)$ is based on a technique proposed by Heineman (Ref. 52) and involves measurement of the relative reactivity of a central test lattice cell and of a copper ($1/v$) absorber. In addition, the relative absorption rates of the materials in the test cell and the copper absorber are measured. These experimental quantities are combined with calculations of various lattice parameters to yield $K_{\infty}(T)$.

Uncertainties on the measured $K_{\infty}(T)$ were assigned on the basis of random experimental errors. However, there are systematic uncertainties that can affect calculated values of K_{∞} . One of these is an estimated uncertainty of $\pm 1\%$ in the amount of U^{233} in the coated particles used in the fuel mixture, which is worth 0.004 in K_{∞} for each lattice (Ref. 49). Significant impurities (equivalent to 40 atomic parts-per-million of natural boron, plus-or-minus a

Table 26 - Properties of the HTLTR U233-ThO₂ Lattices

	<u>Lattice #2</u>	<u>Lattice #3</u>	<u>Lattice #5</u>
C/U233*	10470	13990	9460
C/Th232**	193	282	146
Fuel Rod Radius (cm)	0.5969	0.5969	0.9982
Square Lattice Pitch (cm)	1.905	1.905	4.763
Reference	49	50	51

* Carbon-to-U233 atom ratio.

** Carbon-to-Th232 atom ratio.

Table 27 - Experimental and Calculated Values of K_{∞} for HTLTR
Lattice #2

<u>Temperature (°C)</u>	<u>Measured K_{∞}</u>	<u>Calculated Values of K_{∞}</u>	
		<u>ENDF/B-IV</u>	<u>Leonard Th</u>
20	1.0587 ± 0.0014	1.0544	1.0626
150	1.0471 ± 0.0017	1.0428	1.0513
300	1.0367 ± 0.0014	1.0334	1.0421
500	1.0297 ± 0.0011	1.0251	1.0347
750	1.0245 ± 0.0011	1.0199	1.0305
1000	1.0237 ± 0.0012	1.0176	1.0292

Table 28 - Experimental and Calculated Values of K_{∞} for
HTLTR Lattice #3

<u>Temperature (°C)</u>	<u>Measured K_{∞}</u>	<u>Calculated Values of K_{∞}</u>	
		<u>ENDF/B-IV</u>	<u>Leonard Th</u>
20	1.064 $\pm$ 0.004	1.1089	1.1141
150	1.060 $\pm$ 0.006	1.0998	1.1058
300	1.056 $\pm$ 0.005	1.0940	1.1005
555	1.057 $\pm$ 0.005	1.0888	1.0966
700	1.057 $\pm$ 0.006	1.0875	1.0962
1000	1.060 $\pm$ 0.007	1.0875	1.0976

Table 29 - Experimental and Calculated Values of K_{∞} for HTLTR Lattice #5 (MSBR)

Temperature (°C)	Measured K_{∞}	Calculated Values of K_{∞}		
		ENDF/B-IV	Leonard Th	Ragan ^a (Ref. 54)
20	1.0291 ± 0.0012	1.0473	1.0523	1.0383 ± 0.0042
300	1.0127 ± 0.0010	1.0301	1.0363	1.0216 ± 0.0042
627	1.0065 ± 0.0010	1.0211	1.0284	1.0134 ± 0.0042
1000	1.0037 ± 0.0012	1.0179	1.0267	1.0120 ± 0.0042

^aUsed ENDF/B-III thorium and ENDF/B-II for other data.

factor of three) detected in the fuel mixture of lattice #3 are worth 3-4% in K_{∞} . There is no direct evidence that similar impurities were present in the fuel mixtures of lattices #2 and #5. These systematic uncertainties make interpretation of absolute $K_{\infty}(T)$ values difficult, particularly for lattice #3.

A recent EPRI study (Ref. 53) of these lattices has been carried out to assess the ability of methods and nuclear data to predict the temperature coefficient of reactivity for HTGR's. It was assumed that the possible systematic uncertainties in measured $K_{\infty}(T)$ were sufficiently temperature independent that the $K_{\infty}(T)$ data could be used to test predictions of

$\frac{1}{K_{\infty}(T)} \frac{dK_{\infty}(T)}{dT}$. The calculations were based primarily on ENDF/B-IV cross section data, but sensitivities to alternate cross sections for U233 and Th232 were studied.

The method used to calculate K_{∞} employed a two group scheme:

$$K_{\infty} = \frac{\tau_1 f_1 (1-p)(1+L^2 B^2) + \tau_2 f_2 p}{1 + (1-p)L^2 B^2} \quad (\text{Eqn. 4})$$

where

$$\tau_1 f_1 = \left(\frac{\nu \Sigma_f}{\Sigma_a} \right)_1, \quad \tau_2 f_2 = \left(\frac{\nu \Sigma_f}{\Sigma_a} \right)_2$$

$$p = \frac{\Sigma_{1 \rightarrow 2}}{\Sigma_{1 \rightarrow 2} + \Sigma_{a1}} \quad L^2 = \frac{D_2}{\Sigma_{a2}}$$

The buckling, B^2 , and K_∞ also satisfy the critical equation,

$$K_\infty = 1 = \frac{K_\infty [1 + (1-p)L^2 B^2]}{(1+\tau B^2)(1+L^2 B^2)} \quad (\text{Eqn. 5})$$

where

$$\tau = \frac{D_1}{\Sigma_{1 \rightarrow 2} + \Sigma_{a_1}} .$$

Equations 4 and 5 were solved using an iteration procedure. A lattice cell calculation with zero buckling was used to estimate the parameters of Eqn. 4 and K_∞ . These values were then placed in Eqn. 5 to yield the first estimate of B^2 . New cell parameters were generated using this estimate of B^2 , and K_∞ and B^2 were recalculated. This procedure was repeated until changes in B^2 became negligible.

The $K_\infty(T)$ calculations of Ref. 53 using ENDF/B-IV nuclear data are shown in Tables 27-29. Also shown are the results of calculations using the Leonard thorium cross section data which were also described in Ref. 53. As shown in Table 30, the Leonard thorium set has a lower dilute resonance integral and a 3% higher thermal absorption cross section. Reduced resonance capture overcompensates the larger thermal absorption cross section and leads to calculated $K_\infty(T)$ values ~1% higher than for ENDF/B-IV.

Although the systematic uncertainties make it difficult to draw conclusions about nuclear data from the $K_{\infty}(T)$ values alone, correlation of calculated and measured temperature coefficients permit a more unambiguous interpretation. The particular correlation employed was of the form:

$$K_{\infty}^{\text{meas.}}(T) = K_{\infty}^{\text{calc.}}(T) [A+B(T-293)] \quad (\text{Eqn. 6})$$

$$\begin{aligned} \left[\frac{1}{K_{\infty}(T)} \frac{dK_{\infty}(T)}{dT} \right]_{\text{meas.}} &= \left[\frac{1}{K_{\infty}(T)} \frac{dK_{\infty}(T)}{dT} \right]_{\text{calc.}} + \left(\frac{B}{A+B(T-293)} \right) \\ &\approx \left[\frac{1}{K_{\infty}(T)} \frac{dK_{\infty}(T)}{dT} \right]_{\text{calc.}} + B \end{aligned} \quad (\text{Eqn. 7})$$

since $A \sim 1.0$, $B \sim 10^{-5}$, and $T \sim 10^3$. The parameter A plays the role of a normalizing coefficient, and the parameter B expresses the bias in the calculated temperature coefficients. The analyses based on ENDF/B-IV (Table 31) yield values for B which are consistently positive indicating a nonconservative prediction of $\frac{1}{K_{\infty}(T)} \frac{dK_{\infty}(T)}{dT}$. Therefore, thorium resonance capture is probably overpredicted for these lattices using ENDF/B-IV, which is somewhat inconsistent with other calculated thorium capture parameters (Figure 19).

A change in the thermal eta of U233 to 2.284 (from the ENDF/B-IV value of 2.2972) produced negligible effects on calculated $K_{\infty}(T)$ and B values. Due to its reduced resonance capture, and hence smaller contribution to the temperature coefficient from the Doppler effect, the Leonard thorium data produced B values uniformly $0.4 \times 10^{-5} \cdot K^{-1}$ lower than those based on ENDF/B-IV.

It was concluded in Ref. 53 on the basis of the reduced temperature coefficient bias that the Leonard thorium data is preferred over ENDF/B-IV for predicting temperature coefficients of thorium fueled lattices.

Finally, the $K_{\infty}(T)$ predictions for lattice #5 can be compared with a previous study made by Ragan at the Oak Ridge National Laboratory (Ref. 54). Computational methods and nuclear data used in the design of the MSBR were tested against the HTLTR lattice (lattice #5) which most nearly mocked up the neutronic properties of the MSBR. The results, which are shown in the last column of Table 29, have been adjusted for known errors in the calculational strategy. The quoted uncertainty in the calculations is dominated by the $\pm 0.4\%$ uncertainty in K_{∞} due to the $\pm 1\%$ in U233 content. Most of the cross sections were taken from ENDF/B-II files; those for thorium were taken from ENDF/B-III. The calculated values of $K_{\infty}(T)$ are higher than the measurements, but by only 1% , and have nearly the same dependence on T as those generated in the more recent Ref. 53 study.

Table 30 - Comparison of Thorium Data Sets used in HTLTR Analyses

	<u>ENDF/B-IV</u>	<u>Leonard Set</u>
$\sigma_a(.0253 \text{ eV})$	7.40 b	7.615 b
$RI_a(I > 0.50 \text{ eV})$	85.8 b	83.7 b

Table 31 - Calculated Values^a of the Temperature Coefficient
Bias Parameter B for the HTLTR Lattices

	<u>Lattice #2</u>	<u>Lattice #3</u>	<u>Lattice #5</u>
ENDF/B-IV Data	$+0.21 \pm 0.07 \times 10^{-5}$	$1.55 \pm 0.19 \times 10^{-5}$	$+0.41 \pm 0.09 \times 10^{-5}$
Leonard Thorium	$-0.18 \pm 0.07 \times 10^{-5}$	$1.08 \pm 0.19 \times 10^{-5}$	$+0.03 \pm 0.09 \times 10^{-5}$
Leonard Thorium and Alternate U233	--	$1.11 \pm 0.19 \times 10^{-5}$	--

^aAll values are in units of $^{\circ}\text{K}^{-1}$.

B. HTGR Critical Experiments

Critical experiments in support of the HTGR program were carried out at Gulf General Atomic from 1966 - 1969 (Ref. 55). The reactivity worths of materials used in HTGR-type reactors, including samples of U233 and Th232 which are of interest for this survey, were measured in a series of U235-graphite cores.

Five cores were constructed having carbon-to-U235 (C/U) ratios of 432, 859, 1718, 2500, and 5000. This provided a wide range of spectrum hardness which could be used to emphasize cross section data in both fast and thermal energy ranges. Each core consisted of uranium-graphite rods arranged on a triangular pitch and had three distinct regions: 1) a center (exact) region made up of fuel elements of the desired C/U ratio, 2) a driver region containing fuel elements and pure graphite elements in combination to produce the same C/U ratio as the exact region, and 3) a reflector region of pure graphite elements.

This simple design was intended to allow analysis by one-dimensional transport methods. Because geometrical effects were minimized, the experiments are useful for testing cross section data. A comparison of analytical and experimental results is reported in Ref. 55, based mainly on ENDF/B-I data for the fissile nuclides. Although these measurements have not been reanalyzed, we have attempted to infer whether the original calculation/experiment comparisons would be improved using ENDF/B-IV data.

The reactivities of test elements were measured relative to that for reference graphite elements of the same density. Measured worths of U233 and Th232 test elements (Tables 32 and 35) are considered in Ref. 55 to be reliable to $\pm 1\%$. The majority of the measurements were analyzed using the GAZE one-dimensional diffusion theory program. Details of the calculational method can be found in Ref. 55, but only the results are presented here.

Table 32 shows the calculation-to-experiment comparisons for U233. The calculations were performed using two sets of U233 data which are summarized in Table 33. In both cases ENDF/B-I U235 cross sections were used in the adjacent core regions. It is seen that the ORNL-RPI U233 cross section data (set #2) improved the calculation-to-experiment comparisons. However, the calculated reactivity worths were still 4-9% higher than those measured.

Sensitivities to several U233 data components were also investigated. For the heaviest loaded U233 element (C/U-432), the reactivity worths in four of the cores were recalculated for several different perturbations of the ORNL-RPI U233 data. These results are reproduced in Table 34. Also shown are the calculated Fast/Thermal flux ratios (2.38 eV cutoff). It is to be noted that the discrepancy increases with Fast/Thermal flux ratio. Data adjustments which were effective in reducing the discrepancies include a reduction in σ_a of U233, an increase in the capture cross section below 2.38 eV, and a decrease in the epithermal ($E > 2.38$ eV) fission cross section.

Table 32 - Comparison of Calculated and Measured
U233 Reactivity Worths in HTGR Lattices

U233 Element ^a	Calculated ^b Worth (\$)	Calculated ^c Worth (\$)	Measured Worth (\$)	% Difference	
				Calc.-Expt.	x 100
				Expt.	
<u>Core C/U-5000</u>					
5000	--	--	0.110	--	
1718	--	--	0.284	--	
432	--	0.930	0.854		8.2
<u>Core C/U-2500</u>					
5000	--	0.080	0.076		5.0
1718	--	0.187	0.186		0.5
432	0.593	0.584	0.565		3.2
<u>Core C/U-1718</u>					
5000	--	0.061	0.054		11.5
1718	--	0.137	0.135		1.5
432	0.439	0.431	0.415		3.7
<u>Core C/U-859</u>					
5000	--	0.031	0.025		19.4
1718	--	0.080	0.079		1.3
432	0.265	0.263	0.243		7.6
<u>Core C/U-432</u>					
5000	--	0.020	--	--	
1718	--	0.051	0.052		-2.0
432	--	0.181(d)	0.164		9.4

^a Numbers signify carbon/U233 atoms in the test element.

^b Used the U233 data from Ref. 56 (Set #1).

^c Used the U233 data from Ref. 57 (Set #2).

^d Calculation done with the 1DF transport code.

Table 33 - U233 Cross Section Data used in HTGR Analyses

	<u>Set #1</u> <u>(Ref. 56)</u>	<u>Set #2</u> <u>(Ref. 57)</u>	<u>ENDF/B-IV</u>
	<u>2200 m/s Values</u>		
σ_{mv}	48.2	48.2	45.9
σ_{nf}	525	525	525.1
σ_c	0.0918	0.0918	0.0874
σ_f	2.293	2.293	2.297
σ_a	2.503	2.503	2.4978
	<u>Epithermal Values</u>		
Resonance Capture ^a Integral	135	140	134.2
Resonance Fission ^a Integral	775	770	761.7
σ_{epi}	0.174	0.182	0.176

^aEpithermal values to a 0.5 eV cutoff.

An examination of the ENDF/B-IV nuclear data for U233 (Table 33) reveals that within the framework of the same calculational model its use will probably not significantly improve the calculation-to-experiment comparisons. Even though the fission integral is slightly reduced, the capture integral is also smaller so that the epithermal alpha is 0.176 versus the 0.182 of the ORNL-RPI set. In addition, the thermal capture is smaller, which will also tend to widen the calculation-to-experiment discrepancies.

Experimental and calculated reactivity worths for thorium-loaded elements are shown in Table 35. The thorium data were taken from the 1966 Drake and Nichols evaluation, and calculated negative worths are consistently higher than measurement by an average of 6.5%. Except for the C/U-432 core, there is a slight tendency for the calculated results to become worse as the C/U atom ratio is decreased and the flux spectrum hardens.

Despite several modeling approximations, which are discussed in Ref. 55, the calculated thorium worths indicate that thorium resonance capture is overpredicted by about 6%. It is expected that this should also be true for ENDF/B-IV, although a reanalysis has not been performed to confirm this. New analyses of these experiments using consistent ENDF/B-IV data and current generation methods will be required to draw further conclusions and are considered to be worthwhile. For now, this review suggests that for graphite moderated systems ENDF/B-IV thorium tends to overestimate resonance capture.

Table 34 - Summary of U233 Reactivity Worth Perturbations
in HTGR Lattices^a

Core	C/U-2500	C/U-1718	C/U-859	C/U-432
Fast/Thermal Flux Ratio (2.38 eV Cutoff)	3.40	4.62	12.07	15.33
Measured Worth (\$)	0.565	0.415	0.243	0.164
Best Calculated Worth (\$)	0.584	0.431	0.263	0.181
Discrepancy (%)	+3.4	+3.9	+8.2	+9.4
Discrepancy % if ^b :				
²³³ σ_c is increased 10% for				
a) $2.38 \leq E \leq 14.9 \times 10^6$ eV	+2.7		+6.2	
b) $0.1 \leq E \leq 0.414$ eV	+2.8		+7.8	
c) $0.1 \leq E \leq 2.38$ eV	+1.9		+5.8	
²³³ σ_f is decreased 10% for				
$2.38 \leq E \leq 14.9 \times 10^6$ eV	+0.9		0.4	
²³³ σ_f changed to 2.494 from 2.503				
	+2.1		+6.6	

^aCalculations were made for the heaviest loaded U233 special element (C/U-432).

^bData changes were made one at a time.

Table 35 - Comparison of Calculated and Measured
Thorium Reactivity Worths in HTGR Lattices

<u>Special Thorium^a</u> <u>Element</u>	<u>Calculated</u> <u>Worth (\$)</u>	<u>Measured</u> <u>Worth (\$)</u>	<u>% Diff.^b</u>
<u>Core C/U-5000</u>			
Th(50)	--	-0.046	--
Th(100)	--	-0.085	--
Th(200)	--	-0.157	--
Th(300)	-0.237 ^b	-0.218	8.0
Th(500)	-0.349 ^b	-0.330	5.4
<u>Core C/U-2500</u>			
Th(50)	-0.059	-0.053	10.2
Th(100)	-0.105	-0.094	10.5
Th(200)	-0.180	-0.171	5.0
Th(300)	-0.244	-0.232	4.9
Th(500)	-0.349	-0.342	2.0
Th(500)	-0.343 ^c	-0.342	0.3
<u>Core C/U-1718</u>			
Th(50)	-0.061	-0.055	9.8
Th(100)	-0.108	-0.093	13.9
Th(200)	-0.182	-0.170	6.6
Th(300)	-0.245	-0.230	6.1
Th(500)	-0.347	-0.331	4.6
Th(500)	-0.342 ^c	-0.331	3.2
<u>Core C/U-859</u>			
Th(50)	-0.057	-0.056	1.8
Th(100)	-0.102	-0.092	9.8
Th(200)	-0.177	-0.159	7.6
Th(300)	-0.230	-0.211	8.3
Th(500)	-0.320	-0.296	7.5
Th(500)	-0.311 ^c	-0.296	4.8

(Continued)

Table 35 (Continued)

<u>Special Thorium^a</u> <u>Element</u>	<u>Calculated</u> <u>Worth (\$)</u>	<u>Measured</u> <u>Worth (\$)</u>	<u>% Diff.^b</u>
<u>Core C/U-432</u>			
Th(50)	--	-0.048	--
Th(100)	-0.085	-0.080	5.6
Th(200)	-0.143	-0.139	2.8
Th(300)	-0.191	-0.185	3.1
Th(500)	-0.265	-0.258	2.6
Th(500)	-0.252 ^c	-0.258	-2.4

^aNumber in parentheses signifies grams Th/element.

^b% Diff. = (Calculated Worth-Experiment)/Experiment X 100.

^cResonance calculation used the Nordheim integral approximation for both the absorber and moderator. All others used the asymptotic approximation of the moderator to the collision density.

V. Summary

This paper has reviewed selected U233 and Th-U233 experiments useful as thermal reactor data-testing benchmarks. Two major questions were addressed:

- (1) To what extent do these experiments provide a consistent and adequately detailed test of the differential data?
- (2) How well do current (ENDF/B-IV) differential data predict the integral results?

The existing integral experiments provide good beginning-of-life criticality information for these systems. However, except for a few thorium capture measurements (which are fairly consistent) integral parameter results are extremely limited in number and scope.

In the area of criticality, the homogeneous U233-H₂O systems provide a clear-cut picture: With ENDF/B-IV data, calculated eigenvalues show an upward trend of approximately 2% with decreasing H/U. There is a similar pattern for homogeneous U235 systems, and these trends appear to be due mostly to an underprediction of fast neutron leakage.

For H₂O-moderated thorium-uranium lattices, eigenvalues based on full-energy-range, explicit-geometry Monte Carlo calculations are generally high, consistent with the trend observed for homogeneous criticals. This suggests that leakage is underpredicted with ENDF/B-IV for these lattices also, and that thorium capture is reasonably well predicted (Figure 16). For the BNL D₂O-moderated exponentials, on the other hand, calculated eigenvalues are low by 1.5%.

In the area of integral parameters, the number of useful measurements is limited. Although the ENDF/B-IV thorium evaluation does not reflect recent

experimental data, predicted thorium resonance capture in lattices is only slightly below the nominal best integral result. However, the spread of integral values is large - from ~8% above calculation to ~3% below (Figure 16). It may be noteworthy that in the D₂O-moderated systems calculated thorium capture is slightly higher (relative to experiment) than in the H₂O-moderated systems.

There is a good deal of evidence from these analyses to support hardening of the prompt fission neutron spectrum of U233. The best spectrum representation is obtained with the Standard model, which uses two fully-accelerated fission fragments and a stationary source. This model yields a prompt mean energy of $\bar{E}_p = 2.082$ MeV, and effectively removes the trend in calculated K_{eff} versus leakage (Figure 17).

The ENDF/B-IV U235 prompt fission spectrum is also too soft. In this case equally good fits to recent differential experiments are obtained with the Watt model ($\bar{E}_p = 2.027$ MeV) and the Standard model ($\bar{E}_p = 2.073$ MeV). For U235 systems, K_{eff} is also sensitive to resonance alpha (the ratio of capture/fission integrals), and the ENDF/B-IV value is 7 (+ 4)% below integral experiment. Both the Standard and Watt spectrum models can eliminate the trend of K_{eff} versus leakage depending on choice of resonance alpha (Figure 5). In addition, the Standard model gives good agreement with integral measurements of age in H₂O and fission-spectrum-average σ_f^{U238} (Table 9, 10). The strong model dependences reflect the fact that differential spectrum measurements do not effectively cover the full energy range of the spectra, especially below 1 MeV.

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