Sources of Neutronics Data Involving Thorium or ^{233}U and Light Water Moderation

L. C. Davenport

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Pacific Northwest Laboratory
Richland, Washington 99352

SUMMARY

A literature search has been conducted to locate sources of neutronics data for light water moderated systems which contain thorium and/or uranium-233. It is concluded that insufficient data is currently available to validate neutronics design methods for licensing the $^{233}\text{UO}_2\text{-ThO}_2$ fuel cycle in light water reactors.

A summary of the neutronics data sources found is reported in this document. These sources include critical and exponential experiments with lattices of fuel rods containing $^{233}\text{U} + \text{Th}$ or $^{235}\text{U} + \text{Th}$. A few experiments using homogeneous aqueous solutions of $^{233}\text{UO}_2(\text{NO}_3)_2$ or $^{233}\text{UO}_2\text{F}_2$ are also included.

The only critical lattice data using both $^{233}\text{U} + \text{Th}$ came from the LWBR program. All these experiments were zoned radially and in most cases axially also. Geometrically clean lattice critical data were measured for the CETR and TUPE programs. Both series used $^{235}\text{UO}_2\text{-ThO}_2$ pellets. A series of 21 exponential experiments using 3% $^{233}\text{UO}_2$ - 97% ThO_2 fuel vibratory compacted to 92% of theoretical density in Zircaloy-2 tubing was performed at BNL using both unpoisoned and boric acid poisoned H_2O moderator.

For completeness, homogeneous systems are listed in which basic neutronics data have been measured. However, it is expected that most data concerning homogeneous systems will be applied to criticality safety problems rather than neutronics methods validation.

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Tables and figures from other documents, along with their supporting material, are reproduced and included.

INTRODUCTION

A literature search was performed for applicable experimental data on systems containing either the fissile isotope ^{233}U or the fertile element thorium, but hopefully both together. This report lists the sources of neutronics data which have been found in the search and summarizes what measurements have been performed. A similar compilation of criticality safety data has already been published [1].

Very little of the neutronics data was found to be available currently which is applicable to the verification of design methods for $^{233}\text{UO}_2\text{-ThO}_2$ fueled light water reactors. The only critical lattice data using $^{233}\text{U} + \text{Th}$ came from the LWBR program. However, all these experiments were zoned radially and in most cases axially also. The following identifies what information is available and indicates features which make much of it inappropriate for use in design method verification and validation.

The earliest (and geometrically simplest series) were the small "Seed and Blanket" experiments utilizing $^{233}\text{UO}_2$ and $^{235}\text{UO}_2$ as fissile fuel to allow direct comparison of lattice characteristics for the two fissile isotopes. However, the uranium was diluted only with ZrO_2 in the seed rods. Thorium was contained just in the blanket, either as ThO_2 or as 1 wt% $^{233}\text{UO}_2\text{-ThO}_2$.

In the second and third series of LWBR experiments, geometrical complexity was greatly increased since the objective was to provide data for developing and refining design and analysis methods and for determining power distributions in representative core mockups. The "BMU Series" used nine different fuel rod types to achieve a complex pattern of radial and axial zoning, and the "Detailed Cell" was similar. Heterogeneities, such as control rods and multiple seed-blanket modules were present.

Geometrically clean lattice criticals were performed for the CETR and TUPE programs, but both series used ^{235}U as the fissile isotope. CETR fuel consisted of $^{235}\text{UO}_2\text{-ThO}_2$ pellets in 304 stainless tubing, and supported the design of the Indian Point PWR. Part of the lattices were designed without heterogeneities, but the other part studied three-zoned, canned element cores with offset cans to allow space for control rods or followers. TUPE fuel consisted of $^{235}\text{UO}_2\text{-ThO}_2$ pellets in aluminum tubing, and represented a parallel

series of five BWR experiments. In addition to the five clean core experiments, leakage experiments including partial water height and borated moderator were performed in two cases.

A series of 21 exponential experiments using 3% $^{233}\text{UO}_2$ - 97% ThO_2 fuel vibratory compacted to 92% of theoretical density in Zircaloy-2 tubing was performed at BNL using both unpoisoned and boric acid poisoned H_2O moderator. Although the lattices were clean (no zoning and no control rods), criticality was not achieved, thus limiting the usefulness of the data.

For completeness, homogeneous systems are listed in which basic neutronics data have been measured. However, it is expected that most data concerning homogeneous systems will be applied to criticality safety problems rather than neutronics methods validation.

In presenting information that was located while preparing this report, it was deemed most efficient and accurate to directly reproduce some of the introductory and summary sections from seven of the original source documents. Much of the literature cited might be worth obtaining if the reader intends to do further study in this area. Some literature is only supplementary and is simply listed for completeness.

Every reasonable effort has been made to summarize all applicable data and it is unlikely that any major unclassified data sources were overlooked. Search methods used included both computer and manual searches through Nuclear Science Abstracts, manual searches through all Babcock and Wilcox reports, manual searches through all Westinghouse WAPD-T- and WAPD-TM- series reports, and cross checks of all references listed in useful reports located. Table 1 summarizes the available data.

TABLE 1. Experimental Measurements Performed in Lattices
Containing Thorium or ^{233}U and Light Water

	CETR	TUPE	Small Seed & Blanket	BMU Series	Detailed Cell	BNL Exponentials
Critical Mass/Critical Configuration	1	1	1	1	1	
Control Worth	1					
Boron Worth in Moderator	1	1				1
Reactivity Coefficients:						
Moderator Height	1	1				
Temperature	1		1		1	
Water Channels						
Voided Channels						
Absorbing Materials	1			1		
Fissionable Materials				N.A.		
k_{∞} , Infinite Media Multiplication Factor	1	1				
M^2 , Migration Area	1	1				
λ , Reflector Saving or Extrapolation Length	2	2				2
B_0 , Material Buckling	2	2				2
Thorium Capture Traverse				2		
Thorium Fast-Fission Traverse			2	2	2	
^{233}U Fission Traverse			2	2		
^{235}U Fission Traverse			2			
Epi-Cadmium Activation Traverse	2	2				
Thermal (Bare Foil) Activation Traverse		2	2			2
Thorium Capture Activation Within Fuel Pellets				2	2	
^{233}U Fission Activation Within Fuel Pellets				2	2	
Grid Flux Depression					2	
Thermal Flux Distribution in Control Rods & Followers	2					
Fast Advantage Factor			2			
Thermal Disadvantage Factor	2	2	2			2
CR^{02} , Thorium Cadmium Ratio	2	2				N.A.
p^{02} , Thorium Resonance Escape Probability		2				
p^{02} , Thorium Epi-thermal-to-Thermal Capture Ratio			2	2	2	2
δ^{02} , Ratio of Thorium Fission to ^{233}U Fission					2	
CR^{23} , ^{233}U Cadmium Ratio	2					
p^{23} , ^{233}U Resonance Escape Probability						
δ^{23} , ^{233}U Epi-Thermal-to-Thermal Fission Ratio			2	2	2	
CR^{25} , ^{235}U Cadmium Ratio	2	2				
p^{25} , ^{235}U Resonance Escape Probability		2				
δ^{25} , ^{235}U Epi-Thermal-to-Thermal Fission Ratio			2			
Power Distribution Across Entire Core	3			3	3	
Power Distribution & Peaking Within a Fuel Bundle	3,2				3	
Axial Fuel Rod Gap Effect	3,2			3	3	
Neutron Energy Spectrum						
Kinetics Measurement						

Legend

- 1 Reactivity Measurement
- 2 Foil Irradiation
- 3 Gamma Scan

HETEROGENEOUS EXPERIMENTS

Nearly all of the 41 references cited (involving Th or ^{233}U in light water moderation) originated in only four companies or laboratories. Babcock and Wilcox performed the Consolidated Edison Thorium Reactor (CETR) and Thorium Uranium Physics Experiments (TUPE). Westinghouse - Bettis Atomic Power Laboratory has done a variety of experiments under the Light Water Breeder Reactor (LWBR) Development Program. Brookhaven National Laboratory performed a series of exponential experiments. All lattice experiments are summarized in Table 2, a presentation which was first used in reference [1]. Finally, Gwin and Magnuson from Oak Ridge measured eta and other parameters, but this was in homogeneous systems.

Critical lattice experiments using ^{235}U as fissile and ^{232}Th as fertile isotopes were performed by Babcock and Wilcox to support the design of the Consolidated Edison Thorium Reactor — the Indian Point PWR. There were two objectives: (1) to provide data on the infinite medium properties of the lattice inside the canned elements in the various CETR loading zones, and (2) to obtain specific information on the full-size CETR core, specifically control rod worth and power distribution measurements.

Babcock and Wilcox also performed the TUPE critical lattice experiments using ^{235}U as fissile and ^{232}Th as fertile isotopes to support BWR reactor designs. The objective of TUPE was to measure selected neutronics and criticality parameters for a series of cold clean criticals. Two fuel types were used, with Th/ ^{235}U atom ratios of 25.34 and 15.00. Metal to water volume ratios covered the range from 0.3 to 1.0. Measurements included critical mass, material buckling from flux maps, cadmium ratios for Th and ^{235}U , and thermal flux depression.

An extensive series of critical lattice experiments has been performed at Westinghouse - Bettis Atomic Power Laboratory in support of the LWBR program. The objective has been to develop a technology that will significantly improve the utilization of the nation's nuclear fuel resources employing light water technology. To achieve this, work has been directed toward analysis, design, component tests, and fabrication of a water-cooled, thorium oxide fuel cycle breeder core to be installed and operated at the Shippingport Station.

TABLE 2. Summary of Critical Lattice Experiments with Light Water Moderated ^{233}U or Thorium

	Fissile Material(a)	Fertile or Diluent Material(a)	Thorium Fissile Ratios(b)	Rod O. D. (in.)	Metal/Water Ratios	Lattice Type	Number of Experiments	Program Name
	^{235}U	^{232}Th	15.0 & 25.8	0.312	0.15 - 1.12	Square	10	CETR
	^{235}U	^{232}Th	15.0 & 25.3	0.309	0.48 - 1.02	Square	5	TUPE
Seed	^{233}U or ^{235}U	Zr	0	0.225	0.22 & 0.64	Sq. & Hex.	10	Small Seed & Blanket
Blanket	^{233}U or None	^{232}Th	99, -	0.570	0.95 & 9.3			
Seed	^{233}U or None ^(c)	^{232}Th	7,10,19,49,-	0.25	1.19 & 1.64	Hexagonal	8	BMJ Series
Blanket	^{233}U or None	^{232}Th	49, -	0.63	2.89 & 1.09			
Seed	^{233}U or None ^(c)	^{232}Th	15, 23, -	0.305	1.65	Hexagonal	7	Detailed Cell
Blanket	^{233}U or None	^{232}Th	34, 46, 57,-	0.528, 0.572, 0.626	1.7 - 2.9			
	^{233}U	^{232}Th	32	0.499	0.15 - 1.35	Hexagonal ^(d)	21	BNL Exponentials

(a) All materials are in oxide form

(b) A dash indicates no fissile content

(c) Axial reflector region

(d) Exponential experiments.

The earliest and geometrically simplest series of LWBR experiments was referred to as the small ^{233}U fueled seed and blanket criticals. They used either ^{233}U or ^{235}U as fissile fuel to allow direct comparison of their criticality and neutronic characteristics. The lattices were radially zoned into seed and blanket regions, with thorium in the blanket only. ZrO_2 was used to dilute the fully enriched uranium in the seed. Additional differences between seed and blanket regions included rod size and lattice pitch.

The BMU series of cores assembled for the LWBR program investigated specific effects important to breeder reactor development. Investigations concerned the effects of zoning, coupling, and grids of relatively large volumes. Complex radial and axial fuel zoning patterns were used as well as multiple seed modules.

The $^{233}\text{UO}_2\text{-ThO}_2$ detailed cell criticals assembled for the LWBR program devoted major effort to three-dimensional power distribution measurements in the seed and blanket regions of the core. The objective was to provide basic experimental data and experimental confirmation of the nuclear analysis model calculations. Again, complex radial and axial fuel zoning patterns were used in the assemblies.

Uniform subcritical lattices were constructed at Brookhaven National Laboratory with 3% $^{233}\text{UO}_2\text{-97% ThO}_2$ fuel in light water moderator. The lattices were clean with no perturbing influences. Their objective was to provide a data set covering a wide range of lattice parameters for use in evaluating calculational methods. However, since they were subcritical, flux measurements were used to calculate the material buckling of the lattices. Measurements were made for hexagonal lattices using a single rod type, at eight different lattice pitches from about 1:1 to 9.1:1. These lattices poisoned with boric acid, yielded a total of 21 values of B_m^2 .

CONSOLIDATED EDISON THORIUM REACTOR (CETR) EXPERIMENTS

Neutronics experiments were performed on two types of lattices in this program. The first type of lattice was geometrically clean and lacked heterogeneities due to fuel element cans or water channels between cans. The second type of lattice studied used three-zoned, canned-element cores that had the same cross-sectional dimensions as the above clean lattices. Both types of lattices had a cylindrical array of fuel pins on a square pitch.

The main report describing the criticality and neutronics experiments is BAW-119 Rev. 1 [2]. The objectives of the experiments and the measurements performed are summarized in their two page introductory section. These pages are included in this report. Most of the original report includes information on neutronics experiments but is too long to reproduce here. However, about 15% of it is reproduced in reference [1] including the complete physical description of the cores.

Reference [3] describes the nuclear characteristics of the Consolidated Edison Thorium Reactor (CETR). The major aspects of nuclear design discussed are the core operating lifetime, control rod worth, and the power density distribution. A large number of design calculations and comparisons with experiments were performed for the CETR, and these are discussed at length.

Reference [4] only covers limited aspects of the CETR experiments. The report discusses the four clean lattices and their measured material buckling, rate of change of reactivity with water height, thermal flux ratios, cadmium fractions for ^{235}U and thorium, derived k_{∞} and M^2 , conversion ratio, effective resonance integral for ThO_2 , and reactivity life considerations.

Reference [5] is much more limited in scope, dealing only with criticality safety. The CETR and TUPE lattices are listed solely in Table 2 of the reference [5].

References [6] and [7] are the original paper submitted to the IAEA panel on light water lattices, and the final published proceedings of the

panel, respectively. Experiments with two fuels of interest here are designated CETR-4 and CETR-7, while the TUPE-4 and TUPE-7 fuels will be covered in the next section of this report. Reference [7] appears to be the best summary of neutronics data available outside of the actual Babcock and Wilcox reports. The summary and first four tables are reproduced later in this report. Following the neutronics presentation, the paper discusses the method of analysis, detailed analysis and comparison with experiment, and analysis by one-group model and comparison with experiment.

Reference [8] appears to be an abbreviated presentation of reference [7] which includes only the CETR and TUPE fuels.

In summary, references [2-8] appear to present the full range of CETR neutronic experimental results, analyses, calculations, and comparisons that have been published. These reports were all authored by Babcock and Wilcox personnel.

INTRODUCTION

This report describes the critical experiments with pin-type oxide fuel elements for the Consolidated Edison Thorium Reactor (CETR).

The objectives of the experiments were twofold:

1. To provide data on the infinite medium properties of the lattice inside the canned elements in the various CETR loading zones, and
2. To obtain specific information on the full size CETR core, specifically control rod worth and power distribution measurements.

To accomplish the first objective, measurements were done on four ThO_2 - UO_2 cores containing no heterogeneities from can walls or water channels between cans. Each core had a cylindrical array of pin elements on a square pitch. The measurements included the following:

1. Critical mass determination
2. Flux traverses to determine buckling
3. Cd ratio of ThO_2 to determine resonance escape probability for thorium
4. Cd ratio of U-235 to determine the ratio of resonance to thermal multiplication
5. Disadvantage factors
6. Rate of change of reactivity with water height as a function of water height*
7. Lattice poisonings.*

To accomplish the second objective, measurements were done on a three-zone, canned-element core with the same cross sectional dimensions

* These two measurements provide a relationship between infinite multiplication and migration area.

as the reference core. These measurements included:

1. Rod worth determinations.
2. Power distribution measurements.
3. Thermal flux distribution through control rods and rod followers.

One important use of these measurements is as a check on theoretical calculations. Where the results and calculations agree well, the theory may be used to calculate properties of cores slightly different from the cores measured.

This report also describes the measurements and gives a brief interpretation of the results in some cases. A comparison of experimental results with theoretical calculations and the relationship of this comparison to the reference design core is presented in "Consolidated Edison Thorium Reactor Physics Design", BAW-120, Rev. 1.*

* Barringer, H. S., et al., Consolidated Edison Thorium Reactor Physics Design, BAW-120, Rev. 1, July 1960, The Babcock and Wilcox Company, Lynchburg, Atomic Energy Division.

ANALYSIS OF WATER MODERATED UO_2 AND ThO_2 LATTICES^(a)

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Abstract — Résumé — Аннотация — Resumen

ANALYSIS OF WATER MODERATED UO_2 AND ThO_2 LATTICES. An extensive series of critical and exponential experiments has been performed with uniform lattices of fuel rods composed of low enriched UO_2 and of fully enriched UO_2 in ThO_2 . These lattices were moderated by H_2O and by mixtures of D_2O and H_2O . Lattice spacings ranged from somewhat looser lattices than needed for maximum buckling to very tight lattices in which more than half the fissions were epithermal. Measurements included buckling, ratio of epithermal to thermal captures in U^{235} (ρ_{25}) and in Th^{232} (ρ_{22}), ratio of epithermal to thermal fissions in U^{235} (δ_{25}), ratio of thermal flux in the moderator to that in the fuel (ϕ_M/ϕ_F), and water height reactivity worth leading to $\delta k_{eff}/\delta B^2$.

The primary theoretical analysis of these lattices utilizes a 39-group, Greuling-Goertzel slowing down model in the B_1 approximation for a bare reactor. A comparison analysis is made with the usual one-group, bare reactor model, modified to account for epithermal fissions. The simple model correlates the more thermal lattices with good accuracy, but for the lattices with larger fractions of epithermal fissions the epithermal neutron flux spectrum departs markedly from the usual $1/E$ shape and the multi-group treatment is required for accurate analysis. The multi-group model using experiment bucklings computes criticality of each of the more than 30 lattices in the series well, for most lattices within $\pm 1\%$ in k_{eff} . Calculated and measured critical bucklings agree well over the entire range $-5M^{-2} < B^2 < 115M^{-2}$, for most lattices within $\pm 2M^{-2}$.

Four types of fuel were used in the series: 3% enriched UO_2 , 4% enriched UO_2 , ThO_2 with Th to U^{235} atom ratio = 25, and ThO_2 with Th to U^{235} atom ratio = 15. Although the fuel rod diameters were small, ranging from 0.86 cm to 1.27 cm, calculations indicate 10 to 20% reduction of the effective resonance integral of U^{235} due to self-shielding in the resonances. Most of the computed shielding effects occur within the individual fuel rod, with smaller effects coming from the lattice mutual shielding between fuel rods. With this shielding taken into account, calculated values of δ_{25} are in good agreement with measured values over the range $0.05 < \delta_{25} < 1.6$.

Single lump resonance integrals for U^{235} calculated from resonance parameters are in agreement with recent values measured by Pettus, using both activation and reactivity determinations. Similar calculations for Th^{232} are in appreciably poorer agreement with experiment. Because of this, the resonance cross-sections of Th^{232} were adjusted to agree with the measured integral values. Lattice mutual shielding Dancoff effects are very important for the tighter lattices, and there is some uncertainty in determining them. The computed effective lattice resonance integrals lead to calculated values of ρ_{25} and ρ_{22} in good agreement with experiment in the ranges $4 < \rho_{25} < 16$ and $1.3 < \rho_{22} < 8$. Because of the large values of ρ in some of these lattices, a technique was developed to measure ρ directly utilizing intermediate dysprosium foil activations. This technique gave appreciably improved accuracy compared to conventional cadmium ratio measurements.

Calculation of thermal flux ratios in water lattices is complicated by both geometrical and spectral considerations. Two simplified methods are used to calculate ϕ_M/ϕ_F and the results are not in very good agreement with measured values. Because nearly all thermal neutrons in these lattices are absorbed in the fuel, the poor agreement has little effect on the calculated critical buckling or on other lattice parameters. More complex treatments involving both geometrical and spectral dependence are needed for more accurate calculations of this ratio.

Measured and calculated values of $\delta k_{eff}/\delta B^2$ are in fair agreement over the range 30 to 70 cm^2 . The results are consistent with a separable Gaussian non-leakage probability, but the multi-group calculations indicate an appreciable interaction between leakage and resonance absorption. Thus values of the age can not be determined directly from water height measurements, but rather these data give a different extensive parameter for analysis.

(a) Reproduced with IAEA permission from:
 (Proc. Panel, Vienna, 1962), Technical
 Reports Series No. 12, pp. 177-210, IAEA.
 Vienna (1962).

TABLE I
FUEL ROD DESCRIPTION

Fuel designation	Fertile material	U ²³⁵ Fertile atom ratio	Clad material	Fuel* density g/cm ³	Rod diam cm	Fuel diam cm	Clad thickness cm	Active rod length cm
NMSR-3	U ²³⁸	0.03128	Stainless steel	9.28	1.270	1.126	0.072	168.9
NMSR-4	U ²³⁸	0.04242	Stainless steel	9.43	1.270	1.126	0.072	168.9
SSCR-4	U ²³⁸	0.04242	Stainless steel	9.46	1.208	1.126	0.041	169.4
CETR-4	Th ²³²	0.0388	Stainless steel	8.35	0.792	0.660	0.048	121.9
CETR-7	Th ²³²	0.0667	Stainless steel	8.35	0.792	0.660	0.048	121.9
TUPE-4	Th ²³²	0.0395	Aluminium	8.45	0.785	0.594	0.086	152.4
TUPE-7	Th ²³²	0.0667	Aluminium	8.33	0.782	0.660	0.036	152.4

* Fuel is in the form of either UO₂ or ThO₂.

TABLE II
LATTICE CONFIGURATIONS

CASE	Fuel designation	Pitch, cm	M/W*	Mole % D O in moderator	Type of experiment
1	NMSR-3	2.196	0.355	0	Critical
2	"	1.684	0.807	0	"
3	"	1.554	1.102	0	"
4	NMSR-4	2.381	0.288	0	"
5	"	2.196	0.355	0	"
6	"	1.684	0.807	0	"
7	"	1.554	1.012	0	"
8	CETR-4	1.023	0.890	0	"
9	"	0.966	1.116	0	"
10	CETR-7	1.933	0.152	0	"
11	"	1.367	0.358	0	"
12	"	1.115	0.656	0	"
13	"	1.023	0.890	0	"
14	"	0.966	1.116	0	"
15	TUPE-4	1.446	0.301	0	"
16	"	1.222	0.480	0	"
17	TUPE-7	1.222	0.480	0	"
18	"	1.023	0.860	0	"
19	"	0.978	1.024	0	"
20	SSCR-4	1.511	1.0064	0	"
21	"	1.511	1.0064	49.66	"
22	"	1.511	1.0064	69.70	"
23	"	1.511	1.0064	73.77	"
24	"	1.511	1.0064	73.77	"
25	"	1.511	1.0064	76.50	"
26	"	1.511	1.0064	80.65	Exponential
27	"	1.511	1.0064	89.14	"
28	TUPE-7	0.983	0.990	55.38	"
29	"	0.983	0.990	60.40	Critical
30	"	0.983	0.990	71.94	Exponential
31	"	0.983	0.990	81.16	Critical **
32	"	0.983	0.990	81.96	Exponential

* Metal-to-Water ratio (M/W) is the volume ratio of fuel rods to water moderator.

** Central zone of a two-zone configuration.

TABLE III

EXPERIMENTAL AND CALCULATED LATTICE PARAMETERS

CASE	Critical buckling, (M^{-2})		Calculated k_{eff} with measured B^2	$\delta k_{eff}/\delta B^2$, cm^2	
	Expt.	Calc.		Expt.	Calc.
1	69	64.7	0.988	27.5	29.4
2	64	60.6	0.989	32.3	33.7
3	51	49.8	0.996	21.4	36.9
4	92	92.2	1.001	---	27.0
5	93	97.3	1.012	23.5	27.2
6	86	83.8	0.993	25.7	31.6
7	69	69.3	1.001	28.5	35.0
8	29.3	27.7	0.994	39.8	38.2
9	26.2	21.5	0.981	---	41.3
10	57	64.8	1.022	---	27.6
11	94	93.7	0.999	---	27.6
12	84	82.7	0.996	---	30.9
13	72	70.8	0.995	---	33.9
14	61	60.0	0.996	34.5	36.9
15	53.55	55.8	1.007	---	32.3
16	64.01	66.0	1.007	33.5	34.1
17	114.2	115.5	1.004	27.7	29.7
18	94.25	92.1	0.993	39.4	35.8
19	85.31	83.1	0.992	43.7	38.5
20	88.0	85.2	0.991	31.9	33.8
21	44.0	41.0	0.985	48.1	51.0
22	18.60	18.4	0.999	60.2	64.8
23	14.33	15.8	1.009	62.1	69.0
24	14.09	15.6	1.011	63.2	68.9
25	10.77	12.8	1.013	65.4	72.0
26	6.53	7.9	1.010	---	77.1
27	-4.68	-2.0	1.025	---	89.5
28	44.8	41.7	0.981	---	61.3
29	39.0	37.1	0.988	59.3	65.1
30	31.4	28.1	0.961	---	76.2
31	---	16.6	---	---	88.8
32	22.7	15.8	0.940	---	89.8

TABLE IV

EXPERIMENTAL AND CALCULATED LATTICE PARAMETERS

CASE	δ_{25}		ρ_{Fertile}		Fertile material	ϕ_W/ϕ_F	
	Expt.	Calc.	Expt.	Calc.		Expt.	Calc.
1	-	0.064	-	1.09	U	-	1.21
2	-	0.137	-	2.26	U	-	1.17
3	-	0.188	-	3.01	U	-	1.15
4	-	0.067	-	1.15	U	-	1.31
5	-	0.080	-	1.38	U	-	1.28
6	-	0.177	-	2.95	U	-	1.22
7	-	0.245	-	3.97	U	-	1.21
8	0.157	0.163	-	1.01	Th	1.14	1.12
9	0.210	0.206	1.28	1.26	Th	-	1.12
10	-	0.047	-	0.31	Th	-	1.27
11	-	0.100	-	0.66	Th	-	1.22
12	-	0.018	-	1.17	Th	-	1.19
13	0.221	0.244	1.49	1.58	Th	1.19	1.18
14	0.292	0.310	2.08	1.99	Th	-	1.18
15	0.053	0.046	-	0.31	Th	1.22	1.14
16	0.085	0.070	-	0.45	Th	1.16	1.13
17	0.130	0.123	-	0.81	Th	1.22	1.21
18	0.181	0.220	-	1.41	Th	1.12	1.19
19	0.266	0.263	-	1.67	Th	1.22	1.18
20	0.266	0.238	4.18	3.88	U	1.27	1.21
21	0.495	0.439	8.3	7.11	U	1.20	1.20
22	0.73	0.668	10.4	11.32	U	1.13	1.21
23	-	0.755	-	12.23	U	-	1.21
24	-	0.755	-	12.23	U	-	1.21
25	0.88	0.823	16.4	13.36	U	1.17	1.21
26	1.02	0.955	-	15.58	U	-	1.21
27	1.56	1.590	-	23.80	U	-	1.22
28	0.56	0.524	-	3.43	Th	-	1.19
29	0.65	0.578	-	3.82	Th	1.18	1.19
30	0.81	0.771	-	5.23	Th	-	1.19
31	1.20	1.045	8	7.37	Th	1.16	1.19
32	1.16	1.083	7	7.65	Th	-	1.20

THORIUM URANIUM PHYSICS EXPERIMENTS (TUPE)

Neutronics experiments were performed using five geometrically clean lattices. Each used fuel pins of a single type located on a square pitch, in a cylindrical array. In addition, an extensive series of leakage experiments was performed on four of the lattices. For two of them, measurements were made with boric acid added to the water at several different concentrations.

The main report describing the criticality and neutronics experiments is BAW-1191 [9]. The objective and summary are included in this report. Most of the original report includes information on neutronics experiments and is too long to reproduce here. However, about 25% of the original is reproduced in reference [1] including the complete physical description of the cores plus the entire experimental program.

References [10-13] are the only monthly reports published during the project. They mainly discuss project status and interim results. It is concluded that these reports need not be obtained if one has access to the final report [9].

References [5-8] have already been discussed under the CETR experiments. Again reference [7] appears to be the best summary of neutronics data available outside of the actual Babcock and Wilcox reports.

It is concluded that references [5-13] present the full range of TUPE neutronic experimental results, analyses, calculations, and comparisons that have been published before 1970. Again, these reports were all authored by Babcock and Wilcox personnel.

In the last two years the TUPE lattices have been proposed as thermal reactor benchmarks for testing the newer ENDF/B cross section sets. All five geometrically clean lattices have been used according to references [14-15]. In addition, J. Hardy Jr. and others at Bettis Atomic Power Laboratory have produced other undocumented analyses using TUPE data in reviewing thorium-U233 benchmarks for the Cross Section Evaluation Working Group.

It is expected that a number of additional reports will appear which discuss TUPE lattices for thermal reactor benchmark testing. Hence, references [14-15] should be considered a sample of the current literature in this area.

OBJECTIVES

The objectives of TUPE experiments are to measure lattice parameters for a series of cores with metal-to-water volume ratios (M/W) in the range of 0.3 to 1.0 and for $N_{O_2}/N_{H_2} = 25/1$ and $15/1$. Specifically, material buckling (B_M^2), critical size and mass, and nonleakage probability as a function of buckling, $P(B^2)$, are to be determined for each core.

If the above objectives are met then the infinite multiplication constant, k_∞ , is

$$k_\infty = P^{-1}(B_M^2),$$

and k_{eff} for any core of geometric buckling B_g^2 is

$$k_{eff} = P^{-1}(B_M^2) P(B_g^2).$$

Additional objectives are to determine as many detailed two-group parameters as possible, including separation of k_∞ into slow and fast components, resonance escape probability from thorium, and ratio of average thermal flux in water to average thermal flux in fuel.

SUMMARY

The Thorium Uranium Physics Experiment (TUPE) is a study of uniform lattice cores moderated with light water. TUPE cores contain thorium oxide-uranium oxide pins clad with aluminum and have metal-to-water volume ratios, M/W , ranging from 0.3 to 1.0.

Critical experiments for the Consolidated Edison Thorium Reactor (CETR) have provided data on thorium oxide-uranium oxide fuel, in stainless steel cladding, moderated with light water. These data are for cores with M/W near one — the region of interest in pressurized water reactor design. Experiments at Argonne National Laboratory (ANL) have provided data on similar fuel in aluminum cladding moderated with heavy water. These data are for the low metal-to-water volume ratios that accompany heavy water moderation.

TUPE is intended to complement these data by providing information on lattices with ratios between the CETR and ANL values — the region of interest for boiling water reactor design.

The basic aim of the TUPE project is to measure lattice parameters of cores with ratios ranging from 0.3 to 1.0, and to compare the results with different theoretical models.

A series of "clean core" measurements were done on five lattices (the contract required three), including (1) flux mappings, (2) critical mass determinations, M_c , (3) cadmium ratio measurements for U-235 - C_{25} and thorium, C_{02} and (4) flux depression measurements.

These data were then analyzed to obtain material buckling, B_m^2 ; the ratio of infinite medium thermal multiplication to infinite medium resonance multiplication, k_2/k_1 ; the thermal disadvantage factors for moderator $\bar{\phi}_w/\bar{\phi}_f$ and for cladding, $\bar{\phi}_{Al}/\bar{\phi}_f$; the thorium resonance escape probability, p_{02} ; and the U-235 resonance escape probability. These results are summarized in Table 1.

Leakage experiments, including partial water height experiments and lattice poisoning experiments, were performed on two cores.

Leakage experiments were performed on two additional cores without lattice poisoning (the contract required leakage experiments on two cores). The leakage experiments consisted of determination of the water height coefficient of reactivity, $\partial\rho/\partial h$, as a function of water height and boric acid concentration in the moderator. These data were analyzed to determine migration area, M^2 ; and infinite medium multiplication constant, $k_\infty = k_1 + k_2$. These results are also summarized in Table 1.

The above results were checked against several of the available calculational models. The two-group diffusion model was selected as a standard and this model was used in calculations. A four-group diffusion model and a 20-group diffusion (Multigroup) model have been used for comparison purposes.

Two methods of handling the two-group model are available. The first, referred to as the "Two-Group, Two-Region Model" is a computer calculation based on two-group multiregion diffusion theory. The second, referred to as the "Modified Two-Group, One-Region Model", is essentially a hand calculation based on obtaining the parameters which appear in the two-group formulation of the "four-factor formula" for the multiplication constant. Complete calculations of the TUPE assemblies were done by both methods. The metal-to-water volume ratio was selected as the independent variable and calculations were made over a range of ratios that includes all TUPE assemblies. Calculated values for all parameters measured are included in Table 1.

The most direct measurement that can be made in a critical experiment on a given lattice is the critical size. A comparison of the measured critical radii with radii calculated using the Multigroup Program (Table 1) shows excellent agreement for all cores studied.

The measurement of individual parameters contributing to the critical size is not as direct and in general does not give as good agreement with calculated values.

Thermal Properties

The quantities leading to the calculation of the thermal utilization were calculated using both the Wigner-Wilkins and Maxwellian thermal

flux spectra. The differences in the resulting f -values are negligible. There is good agreement between computed and experimental values for the flux ratio in water to the flux ratio in fuel, in the sense that the differences do not affect the value of f .

Resonance Properties

Agreement between the calculated and measured values of the U-235 resonance escape probability is good. Results indicate that the total infinite multiplication is insensitive to this quantity, but that the use of a single, infinitely dilute, value for all calculations is not justified if it is important to differentiate between thermal and resonance multiplication.

The agreement between the measured and calculated thorium resonance escape probabilities is poor. The measured values are derived from the measured cadmium ratios, and are lower than calculated values.

Infinite Multiplication Factor

The measured and computed infinite multiplication factors are shown in Table 1. Measured values obtained using both a one-group model and a Gaussian slowing down model are given. The two results indicate the range of results possible using different models with the correct model probably somewhere in between. A comparison of experimental results with calculated values indicates that the correct model may shift toward the Gaussian model as the metal-to-water ratio increases.

Leakage Constants

The effective age has been computed using the Selengut-Goertzel kernel for hydrogen slowing down, and Fermi age theory for slowing down in other materials. This calculation was done using both the B&W Spectral Code and the MUFT-4 Program.

The measured migration areas based on a one-group model agree well with the migration areas calculated by the Spectral Code, but the measured areas lead to high infinite multiplication values, and the calculated areas lead to high critical radius values. The ages calculated by the MUFT-4 Program are lower than the Spectral Code results and are believed to be closer to the true values.

TABLE 1
SUMMARY OF RESULTS

Parameter	Core Number				
	25 $\sqrt{2}$ B	25 D	15 D	15 B	15 A
M/W	0.301	0.480	0.480	0.860	1.024
N ₀₂ /N ₂₅	25.34	25.34	15.00	15.00	15.00
N _H /N ₂₅	528	329	165	131	78.0
R _C , cm*	27.3 27.2▽	23.5 23.2▽	15.5 15.8▽	17.0 17.6▽	18.1 18.3▽
M _C , kg*	13.46	13.99	9.73	20.79	25.96
B _m ² × 10 ⁻³ , cm ⁻²	5.355	6.401	11.42	9.425	8.531
C ₂₅	19.8 19.9▽	12.8 13.4▽	8.7 9.5▽	5.5 5.7▽	4.8 4.9▽
C ₀₂	2.82 4.13▽	2.26 3.12▽	1.78 2.24▽	1.48 1.70▽	1.37 1.56▽
k ₂ /k ₁	18.8 18.9▽	11.8 12.4▽	7.7 8.5▽	4.5 4.7▽	3.8 3.9▽
$\bar{\phi}_w/\bar{\phi}_f$	1.22 1.14▽	1.16 1.12▽	1.22 1.19▽	1.12 1.18▽	1.22 1.18▽
$\bar{\phi}_{Al}/\phi_f$	1.04▽	1.05▽	1.08▽	1.09▽	1.10▽
P ₀₂	0.916 0.945▽	0.878 0.918▽	0.870 0.910▽	0.810 0.864▽	0.756 0.849▽
P ₂₅	0.960 0.956▽	0.933 0.932▽	0.886 0.895▽	0.816 0.822▽	0.786 0.792▽
M ² , cm ² **		41.7	36.6	54.4	57.4
M ² , cm ² ★		32.9	28.5	37.7	38.4
M ² , cm ² △	41.7▽	45.5▽	45.3▽	52.2▽	54.8▽
M ² , cm ² □			35.3▽	42.7▽	
k _∞ **		1.27	1.42	1.49	1.49
k _∞ ★		1.23	1.38	1.41	1.39
k _∞	1.22▽	1.29▽	1.48▽	1.42▽	1.40▽

* Reflected cylinder — 5 ft high

** One-Group Model

★ Gaussian Model

▽ Calculated Values

□ MUFT Code B — Self-Consistent Age Approximation — One-Group Model

△ Spectral Code, One-Group Model

SMALL ^{233}U FUELED SEED AND BLANKET CRITICALS

Eight lattice arrangements were studied in this series of experiments. Rod-type seed fuel elements containing 26 wt% $\text{UO}_2\text{-ZrO}_2$ (fully enriched with either ^{233}U or ^{235}U) were utilized so that a direct comparison could be made between the lattice characteristics of the two fuels. A surrounding blanket region contained rod-type elements with either natural ThO_2 or 1 wt% $^{233}\text{UO}_2\text{-ThO}_2$. The eight assemblies were of two principal types. The first was rectangular with a central seed region surrounded by a wet blanket with a metal-to-water ratio of about unity. The second was hexagonal with a central seed region surrounded by a tightly packed dry blanket with a metal-to-water ratio of about 9.2. Criticality was achieved by adjusting the size of the seed region, yielding a clean geometry. All eight lattices were run at room temperature. Additionally, two of the wet blanket lattices were studied at 480° F and 650 psi.

The primary report describing the criticality and neutronics experiments is WAPD-TM-614 [16]. The introductory and general description pages are reproduced in this report. Many sections of the original report include information on the neutronics measurements, but the report is too long to reproduce all of it. The experiments did include measurements of critical axial bucklings, thermal disadvantage factors, fast advantage factors, seed power shapes, fast flux shapes through seed and blanket regions, the ratio of epithermal-to-thermal thorium captures in blanket regions, and the ratio of fissions in the seed to thorium captures in the blanket. Calculations were performed with diffusion theory and with other, more sophisticated techniques for comparison with experiments.

More than a year before the final report was published, the experimental program was described in references [17] and [18]. The former, a paper presented at the ANS meeting, can be a useful summary of the experimental program if the main report is not available.

References [19] and [20] cover the same topic of thermal disadvantage factors and fast advantage factors except that [19] is only a summary whereas [20] is the complete paper. Reference [21] continues further by discussing

neutron leakage effects on the fast advantage factor. Reference [22] is a summary comparing measured and calculated activation shapes taken along a radial. Finally, reference [23] is an invited ANS paper that again summarizes the general critical experiment program. However, it gives perspective on the entire program, so it is reproduced in full in this report.

The second set of experiments mentioned in reference [23] is given in detail in reference [24] and summarized in reference [25]. Three one-dimensional seed-blanket slab cores were used to measure neutronics parameters in ThO_2 blanket regions as a function of distance from seed regions. A relatively narrow blanket region with non-asymptotic neutron spectrum was used to represent a typical seed-blanket power reactor situation. The main purpose of the experiments was to test the ability of the calculational model and the nuclear data to predict blanket parameters in an environment representative of a large seed-blanket reactor. Further information is available in the introduction of reference [24] which is reproduced in this report.

A series of one-, two-, and three-module cores containing highly enriched uranium seeds and thorium oxide blankets is described in reference [26]. The measured core parameters included k_{eff} , seed power distribution, fast and thermal activation traverses, epithermal to thermal thorium capture ratios, epithermal to thermal ^{235}U fission ratios, thermal disadvantage factors, fast advantage factors, plus some information on core intermodule coupling. Each module consisted of a narrow annular hexagon of ^{235}U seed plus an inner and outer ThO_2 blanket. Measured parameters were compared with diffusion and transport theory models with reasonable agreement, and areas of disagreement are noted and explained.

I. INTRODUCTION

As part of the overall effort to develop the physics technology required to design a light water-moderated seed-and-blanket reactor utilizing the uranium-233-thorium fuel cycle, an experimental program was initiated to provide confidence in the fundamental data available for uranium-233 and to verify the ability to calculate the physics characteristics of these systems. A series of eight small seed-and-blanket critical assemblies were studied at the Bettis Atomic Power Laboratory. Rod-type seed fuel elements which contained either uranium-233 or uranium-235 were utilized so that a direct comparison could be made between the lattice characteristics of the two fuels. Also, blanket regions which contained rod-type elements with either natural ThO_2 or 1 w/o $\text{U}^{233}\text{O}_2\text{-ThO}_2$ were compared. The eight assemblies were of two principal types. The first type was a rectangular array having a central seed region surrounded by a wet blanket with a metal-to-water ratio of about one, and the second was a hexagonal array having a central seed region surrounded by a tightly packed dry blanket with a metal-to-water ratio of about 9.2. All experiments on these eight assemblies were conducted at a temperature of 20°C . In addition, two of the wet blanket assemblies were studied in the Bettis High-Temperature Test Facility at 480°F and 650 psi.

The assemblies were constructed to be nearly clean critical and were used to measure a wide range of reactor physics parameters for comparison with calculation. Measurements included critical axial bucklings, thermal disadvantage factors, fast advantage factors, seed power shapes, seed-and-blanket activation shapes, ratios of epithermal-to-thermal thorium captures in blanket regions, and ratios of epithermal-to-thermal uranium-233 or uranium-235 fissions through seed-and-blanket regions. Calculations for comparison with experimentation were performed with diffusion theory and with more sophisticated Monte Carlo and neutron transport techniques.

This report summarizes the measurements and analysis performed during the experimental program and describes the LWBR facility and the critical assemblies. Inadequacies in the calculational method are evaluated and possible solutions suggested. Table 1 summarizes the measured and calculated parameters on the 8 cold and 2 hot critical assemblies.

TABLE 1. SUMMARY OF MEASURED AND CALCULATED PARAMETERS ON EIGHT COLD AND TWO HOT CRITICAL ASSEMBLIES

Parameter	Core							Hot SB-3	Hot SB-4	
	SB-1	SB-2	SB-2 1/2	SB-3	SB-4	SB-5	SB-6			SB-7
Seed-and-Blanket Activation Shapes										
U ²³⁵ foils	X	X	--	--	X	X	X	--	--	X
U ²³³ foils	--	--	--	X	--	--	X	X	X	--
Th foils	X	--	--	--	--	X	X	X	X	X
Dy-Al foils	--	--	--	--	--	X	X	X	--	--
Th Fission Shape in Blanket										
ρ^{02*} in Blanket	X	--	--	--	X	X	--	X	X	--
$\delta^{25\dagger}$ in Seed	--	--	--	--	--	X	--	--	--	X
δ^{25} in Blanket	--	--	--	--	--	X	--	--	--	X
$\delta^{23\dagger}$ in Seed	--	X	--	X	--	--	--	X	X	--
δ^{23} in Blanket	--	--	--	X	--	--	--	X	X	--
Thermal Disadvantage Factor										
Seed	X	X	--	X	X	X	X	--	X	--
Blanket	X	--	--	X	X	X	--	X	--	--
Fast Advantage Factor										
Seed	--	--	--	--	--	X	X	--	--	--
Blanket	X	--	--	--	X	--	--	--	--	--

* ρ^{02} is defined as the ratio of epithermal-to-thermal thorium captures.

$\dagger\delta^{25}$ is defined as the ratio of epithermal-to-thermal uranium-235 fissions.

$\dagger\delta^{23}$ is defined as the ratio of epithermal-to-thermal uranium-233 fissions.

II. DESCRIPTION OF LWBR CRITICAL FACILITY, FUEL, AND CORES

B. LWBR Facility

The LWBR critical facility was modified for the performance of experiments with uranium-233 fuel. The developmental nature and lack of experience with uranium-233 dictated a conservative approach in modifying the facility and procedures to study the uranium-233 fuel.

Experiments were conducted in a small 6-foot diameter tank installed in a 17-foot diameter vessel. This larger vessel was also designed to contain critical assemblies, but was used in this instance (with a lid over the top) to prevent the spread of contamination if a fuel rod should leak during the uranium-233 experiments.

C. Description of Fuel

Four types of fuel elements were obtained for this program, and are given in Table 2.

The two seed types and the two blanket types had the same physical characteristics and are shown in Figure 6. These seed fuel elements were 0.255-inch OD x 20.8 inches long, while the blanket elements were 0.570-inch OD x 20.8 inches long. The seamless tubing for these rods was Zircaloy-2, and had a specified clad thickness of 0.016 inch for the seed and 0.038 inch for the blanket. All tubing was free-standing for the intended application at 480° F and 650 psi.

All elements contained 15.00 ± 0.06 inch of pressed and sintered fuel pellets with a 2.0-inch Zircaloy-2 insert in each end. Zircaloy end-caps were welded on each end of the rods. As shown in Figure 6, these end-caps had threaded holes which were used during core assembly. Also shown in Figure 6 is a typical test rod with removable end-caps. These caps contained double O-ring seals which could be tightened to make a gas-tight seal.

The uranium-233 containing pellets were manufactured from approximately 13 kg of uranium-233, which contained 38 ppm of uranium-232. Uranium-233 in the form of uranyl nitrate was solvent extracted by Oak Ridge National Laboratory (ORNL) in two batches. The first batch of 6.950 kg was used for the blanket pellets, while the second

TABLE 2. DESCRIPTION OF FUEL ELEMENT TYPES

Type	Nominal Composition	Number of Rods	Fissile Fuel (kg)
Seed	26 w/o $\text{U}^{233}\text{O}_2\text{-ZrO}_2$	396	5.3
Seed	26 w/o $\text{U}^{235}\text{O}_2\text{-ZrO}_2$	625	8.8
Blanket	ThO_2	1386	- -
Blanket	1 w/o $\text{U}^{233}\text{O}_2\text{-ThO}_2$	1331	5.4

TABLE 3. DESCRIPTION OF LATTICE CONSTRUCTION

Core	Seed Composition	Blanket Composition	Blanket Type	H/U* in Seed	W/U† in Seed	H/Th* in Blanket	W/Th† in Blanket	Core Geometry	Lattice Type
SB-1	26 w/o $\text{U}^{235}\text{O}_2\text{-ZrO}_2$	ThO_2	Wet	37	26	4.4	1.97	Rectangular	Square
SB-2	26 w/o $\text{U}^{235}\text{O}_2\text{-ZrO}_2$	ThO_2	Wet	38	27	4.4	1.97	Rectangular	Square
SB-2.5	26 w/o $\text{U}^{235}\text{O}_2\text{-ZrO}_2$	- -	- -	38	27	- -	- -	Rectangular	Square
SB-3	26 w/o $\text{U}^{235}\text{O}_2\text{-ZrO}_2$	1 w/o $\text{U}^{233}\text{O}_2\text{-ThO}_2$	Wet	38	27	4.4	1.97	Rectangular	Square
SB-4	26 w/o $\text{U}^{235}\text{O}_2\text{-ZrO}_2$	1 w/o $\text{U}^{233}\text{O}_2\text{-ThO}_2$	Wet	37	26	4.4	1.97	Rectangular	Square
SB-5	26 w/o $\text{U}^{235}\text{O}_2\text{-ZrO}_2$	ThO_2	Dry	107	76	0.46	0.21	Hexagonal	Triangular
SB-6	26 w/o $\text{U}^{235}\text{O}_2\text{-ZrO}_2$	ThO_2	Dry	110	78	0.46	0.21	Hexagonal	Triangular
SB-7	26 w/o $\text{U}^{235}\text{O}_2\text{-ZrO}_2$	1 w/o $\text{U}^{233}\text{O}_2\text{-ThO}_2$	Dry	110	78	0.46	0.21	Hexagonal	Triangular

* Atom ratio.

† Ratio of volume of water to the volume of uranium (or thorium) at 20° C.

Figure 6. Diagram of Fuel and Test Rods for Uranium-233 Experiments.

Figure 7. Schematic View (Top) of Eight Seed-and-Blanket Core Assemblies.

batch of 6.039 kg was used for the seed pellets. ~~An analysis of these batches is given in Appendix A. Appendix B gives the uranium analysis of the finished uranium 233 containing pellets. Similar analysis for the uranium 235 seed pellets is given in Appendix C.~~

The seed pellets were fabricated by cold pressing and sintering to a size of 0.210 ± 0.003 -inch diameter by approximately 0.250 inch in length. The pellets were 100% dimensionally inspected, and oversize pellets were ground to the specified size. Similarly, the blanket pellets were fabricated to a size of 0.489 ± 0.005 -inch diameter by approximately 0.6 inch in length. Pellet length measurements were not required since only the full stack length of pellets was important. The pellet nonperpendicularity and flatness of the end-faces were measured to be less than 0.004 inch and 0.003 inch, respectively, at an assurance level of 95 x 95%. All pellets were of high density. ~~The measured densities are summarized in Appendix D.~~

Chemical analyses were conducted for impurity levels. These were expressed in terms of boron equivalents where the conversion factors were determined for the expected neutron spectrum. For the uranium-233 solution, the total boron equivalence was less than 80 ppm. (For most elements, the value used in the total was the lowest level of detectability of the analysis.) The only significant impurity was found in the blanket elements. It was found that the thorium starting material was contaminated with from 1 to 5 ppm of gadolinium, where the value depended on the sampling plan used in the drum of starting material. Except for this gadolinium, all impurities were well within the limits established by the specifications.

D. Description of Cores

The assemblies studied were of the seed-and-blanket type composed of a central seed region surrounded by either a wet or dry outer blanket. Overall core height was 20.83 inches, with an active fuel length of 15 inches. All cores were approximately clean critical, with an excess reactivity of only a few tenths of one percent. The eight assemblies constructed are shown schematically in Figure 7 ~~and are shown in greater detail in Figures 16 through 23. Table 3 gives a detailed description of the lattice.~~

Four B-10 stainless steel blade-type control rods were located in the seeds for shutdown and control. When inserted, these blades fitted between the seed fuel rods and were guided only by the upper core structure grid. During operations the blades were almost fully withdrawn and caused no perturbation in the lattice.

The top row of assemblies in Figure 7 (SB-1, SB-2, SB-2.5, SB-3, and SB-4) are rectangular and except for SB-2.5, which is a bare seed, have a wet blanket. ~~The mechanical arrangement for these cores is shown in Figure 8.~~ The core rested on a 3/4-inch stainless steel base plate, which in turn rested on a 1-inch stainless steel core support structure. The lattice spacing and seed-and-blanket geometry were maintained by the upper and lower fuel rod spacer plates of 1/4 inch stainless steel. An intermediate support plate, made by welding 0.02 x 0.100 x 11-inch zirconium rectangular wire into a 11 x 11-inch grid, was used to reduce any fuel rod bowing. ~~The design of the upper spacer plate is presented in Figure 9.~~ Fuel rods were placed on a square array, and this design permitted the interchange of four seed rods with one blanket rod. The seed sizes were adjusted to obtain the required clean critical configuration.

The bottom row of assemblies (SB-5, SB-6, and SB-7) are hexagonal in shape with a dry outer blanket and fuel rods in a triangular array. The blanket fuel rods were spaced with a 0.001-inch nominal clearance between rods. ~~A view of this arrangement is given in Figures 10 and 11.~~ The design provided for the interchange of one seed fuel rod with one blanket rod. This was done by placing plastic rings which had an outer diameter equal to the blanket rod diameter at the top, bottom, and center of each seed rod. With this arrangement, the seed sizes were adjusted such that the cores were clean critical.

VII. HIGH-TEMPERATURE TEST FACILITY EXPERIMENTS

A. Introduction

Critical experiments were performed at 480° F and 650 psi on two wet blanket assemblies in the Bettis High-Temperature Test Facility. The first assembly was a hot version of the SB-3 core, and the second similar to the SB-4 core. Fuel loadings of both cores were increased in the hot versions to account for the negative temperature coefficient and allow the cores to be critical at 480° F.

Measurements similar to those made on the cold cores were performed hot and compared with analytical model calculations. The information obtained serves as a check of the ability of the model to calculate the physics characteristics of these systems in the operating temperature range.

1. ^{233}U -Th Seed-Blanket Critical Experiments, S. Milani (W-BAPL), Invited (a)

Critical experiments utilizing ^{233}U and Th fuels have been conducted at the Bettis Atomic Power Laboratory since 1964. The critical assemblies were of the seed-and-blanket type and the experiments were performed in support of the Light-Water Breeder program. Two sets of experiments have been conducted to date. The first set consisted of 15-in. high assemblies which contained 26 wt% UO_2 - ZrO_2 seed fuel in which the uranium was either ^{233}U or ^{235}U , and blanket fuel of ThO_2 or 1 wt% $^{233}\text{UO}_2$ - ThO_2 . All elements were rod type with Zr clad. The second set consisted of 12 wt% $^{233}\text{UO}_2$ - ThO_2 seed fuel 28-in. high, and ThO_2 blanket fuel. Future experiments are planned in this second set with lower concentrations of ^{233}U in the seed fuel and with blanket fuel that contains ^{233}U .

Some results from the first set of experiments have been reported previously¹⁻⁴ along with a complete description of the fuel and assemblies. In addition to the reported experiments which were conducted at 68°F, two of these assemblies were studied at 480°F in the Bettis High Temperature Test Facility. Each assembly contained a blanket of 1 wt% $^{233}\text{UO}_2$ - ThO_2 and the seed for one contained $^{233}\text{UO}_2$ - ZrO_2 and for the second, contained $^{233}\text{UO}_2$ - ZrO_2 . Essentially the same agreement between experiment and analysis was found for the hot experiments as was found for the cold. For example, the critical eigenvalues as calculated by four-group, two-dimensional diffusion-theory calculations using the experimentally determined axial bucklings were within 0.4% of similarly calculated cold eigenvalues for the two assemblies.¹ Also, all the experiments of this set reported previously were of the "clean" type; i.e., very little structural material other than the fuel rods existed in the active region of the core. An additional "design"-type experiment was performed in which Zr structural material was included. This assembly had a ^{233}U bearing seed, an inner blanket bearing ^{233}U , and an outer blanket of ThO_2 rods. A converged source RECAP⁵ Monte Carlo calculation in which each fuel rod was described explicitly in three dimensions was performed. The critical eigenvalue from this problem was in close agreement with experiment.

The first assembly in the second set was a slab-type seed-and-blanket assembly. The seed had critical dimensions of $5.6 \times 22 \times 28$ in. and consisted of 0.25-in. Zr-clad rods that contained 12 wt% $^{233}\text{UO}_2$ - ThO_2 fuel pellets. The seed pitch was 0.310-in. triangular and the H/U = 30. The blanket reflected the seed on two sides and consisted of 0.625-in. Zr-clad ThO_2 rods. It had a height of 42 in. and extended above and below the seed. Each section of the blanket was 24-in. long and 10.9-in. thick. The assembly was clean critical in that little structural material existed in the fuel region and the control rods that were Hf blade-type rods were almost fully withdrawn at critical. The measured rods-out eigenvalue was 1.003. A preliminary calculated eigenvalue using a two-dimensional diffusion-theory calculation similar to those of the first set gave an eigenvalue of 0.986. This is approximately 2% lower than the eigenvalues obtained from the first set of experiments.

Experiments in the future will utilize lower weight percents of UO_2 - ThO_2 in the seed and ^{233}U -bearing blankets. Emphasis will be placed on assemblies that mockup the LWB design core.

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I. INTRODUCTION

Relative reaction rates were measured and calculated in ThO_2 blanket regions as a function of distance from seed regions in three one-dimensional seed-blanket slab cores. These cores contained ThO_2 blankets with H/Th atom ratios of 0.51, 1.55, and 5.1 to cover a wide range of blanket rod spacings of interest in power reactor design. The blanket regions were relatively narrow so that lattice parameter measurements were made in a non-asymptotic neutron spectrum. The ability to predict lattice parameters in such an environment provides a stringent test on the reliability of the calculational model and the nuclear constants used.

The parameters which are reported here are: (1) ρ^{02} , the ratio of episcadmium to subcadmium captures in blanket thorium, (2) the ratio of captures in thorium to fissions in U-235 in the blanket, (3) δ^{25} , the ratio of episcadmium to subcadmium fissions in U-235 in the blanket, (4) δ^{28} , the ratio of fissions in U-238 to fissions in U-235 in the blanket, (5) the spatial distribution of episcadmium thorium captures in a ThO_2 pellet for an H/Th = 5.1 lattice, (6) one-dimensional core initial modified conversion ratio, the ratio of thorium captures in the blanket to U-235 fissions in the seed, (7) ratio of episcadmium absorptions in Mn, In, and W to total fissions in U-235, and (8) cadmium ratio of U-235 in the seed. Measurements 1, 5, 6, and 8 above deal with a direct measurement of important neutron reactions occurring in the core. Measurements 2, 3, 4, and 7 involve the use of various detecting foil materials in the natural thorium blanket and were performed as means for comparing calculated and measured detailed spectra in the blanket. The experimental techniques used to measure these parameters are described.

The experimental results of the measurements are also compared with calculated values. The calculations were done using one-dimensional few group diffusion theory and multigroup transport theory codes. The multigroup calculation was done using the code P3MG employing 55 energy groups with a one-group, double P-1 thermal treatment and a P-3 approximation in the fast groups.

The overall agreement between calculation and experiment is considered good, particularly in the directly-measured, important neutron reactions in the core. There appears to be some discrepancy, however, in the values of δ^{28} which increase with increasing hydrogen-to-thorium atom ratio and in δ^{25} .

BMU SERIES OF ^{233}U FUELED CRITICAL EXPERIMENTS

Eight ^{233}U fueled critical assemblies were used to investigate fuel zoning, coupling between modules, the temperature effect, and power flattening for the LWBR program. Reference [27] emphasizes presentation of the experiments with only a few calculations included. Reference [1] reproduces about 40% of the report.

Measurements in the BMU series included core activation shapes, region power fractions, ratio of epithermal to thermal ^{233}U fission, ratio of epithermal to thermal thorium neutron capture, reactivity worth of eight structural materials, importance of neutrons from ^{252}Cf , measurement with different rod support grids, activation shapes within fuel pellets, effect of stainless steel plates on thorium capture, and effect of an axial fuel rod gap. The BMU lattices were constructed with both axial and radial fuel heterogeneity. The cores BMU-1A, BMU-1B, and BMU-1C were built to investigate zoning and operation at high temperature. BMU-2-1 was a single module version used as the reference core for investigating coupling in multiple module cores and the effects of rod support grids with relatively large metal volumes. BMU-2-3 was a three-module core in a triangular array, while BMU-2-4U was a four-module core arranged in a straight line. BMU-2-4PF was designed to investigate power flattening by preferential fuel loading in the outermost of the four modules. BMU-2-4T used a tilted loading to introduce a cross-gradient across the core.

^{233}U OXIDE-THORIUM OXIDE DETAILED CELL CRITICAL EXPERIMENTS

A critical assembly called the detail cell was constructed for the LWBR program to provide basic experimental data and to confirm nuclear analysis methods. Extensive three-dimensional power distribution measurements were performed with different seed and blanket configurations. A unique feature of the critical mock-up was the capability to move the entire seed assembly remotely. The final report, reference [28], is about 50% reproduced in reference [1].

Measurements in the detail cell included reactivity coefficient, power distribution, ^{232}Th fission distribution, ratio of epithermal to thermal ^{233}U fission, ratio of epithermal to thermal ^{232}Th neutron capture, ratio of ^{232}Th fission to total ^{233}U fission, activation distribution across individual fuel pellets, and grid flux depression. The detail cell was constructed with both axial and radial fuel heterogeneity, and was designed to produce information directly applicable to licensing a LWBR.

BNL $^{233}\text{UO}_2$ - ThO_2 FUELED EXPONENTIAL EXPERIMENTS

An extended series of 21 uniform subcritical lattices were constructed of 3% $^{233}\text{UO}_2$ - 97% ThO_2 fuel rods in light water moderator. Eight different hexagonal lattice pitches were used, each with pure H_2O moderator, plus additional cases with boric acid in the moderator. Flux measurements were used to determine material buckling, reflector savings, dysprosium disadvantage factor and the ratio of epithermal to subthermal capture in thorium in the unpoisoned cores; or material buckling and reflector savings in the poisoned cores. The stated objective of BNL in performing these experiments in about 1964 was to provide a data set covering a wide range of lattice parameters for use in evaluating calculational methods.

The main report describing the subcritical and neutronics experiments is reference [29], published in November 1970. Since it should be readily available, only three tables are reproduced here which give the fuel pin composition and measured results for the light water moderated cases. More detail on each of the experimental measurements is available as reference [30]. The series of experiments are also discussed in the annual report for 1965 [31], but only preliminary analysis values are given.

Three papers were presented by BNL personnel on the exponential experiments in the mid-1960's. Reference [32] presents a preliminary analysis, primarily of reflector savings. Reference [33] gives the full set of measured and calculated parameters for the 21 unpoisoned lattices. Finally, reference [34] discusses the poor agreement between calculations and measurements, and comments that, hopefully, ENDF/B cross sections will give better results.

In November 1974 seventeen thermal reactor benchmarks were specified in reference [35] for use by the Cross Section Evaluation Working Group, but none contained ^{233}U or thorium. In a supplement dated October 12, 1976 six more benchmarks were added, including three unpoisoned BNL ThO_2 lattices at moderator to fuel volume ratios of 0.997, 1.384 and 3.0043. As examples of their use in checking ENDF/B-IV cross sections, reference [36] compares both k_{eff} and the ratio of epithermal to thermal captures in ^{232}Th , and reference [37] compares calculated with derived k_{eff} values.

References [29-34] present the full range of BNL exponential measurements of neutronic results, analysis, calculations, and comparisons that have been published. However, it is expected that additional reports will appear which discuss BNL exponential lattices for thermal reactor benchmark testing. Hence references [36-37] should be considered only a sample of the current literature in this area.

TABLE I (a)

 $^{233}\text{UO}_2\text{-ThO}_2$ Fuel Pin Composition

Average Density = 8.9618 g/cm^3 ^a;
 diameter = $0.430 \text{ in.} = 1.09220 \text{ cm}$
 Active length = $42.5 \text{ in.} = 107.98 \text{ cm}$
 $(^{233}\text{U}/^{233}\text{U} + \text{Th}) \text{ wt.} = 0.03000$

Isotope	Weight Fraction	Atoms/cm ³
^{16}O	0.121197	0.408832×10^{23}
^{232}Th	0.851764	0.198115
^{233}U	0.026343	0.0061021
^{234}U	0.000394	0.0000909
^{235}U	0.000012	0.0000027
^{238}U	0.000289	0.0000655
B	0.0000044	0.0000220
Clad Composition: Zircaloy-2 density = 6.8365 g/cm^3 o.d. = $1.26746 \text{ cm} = 0.499 \text{ in.}$ i.d. = $1.09220 \text{ cm} = 0.430 \text{ in.}$		
Isotope	Weight Fraction	Atoms/cm ³
Zr	0.9827	0.44355×10^{23}
Sn	0.0145	0.005030
Cr	0.0010	0.000792
Fe	0.0013	0.000958
Ni	0.0005	0.000351
Co	5×10^{-6}	0.0000035
B	1.4×10^{-6}	0.0000053
Cd	0.5×10^{-6}	0.0000002
Hf	3×10^{-6}	0.0000007

^aThe fuel was prepared by the sol gel process and consists of compacted particles of a solid solution of UO_2 in ThO_2 . The compacted particles are of three size grades, 60 wt% 6/16 mesh, 15 wt% 50/140 mesh, and 25 wt% -200 mesh.

TABLE II (a)

$^{233}\text{UO}_2\text{-ThO}_2\text{-H}_2\text{O}$ Lattices
 Buckling and Reflector Savings

Moderator Fuel Volume Ratio	H_3BO_3 ^a Conc (mg/cm ³)	"Best" Values ^b of Measured B_m^2 (m ⁻²)	"Best" Values of Measured λ (cm)
0.997	0.0	75.88 ± 2.0	7.60 ± 0.3
	13.3	42.1 ± 0.5	5.30 ± 0.10
	25.0	18.8 ± 0.5	5.10 ± 0.10
1.384	0.0	86.06 ± 1.3	7.24 ± 0.17
	7.82	50.6 ± 0.5	5.60 ± 0.10
	22.9	3.65 ± 0.5	5.10 ± 0.10
1.7134	0.0	89.34 ± 2.0	7.05 ± 0.3
	4.74	60.4 ± 0.5	5.70 ± 0.10
	18.88	-1.8 ± 0.2	5.25 ± 0.15
2.1943	0.0	90.35 ± 1.6	6.80 ± 0.2
	4.71	50.5 ± 0.2	5.75 ± 0.05
	13.3	-2.1 ± 1.0	5.50 ± 0.10
3.0043	0.0	85.54 ± 0.8	6.57 ± 0.1
	4.99	26.5 ± 0.3	5.85 ± 0.20
	7.82	1.99 ± 0.2	5.60 ± 0.15
4.2722	0.0	69.8 ± 1.0	6.44 ± 0.15
	2.68	22.6 ± 0.4	6.0 ± 0.15
	4.96	-7.67 ± 0.2	6.0 ± 0.05
6.8449	0.0	32.2 ± 0.2	6.25 ± 0.05
	1.7	-10.8 ± 1.0	6.40 ± 0.20
9.2747	0.0	-1.22 ± 0.3	6.65 ± 0.05

^aSolutions at normal room temperatures ($\sim 20^\circ\text{C}$).
 (For densities see: The International Critical Tables.)

^bAs measured values are compiled by Price.

TABLE IV (a)

Microscopic Parameters for
 $^{233}\text{UO}_2\text{-ThO}_2\text{-H}_2\text{O}$ Lattices

Moderator Fuel Volume Ratio	H_3BO_3 Conc (mg/cm ³)	Measured ξ_{Dy}	Measured ρ_c^2
0.997	0.0	1.219 ± 0.024	1.380 ± 0.042
1.384	0.0	1.257 ± 0.024	0.928 ± 0.038
1.7134	0.0	1.270 ± 0.024	0.754 ± 0.024
2.1943	0.0	1.302 ± 0.024	0.607 ± 0.026
3.0043	0.0	1.325 ± 0.024	0.435 ± 0.013
4.2722	0.0	1.379 ± 0.024	-
6.8449	0.0	1.430 ± 0.034	0.218 ± 0.008
9.2747	0.0	1.447 ± 0.034	0.170 ± 0.007

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 Nucl. Sci. Eng., 42, 150-161, (November
 1970).

HOMOGENEOUS CRITICAL EXPERIMENTS

Very little has been published that deals with the neutronics aspects of homogeneous critical experiments. Instead, criticality measurements are emphasized almost exclusively and are well summarized in reference [1].

One example of limited neutronics measurements is given in reference [38] which describes thermal neutron flux distributions measured in radial and axial directions through a critical cylinder. Manganese foils were attached to a special carrier which was inserted into the $^{233}\text{UO}_2\text{F}_2\text{-H}_2\text{O}$ solution. The neutron flux is published in the reference as a function of position within the fissile solution or radial water reflector. Additionally, temperature coefficient measurements were made from 15°C to 90°C in a slightly sub-critical system, using a BF_3 counter as the detector.

Gwin and Magnuson measured the thermal value of η and other nuclear properties of ^{233}U and ^{235}U in critical aqueous solutions of uranyl nitrate. The original report is reference [39], whereas the readily available final report [40] gives some slightly revised values for the measured parameters. Spatial neutron flux distributions were measured with ^{238}U , ^{235}U , and cadmium-covered ^{235}U miniature counters. From this, the buckling could be obtained plus an extrapolation distance. The kinetic behavior was measured with the stable periods resulting from incremental additions of solution to a critical cylinder. From this, nonleakage probabilities could be determined as well as some information on the neutron age. Finally, neutron energy distributions were measured using fission foils and fission counting rates, each with and without cadmium covers so that corrections could be applied for the epithermal fissions and absorptions in integral form.

An extensive analysis of the Gwin and Magnuson experiments is available as reference [41]. Much of the paper deals with corrections to the measured k_{eff} , but this is used to draw conclusions about cross sections, neutron leakage, and the ^{233}U fission neutron spectrum.

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This report is a description of sources of neutronics data, and serves as a convenient cataloging of what already exists plus some indication as to its content and usefulness in further work. The author is also indebted to the companies, laboratories, and individual authors who created the original source information that is referenced and reproduced herein.

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