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The Data of Nuclear Reactor Physics: A Bibliography

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Compiled by

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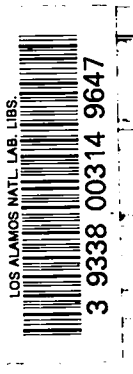

THE DATA OF NUCLEAR REACTOR PHYSICS: A BIBLIOGRAPHY

INTRODUCTION

The analysis of nuclear power reactors depends upon a large variety of nuclear (especially neutron) physics studies, mathematical studies, and the dissection of integral reactor critical experiment assemblies by physical and mathematical means. Until this time, the foundations of reactor physics and engineering has rested reasonably firmly upon direct experimental confirmation of characteristics deemed desirable in the performance of a nuclear power supply.

Because of the increasingly extensive scope and multiplying sources of reactor physics data - which consists of the three general fields of critical experiments, neutron cross-section tabulations, and the confluence of these two in reactivity calculations - it has seemed desirable to extract from the literature those publications that appear to be useful. Therefore a search of Nuclear Science Abstracts from 1956 to February 28, 1967, has been made with the assistance of Carroll B. Mills, and the more clearly informative abstracts collected. A summary of work prior to 1956 may be found in the volumes of the Proceedings of the First International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955.

These abstracts are grouped into the following sections and then are arranged chronologically by year as they appeared in NSA:

Critical experiments, reasonably homogeneous
Critical experiments, lattices
Neutron group averaged cross sections
Reactivity measurements for component materials
Doppler effects on reactivity

There is a certain amount of overlapping so that, for example, values of eta and alpha, from which cross sections may be constructed or checked will appear with cross sections. Also, a great majority of the experiments have been crude prototypes of reactors expected to produce power and so are hopelessly complex or poorly described. Many of these are included. Nevertheless this literature search is expected to ease considerably the search for a particular kind of data, a reactor, or a material, and in any case should encourage those who are the ultimate sources of essential data to publish it in a useful manner.

As a background for general reactor studies, the several sources of the information required before a study of reactor physics (the neutronics of nuclear power) will be listed:

1. The prompt fission number and distribution in energy has been reported by Terrell:

> J. Terrell, "Prompt Neutrons from Fission" from Physics and Chemistry of Fission, Vol. 2, Pg. 3-24, IAEA, 1965

and by Bernard et al.:

> E. Bernard, A. T. G. Ferguson, W. R. McMurray, and I. J. Van Heerden, "Time of Flight Measurements of Neutron Spectra from the Fission of U^{235}, U^{238}, and Pu^{239}," Nucl. Phys. 71, 228-240 (1965).

2. The basic summary tables of neutron cross sections are in the well-known "wallpaper" books published by the Sigma Center of Brookhaven National Laboratory, "Neutron Cross Sections" by M. D. Goldberg et al., BNL-325, 2nd Ed., Suppl. 2, 1964-1966, and "Angular Distributions in Neutron-Induced Reactions" by M. D. Goldberg et al., BNL-400, 2nd Ed., Vol. II, 1962.

An extensive "Calculations of Neutron Cross Sections Using a Local Optical Potential with Average Parameters" has been published as LA-3538-MS (1966) by Ferne P. Agee and Louis Rosen.

Of more direct interest to reactor oriented scientists is the extensive compilation of neutron reactions on a Master Data Tape processed at LASL by Roger Lazarus, using integral reactor constant numerical means set down by C. Hansen, H. Sandmeier, and R. Lazarus (all at LASL). The larger sources of these data are Louis Rosen (LASL), R. J. Howerton (LRL), and L. Parker (AWRE).

3. Extensive summary tables of critical reactor assemblies have been formed by H. C. Paxton, J. T. Thomas, Dixon Callihan, and E. B. Johnson, with the title "Critical Dimensions of Systems Containing U^{235}, Pu^{239}, and U^{233}," TID-7028 (June 1964).

1. CRITICAL EXPERIMENTS

1955

1.
7927

WORK ON U^{235} AND Pu^{239} REACTORS WITH HYDROGEN MODERATORS CONDUCTED AT THE ACADEMY OF SCIENCES OF THE U.S.S.R. G. N. Flerov. p.170-84 in Meetings of the Division of Physical-Mathematical Sciences. Session of the Academy of Sciences of the U.S.S.R. on the Peaceful Use of Atomic Energy. July 1-5, 1955. Moscow, Publishing House of the Academy of Sciences of the U.S.S.R., 1955. 376p. (In Russian)

Results are presented of an experimental study of U^{235} and Pu^{239} hydrogen-moderated reactors. For reactors of cylindrical and spherical shape the critical-mass and critical-volume values were determined for different concentrations of fissionable materials (hydrogen to fissionable material concentration ratios from 2.6 to 600). In some experiments the fissionable material moderator systems under investigation were allowed to become supercritical with the power level rising up to 50 w. The influence of various reflectors (uranium, paraffin, iron, etc.) on the neutron multiplication factor in the reactor was studied. A method for the absolute measurement of the multiplication factor in reactors of different shapes has been developed. The neutron spatial distribution in the reactors was measured. Measurements of the neutron energy distribution in the range from 1 ev up to 345 ev were carried out for reactors with high concentrations of fissionable material. Estimates of the effective number of neutrons per capture, ν_{eff}, for Pu^{239} and U^{233} in intermediate-energy reactors were obtained by a comparative method. The experimental determinations of ν_{eff} in the intermediate neutron energy region show that the prospects of using Pu^{239} hydrogen-moderated reactors for breeding purposes are poor. (auth)

4093 AEC-tr-2435((Pt. 1) (p.133-40))
WORK ON U^{235} AND Pu^{239} REACTORS WITH HYDROGEN MODERATORS CONDUCTED AT THE ACADEMY OF SCIENCES OF THE USSR. G. N. Flerov. p.133-40 of CONFERENCE OF THE ACADEMY OF SCIENCES OF THE USSR ON THE PEACEFUL USES OF ATOMIC ENERGY, JULY 1-5, 1955. SESSION OF THE DIVISION OF PHYSICAL AND MATHEMATICAL SCIENCES. (Translation). 8p.

This paper was originally abstracted from the Russian and appeared in Nuclear Science Abstracts as NSA 9-7927.

1956

2.
9859 CF-54-2-165
Oak Ridge National Lab., Tenn.
CRITICAL CONCENTRATION OF U-235 IN CORE OF HRT. R. B. Briggs. Feb. 23, 1954. Decl. Feb. 8, 1956. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80 (mf OTS).

Data are summarized which show the effect of blanket thickness and composition on the critical concentration of U^{235} in the HRT core. (B.J.H.)

3.
4470 CF-1627
Clinton Labs., Oak Ridge, Tenn.
NEUTRON MULTIPLICATION IN A MASS OF URANIUM METAL. Final Report—Problem Assignment 307-X10P "Snell" Experiment. J. E. Brolley, F. J. Byerley, B. Feld, A. E. Olds, R. Schalettar, L. Slotin, and R. B. Stewart. Apr. 1, 1944. Decl. Dec. 12, 1955. 43p. (A-2489). $7.80 (ph OTS); $3.30(mf OTS).

Measurements were made of neutron multiplication in 35 tons of U metal, using the Clinton pile as a neutron source, a three-group approximation was used to derive the neutron distribution in the metal. Detailed descriptions of the experimental set-up and measurements are also given. (B.J.H.)

4.
3151 WAPD-128
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
CRITICAL EXPERIMENTS ON A HIGHLY ENRICHED HOMOGENEOUS REACTOR. J. R. Brown, B. H. Noordhoff, and W. O. Bateson. May 1955. Decl. Nov. 30, 1955. 55p.

Contract AT-11-1-GEN-14.

Critical experiments with a highly enriched homogeneous reactor have been performed in the range of $H:U^{235}$ atomic ratios from 1633 to 1776. The cylindrical core has a fixed diameter of 36 in. and heights between 23 and 51 in. An aqueous solution of $UO_2(NO_3)_2$ with a U^{235} enrichment greater than 90% was used. An 8-in. radial H_2O reflector was available. Measurements were made of critical heights and of the reactivity worth per inch at these heights as functions of the $H:U^{235}$ atomic ratio. U^{235} critical masses are listed. Additional measurements were made to determine the control rod worths, the temperature coefficient of reactivity and the effect of the radial reflector. (auth)

5.
5399 MonP-402
[Clinton Labs., Oak Ridge, Tenn.]
CRITICAL MASS AND NEUTRON DISTRIBUTION CALCULATIONS FOR THE H_2O MODERATED REACTOR WITH D_2O, H_2O, AND BE REFLECTORS. E. Greuling and B. Spinrad. Oct. 29, 1947. Decl. Jan. 11, 1956. 16p. Contract W-35-058-eng-71. $3.30(ph OTS); $2.40(mf OTS).

Theoretical critical mass calculations for five experimental piles are described and experimentally checked. Agreement within 5 to 10% was obtained for the calculated and experimental values, the calculated values tending to be higher. (D.E.B.)

6.
3230 MonP-357
Clinton Labs., Oak Ridge, Tenn.
CRITICAL EXPERIMENTS ON A SMALL REACTOR OF ENRICHED U^{235} WITH Al-H_2O MODERATOR, AND D_2O, Be AND H_2O REFLECTORS. M. M. Mann and A. B. Martin. Aug. 18, 1947. Decl. Sept. 20, 1955. 21p. Contract W-35-058-eng-71.

Complete descriptions and diagrams are given of the experimental arrangements. Resultant critical masses of U^{235} are tabulated, and neutron flux distributions in the critical assembly are shown. (B.J.H.)

7.
7309 AECD-4044
Oak Ridge National Lab., Tenn.
FURTHER CRITICAL EXPERIMENTS ON A SMALL REACTOR OF ENRICHED U-235 WITH AL-H_2O MODERATOR AND BERYLLIUM REFLECTOR. A. B. Martin and M. M. Mann. Aug. 17, 1948. Decl. with deletions Jan. 6, 1956. 16p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

The experimental facilities and operating procedures which have been used in the conduct of critical experiments to study the characteristics of small enriched reactors have been described in MonP-357. The purpose of the present report is to describe briefly certain improvements that have been made in the experimental facilities and to present the results of further measurements on critical masses and spatial neutron flux distributions in Be-reflected reactors having square and thin slab geometries. The critical mass of a reactor having the usual U-Al-H_2O core and no reflector was measured and preliminary experiments to determine the fast neutron flux in these reactors were carried out. (auth)

8.
11507
HOMOGENEOUS HEAVY WATER MODERATED CRITICAL ASSEMBLIES. PART 1. EXPERIMENTAL. Richard N. Olcott (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 1, 327-41(1956) Aug.

Ten critical assemblies of enriched uranyl fluoride heavy-water solutions have been studied. In six cases, heavy water reflectors surrounded solutions in which the atomic ratio of deuterium to uranium-235 varied from 34 to 430. The remaining four assemblies were without reflector and the deuterium to U^{235} ratio ranged from 230 to 2080. Activation rates within the systems were measured for the resonance detectors In, Au, Pd, and Mn and for the fission detectors U^{235}, U^{238}, Pu^{239}, and U^{233}. (auth)

9.
384
CRITICAL ASSEMBLIES AT LOS ALAMOS. H. C. Paxton (Los Alamos Scientific Lab., N. Mex.). Nucleonics 13, No. 10, 48-50(1955) Oct.

The remotely assembled critical assemblies of bare fissionable metals at Los Alamos are described. Delayed neutron studies with the critical assemblies are also discussed. (B.J.H.)

10.
6452 MonP-48
[Clinton Labs., Oak Ridge, Tenn.]
CRITICAL EXPERIMENTS ON FLUORINATED AND HYDROGENATED MIXTURES CONTAINING ENRICHED URANIUM. Final Report—Problem Assignment PX14-1. A. H. Snell. Nov. 30, 1945. Decl. Jan. 11, 1956. Contains Supplement: CORRECTION FOR INHOMOGENEITY. E. Greuling. May 31, 1946. 29p. Contract W-35-058-eng-71. $6.30(ph OTS); $3.00(mf OTS).

Experiments are described on the critical sizes of assemblies of fluorinated, hydrogenous mixtures containing 24% enriched uranium. A hydrogenous reflector was used. The critical masses of the uranium as a function of the amount of hydrogen in the mixture under the experimental conditions of density and effective molecular weight are tabulated. The effect of Cd and B shielding between the U mixture and the surrounding reflector was also studied, and an experiment was made on varying the shape from a cubical to an elongated geometry. An attempt is made to interpret the criticality results in terms of pure UF_6, leading to the result that at this isotopic concentration, and a density of 4.5, about 36 kg of UF_6 would be critical if lumped surrounded closely by hydrogenous material. (auth)

11.
3700 CF-51-7-106
Oak Ridge National Lab., Tenn.
CRITICALITY OF HRE WITH LOW ENRICHMENT, HIGH CONCENTRATION UO_2SO_4—WATER SOLUTION. L. H. Thacker and H. T. Williams. July 23, 1951. Decl. Dec. 2, 1955. 2p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

1957

12.
13147 K-126
Carbide and Carbon Chemicals Corp. K-25 Plant, Oak Ridge, Tenn.
CRITICAL MASS STUDIES. Part II. C. K. Beck, A. D. Callihan, and Raymond L. Murray. Jan. 23, 1948. Decl. Mar. 25, 1957. 48p. $0.40(OTS).

Using uranium enriched with 30% U^{235} fabricated with fluoroplastic into cubes having nuclear properties of UF_6, a study was made of factors affecting the mass assembled at criticality. The effect of intermixed hydrogen, type of reflector, density of U^{235}, homogeneity, and geometric shape are considered. (auth)

13.
13855 K-343
Carbide and Carbon Chemicals Corp. K-25 Plant, Oak Ridge, Tenn.
CRITICAL MASS STUDIES. PART III. C. K. Beck, Dixon Callihan, J. W. Morfitt, and Raymond L. Murray. Apr. 19, 1949. Decl. Jan. 30, 1957. 77p. Contract [W-7405-eng-26]. $0.45(OTS).

The conditions under which U^{235} (93.4%), contained in aqueous solutions of UO_2F_2, becomes critical in cylindrical aluminum and stainless steel reactors were examined. (L.M.T.)

14.
13938 KAPL-M-IB-13
Knolls Atomic Power Lab., Schenectady, N. Y.
PPA-13 RIGOR MORTIS. I. Bornstein. Oct. 9, 1953. Decl. Mar. 14, 1957. 5p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

PPA-13 (Preliminary Pile Assembly) is a critical assembly, or mock-up, of a reflector-covered intermediate reactor. Some errors were detected in the multigroup models that had been used to calculate the critical mass. Subsequently, the models were revised, the errors eliminated, and the critical mass recalculated. The recalculated critical mass is 32 kg, as compared to an experimental critical mass of 32.5 kg. The composition of the assembly is tabulated. (L.T.W.)

15.
11006 K-406
Carbide and Carbon Chemicals Corp. K-25 [Plant], Oak Ridge, Tenn.
CRITICAL MASS STUDIES. PART IV. Dixon Callihan, D. F. Cronin, J. K. Fox, R. L. Macklin, and J. W. Morfitt. Nov. 28, 1949. Decl. Mar. 15, 1957. 34p. Contract [W-7405-eng-26]. $0.30(OTS).

Some exploratory experiments are reported describing the conditions under which enriched uranium, contained in aqueous solutions of uranyl fluoride, becomes critical in two right cylindrical reactors having parallel axes. Values of the height of solution and the critical mass have been obtained for reactors with diameters ranging from 5 to 20 inches. Data were obtained at four chemical concentrations corresponding to $H:U^{235}$ atomic ratios varying between 30 and 330 and with reactor separations up to 50 cm. In some of the experiments, the reactors were submerged in a water bath while others were done with no reflector. The critical mass of an unenclosed two reactor system was found to depend upon the distance between the components when they were separated even as much as 50 cm, however, the mass in each was then more than 90% of that required to make it singly critical. Interposition of water between two reactors reduced their interaction due to the attenuation of the neutron flux by hydrogen. Two water enclosed reactors which could be made singly critical when approximately equilateral were found to be effectively isolated when separated 15 cm or more. Two which could not be made individually critical were isolated by a separation of a few centimeters. Those which were singly critical at heights large compared to their diameters showed apparent interaction at more than 20 cm spacing. The effect is attributed to the equivalence of a few interaction neutrons in the two component system and the relatively large quantity of uranium which must be placed at the end of long reactors to produce small increases in reactivity. The smallest mass accumulated at criticality in these experiments was 680 gm U^{235} contained in each of two water enclosed reactors ten inches in diameter with sides in contact, each filled to a height of 16.9 cm. The chemical concentration of the fuel corresponded to an $H:U^{235}$ atomic ratio of 329. (auth)

16.
11007 K-643
Carbide and Carbon Chemicals Div. K-25 [Plant], Oak Ridge, Tenn.
CRITICAL MASS STUDIES. PART V. Dixon Callihan, D. F. Cronin, J. K Fox, and J. W. Morfitt. June 30, 1950. Decl. Mar. 19, 1957. 57p. Contract W-7405-eng-26. $0.40(OTS).

An investigation has been made of the conditions under which aqueous solutions of enriched uranyl nitrate become critical in right cylindrical reactors. A comparison was made of the neutron reflectivity of stainless steel, bismuth subcarbonate both dry and as a water slurry, aqueous solutions of natural uranyl nitrate and phosphoric acid. The materials tested, with the exception of dry bismuth subcarbonate, were about as effective reflectors as water. Dry bismuth subcarbonate was considerably less effective. The free nitrogen content of the nitrate solution and the addition of phosphoric acid and metallic bismuth to the reactor core were among the variables studied. Nitric and

phosphoric acid when introduced into the core material were effectively mild poisons when compared with aqueous solutions at the same hydrogen to U^{235} ratio. Bismuth was introduced into the core as an array of aluminum clad bismuth rods. The critical mass of this array was only slightly less than that measured when the bismuth was replaced by a similar array of voids. (auth)

17.
10293 UCRL-4880
California. Univ., Livermore. Radiation Lab.
CRITICAL MASS MEASUREMENTS ON GRAPHITE U^{235} SYSTEMS. James E. Carothers. Apr. 11, 1957. 15p. Contract W-7405-eng-48. $3.30(ph OTS); $2.40(mf OTS).

Measurements have been made on pseudo-cylindrical graphite–enriched uranium unreflected systems. These measurements include both critical mass determinations and time-dependent measurements using a pulsed neutron source to drive the assemblies. (auth)

18.
1313 AERE-R/R-2051
Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.
CRITICAL ASSEMBLIES OF AQUEOUS URANYL FLUORIDE SOLUTIONS. PART I. EXPERIMENTAL TECHNIQUES AND RESULTS. W. G. Clarke, C. C. Horton, and M. F. Smith. Sept. 20, 1956. 37p.

Critical masses of U^{235} and U^{233} in the form of UO_2F_2 in aqueous solution have been measured in cylindrical geometry for a range of concentrations $H/U_{fissile}$ 250 to 850. A cylinder radius of 15.24 cm was used with an effectively infinite radial water reflector. Where possible estimates were made of the critical masses for the system unreflected and also for the system reflected but with the Cd shut off sheath between the core and reflector. Flux measurements were made through the system in axial and radial directions at all concentrations. The effect of change of temperature on the system over the range 15 to 90°C was observed for all concentrations. An assembly for each isotope in a very concentrated solution $H/U^{235} = 60$, $H/U^{233} = 32$ was also made in a 6.35 cm radius cylinder but was not brought to criticality. (auth)

19.
9865 ORNL-1726
Oak Ridge National Lab., Tenn.
CRITICAL MASS STUDIES. PART VII. AQUEOUS URANIUM SLURRIES. D. F. Cronin and Dixon Callihan. July 1, 1954. Decl. Mar. 2, 1957. 33p. Contract W-7405-eng-26. $0.30(OTS).

A series of experiments was performed with aqueous slurries of UO_3, enriched to 93% in U^{235}, as a preliminary study of the nuclear stability of critical and near critical suspensions. The slurries were contained in a 12" diameter cylinder and were mixed by a centrally located variable speed stirrer in a manner giving small surface disturbances. Critical masses were measured at concentrations ranging from 40 to 300 g U/L at several stirrer speeds. The dependence of the critical parameters of well mixed slurries on concentration is essentially the same as found earlier with solutions. The critical masses of imcompletely mixed slurries were measured to be 5% to 10% less than those of well mixed ones under otherwise similar conditions, an amount not strongly dependent on the chemical concentration. Neutron activity excursions of corresponding magnitude were observed in near critical quantities of the slurry following stopping the stirrer in a well mixed system and, in some instances, after starting agitation of one which had settled. Although the activity changes lagged the alterations of the stirrer speed, in most cases requiring a few tens of seconds to develop, the magnitudes of the excursions are sufficient to warrent consideration of this effect in the safety features of reactors designed to use slurry fuels. The critical mass of a $(UO_2)F_2$ solution in the same vessel increased monotonically as the stirrer speed was increased over the same range, the total change being about 3%, due in part at least, to some vortex formation at the higher speeds. It is concluded that the activity excursions in the slurry result from changes in the U distribution more complex than those which occur in settling alone. An analysis of the distribution, relating it to the neutron activity, has not been made so the results of the experiments are not completely interpretable. (auth)

20.
12211 AECD-4245
Los Alamos Scientific Lab., N. Mex.
CRITICAL MASSES OF MILDLY DILUTED U^{235} SYSTEMS DERIVED FROM REACTIVITY CONTRIBUTION DATA. Leon B. Engle. [1956]. Decl. June 6, 1956. 11p. Contract [W-7405-Eng-36].

Reactivity contribution data, obtained from U^{235} critical assemblies at Pajarito can be used to predict the effects of diluents on critical mass. The reactivity contribution data versus radius for diluents in a given assembly may be expressed as a single functional relationship between integral absorption and transport cross sections of the diluents. Using this relation, there can be obtained a general expression for the dependence of critical mass upon volume fraction and the effective absorption and transport cross sections of the diluent. (auth)

21.
13911 CF-56-4-29
Oak Ridge National Lab., Tenn.
URANIUM INVESTMENT IN A CIRCULATING FLUORIDE POWER-PRODUCING REACTOR. W. K. Ergen. Apr. 4, 1956. Decl. Mar. 26, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Combination of critical radius and critical concentration (in grams U^{235}/cm^2) are given for a bare reactor having UF_4 dissolved in $NaZrF_5$. (L.M.T.)

22.
8669 HW-25614
Hanford Works, Richland, Wash.
CRITICALITY CONDITIONS FOR 1.75 PER CENT EN-

RICHED URANIUM SLUGS. John O. Erkman. Sept. 11, 1952. Decl. Mar. 12, 1957. 4p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

23.
7810 CF-56-2-63
Oak Ridge National Lab., Tenn.
CRITICAL PARAMETERS OF A PROTON MODERATED AND PROTON REFLECTED SLAB OF U-235. J. K. Fox and L. W. Gilley. Feb. 7, 1956. Decl. Feb. 14, 1957. 5p. Contract [W-7405-eng-26]. $0.25(OTS).

Slab thickness, critical solution height, H_2O height, and critical volume and mass were determined for H to U^{235} ratios of 44.7 and 51.5 to 1. (D.E.B.)

24.
9185

CRITICAL MASSES OF ORALLOY ASSEMBLIES. Glen A. Graves and H. C. Paxton (Los Alamos Scientific Lab., N. Mex.). Nucleonics 15, No. 6, 90-2(1957) June.

A summary of empirical data on critical configurations of fissionable metals is presented in a form that emphasizes the influence of conditions commonly of concern in nuclear safety questions. (T.R.H.)

25.
11008 KS-219
Carbide and Carbon Chemicals Co. [K-25 Plant], Oak Ridge, Tenn.
MINIMUM CRITICAL MASS OF U-235 FOR SEVERAL VALUES OF THE U-235 ASSAY. R. Gwin and H. F. Henry. Aug. 17, 1951. Decl. Mar. 14, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

26.
3567

REACTOR PHYSICS-8. James F. Hill (Senior Lecturer, Reactor School, Harwell). Nuclear Power 2, 29-32(1957) Jan.

The critical dimensions of bare cylindrical, spherical, or cubical reactors were obtained. A critical reactor with a reflector on the sides, or ends only, or surrounded was analyzed mathematically. (F.S.)

27.
12571 LA-1732
Los Alamos Scientific Lab., N. Mex.
CRITICAL MASSES OF GRAPHITE-TAMPED HETEROGENEOUS Oy–GRAPHITE SYSTEMS. J. C. Hoogterp. May 1954. Decl. Apr. 2, 1957. 34p. Contract W-7405-eng-36. $6.30(ph OTS); $3.00(mf OTS).

Critical mass measurements on graphite-tamped, heterogeneous oralloy (U^{235})–graphite systems have been made as a safety guide for certain oralloy casting procedures. Various concentrations were obtained by alternately stacking 10.5 in. diameter oralloy and graphite plates. In the relation Oralloy critical mass = constant × (fraction of oralloy/in the core volume)$^{-n}$ values for the exponent, n, in the neighborhood of 0.70 were obtained. (auth)

28.
13166 LAMS-786
[Los Alamos Scientific Lab., N. Mex.]
SAFETY TESTS ON HAND STACKING OF U-235 CUBES. V. Josephson. Oct. 11, 1948. Decl. Apr. 4, 1957. 23p. Contract [W-7405-eng-36]. $4.80(ph OTS); $2.70(mf OTS).

Procedures and results for critical assembly tests made on Topsy, to determine safety limits of hand stacking cubes of enriched U^{235} in spherical geometries preparatory to making critical assemblies of U^{235} with normal U tamper, are given. Tests were made of pseudospheres and the critical masses and diameters of these configurations are included. (F.S.)

29.
13894 CF-53-1-294
Oak Ridge National Lab., Tenn.
PREDICTION OF CRITICALITY BEHAVIOR OF A SLURRY UPON SETTLING. P. R. Kasten. Jan. 26, 1953. Decl. Apr. 2, 1957. 20p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

An investigation has been made of the criticality behavior of a slurry reactor as a function of settling of the fuel particles. For a given reactor with an initial H/U^{235} ratio above a certain value, a super-critical condition can be attained as a result of slowing down of the stirrer motor. Complete stopping of the stirrer motor should lead to a less dangerous situation. The slope of the U^{235} critical mass (C.M.) vs H/U^{235} ratio curve appears important in determining the criticality behavior upon fuel settling, a super-critical condition being attainable when $d(C.M.)/d\ H/U^{235}$ is positive, zero, or slightly negative. For more negative values of $d(C.M.)/(d\ H/U^{235})$ slurry settling will result in a sub-critical assembly. (auth)

30.
12632 CF-54-8-221
Oak Ridge National Lab., Tenn.
A REPORT ON CRITICAL DIMENSIONS OF CYLINDERS. Richard C. Keen. Aug. 31, 1954. Decl. Apr. 1, 1957. 18p. Contract [W-7405-eng-26]. $0.30(OTS).

Curves showing the relationship between the critical heights and critical diameters of stainless steel cylinders having wall thicknesses of $^1/_{16}$ in. and containing aqueous solutions of UO_2F_2 and enriched U have been plotted for several values of H to U^{235} atomic ratios. A critical size nomograph is presented for simple geometries. If the dimensions of one critical shape are known, then the dimensions for a series of similar shapes using the same assay material may be determined from the nomograph. Criticality relationships are also considered for cylindrical annuli and cylinders with small axial poison rods. (M.P.G.)

Critical Experiments 31-37

31.
184 HW-43463
[Hanford Atomic Products Operation, Richland, Wash.].
NUCLEAR SAFETY OF RIGHT ELLIPTIC AND RIGHT ANNULAR CYLINDERS. Norman Ketzlach. June 1, 1956. 9p. $1.80(ph OTS); $1.80(mf OTS).

Comparisons are made of elliptical and annular cross-sectional areas (capacity parameters) of safe vessels so that vessel shape may be evaluated as one of the parameters in any design for separations plants. (F.S.)

32.
13831 K-1260
Oak Ridge Gaseous Diffusion Plant, Tenn.
CRITICALITY CALCULATIONS FOR HYDROGENOUS SYSTEMS. J. R. Knight. Nov. 25, 1955. Decl. Mar. 14, 1957. 20p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

A theoretical method of calculating the critical mass of unreflected hydrogenous systems containing U^{235}, based on a spherical harmonics expansion for hydrogen moderation and on age theory slowing down for all other materials, is developed and set up for machine computation on the IBM 607. The calculated critical masses for high assay systems agree with experimental values within 13% for hydrogen to U^{235} ratios from 40 to 750, and agree within 12% for a single calculation at 4.9% U^{235} assay and a hydrogen to U^{235} ratio of 145. The calculating method appears to yield results which are directly applicable in the evaluation of the nuclear safety of uranium processing equipment. (auth)

33.
13954 LA-1653
Los Alamos Scientific Lab., N. Mex.
NEUTRON DETECTOR TRAVERSES IN THE TOPSY AND GODIVA CRITICAL ASSEMBLIES. G. A. Linenberger, L. L. Lowry, J. A. Grundl, and R. N. Olcott. June 1953. Decl. Apr. 1, 1957. 24p. Contract W-7405-eng-36. $4.80(ph OTS); $2.70(mf OTS).

Neutron detector traverses of the untamped U^{235} (Godiva) and the U^{238}- or Ni-tamped U^{235} (Topsy) metal critical assemblies have been obtained by counting γ-activity of U^{236} and U^{238} fission products, fission fragments of Np^{237} in a spiral chamber, and β-activity of Au and S. At a few positions within the assemblies, and for the U^{238} fission spectrum, cross section ratios of a number of pairs of fissionable isotopes were determined by means of a comparison fission chamber. At the center of Godiva, $\sigma_f(25)/\sigma_f(28) = 6.2$, and this ratio for the Topsy $U^{235}-U^{238}$ assembly ranges from 6.8 at the center to 76 at a radius of $8\frac{1}{4}$". Results of radiochemical analyses for fission and other reaction products are listed. (auth)

34.
10294 WAPD-169
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
EXPERIMENTAL AND THEORETICAL STUDY OF CRITICAL SLABS. EFFECT OF DIFFERENT REFLECTOR ARRANGEMENTS AND COMPOSITIONS. C. E. McFarland and A. D. Voorhis. Apr. 1957. 104p. Contract AT-11-1-GEN-14. $0.55(OTS).

Data are presented from critical slab experiments with a thin core of enriched U^{235}, H_2O, and Zr which were surrounded by reflectors of different compositions and arrangements. The reflectors studied were composed of water, Zr-water, and Al-water. In some experiments the cores were bisected by water gaps. Each critical assembly was analyzed by one-dimensional, group diffusion theory using digital codes. The calculated neutron multiplications from a four-group scheme predict criticality in most of the experiments within 1%, and from a two-group scheme within 3%. Calculated neutron flux distributions are fitted with experimental traverses of subcadmium and epicadmium activation (of Mn). The accuracy of the calculations with regard to input data and method is discussed. (auth)

35.
12574 N-2281
Chicago. Univ. Metallurgical Lab.
THE CONTROL PROBLEM AND THE CRITICAL SIZE OF AN ENRICHED, BeO MODERATED PILE. A. V. Martin. May 15, 1946. Decl. Mar. 4, 1957. 8p. (MUC-RGS-AVM-5). $1.80(ph OTS); $1.80(mf OTS).

36.
13664 HW-17519
[Hanford Works, Richland, Wash.].
"25" CRITICALITY CONSULTATION WITH J. W. MORFITT, APRIL 3, 1950. E. T. Merrill and G. Sege. Apr. 4, 1950. Decl. Feb. 19, 1957. 5p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

The application of the critical mass of U^{235} experimental studies was discussed with reference to the application of control in the 25 Process. (F.S.)

37.
4093 LA-1159(Del.)
Los Alamos Scientific Lab., N. Mex.
ORALLOY HYDRIDE CRITICAL ASSEMBLIES. H. C. Paxton, J. D. Orndoff, and G. A. Linenberger. May 22, 1950. Decl. with deletions Jan. 24, 1957. 61p. Contract W-7405-eng-36. $10.80(ph OTS); $3.90(mf OTS).

Critical-mass determinations for pseudospheres of oralloy hydride composition (approximating UH_3) in 8 in.-thick Tu and Ni tampers and in the Tu tamper with Ni liner were made. The critical mass of a hydride cube in the thick Tu also is given. Data on weight and dimensional changes of hydride pieces during the period of use are included. The results of Rossi time-scale measurements on the hydride assemblies are presented. Values of alpha at delayed critical and its variation with mass in the neighborhood of delayed critical are given. Measurements on the activation of various detectors within the hydride assemblies are described. Results as a function of radial position are given for Au, for Au shielded by Au and by Cd, for S and for fission catchers with U^{235} and

U^{238}. Reactivity changes resulting from the introduction of foreign materials into the hydride assemblies are discussed. Apparent regularities with respect to Z and qualitative interpretations of variations with radius are pointed out. From data for various radial positions, changes in critical mass corresponding to small changes in composition and density are computed. (auth)

38.
11489 AECD-4243
Los Alamos Scientific Lab., N. Mex.
ESTIMATED CRITICAL MASSES OF DILUTED ORALLOY. Hugh C. Paxton. June 1956. Decl. Aug. 30, 1956. 4p. Contract W-7405-Eng-36. $1.80(ph OTS); $1.80(mf OTS).

39.
13953 LA-1614
Los Alamos Scientific Lab., N. Mex.
LADY GODIVA: AN UNREFLECTED URANIUM-235 CRITICAL ASSEMBLY. R. E. Peterson and G. A. Newby. Sept. 1953. Decl. Apr. 12, 1957. 49p. Contract W-7405-eng-36. $7.80(ph OTS); $3.30(mf OTS).

A spherical, unreflected U^{235} critical assembly has been in operation since August, 1951. A remotely-controlled mechanical system is used to assemble subcritical components of the sphere, and reactivity is adjusted with U^{235} control rods positioned in the sphere. In addition to investigations of the neutron spectrum of the assembly, observation of the changes of reactivity produced by inserting foreign materials into the assembly, and the determination of parameters such as the temperature coefficient of reactivity, studies have been made of the behavior of the assembly at reactivities above prompt critical. (auth)

40.
1322 GAT-189
Goodyear Atomic Corp., Portsmouth, Ohio.
CRITICAL GEOMETRIES FOR BARE CYLINDERS. J. A. Pond. July 20, 1956. 23p. Contract AT(33-2)-1. $0.25 (OTS).

A modified two-group treatment is used to study neutron transport in aqueous solutions of uranyl fluoride. For a bare reactor, the critical condition $k = \eta f U_t U_f = 1$ is investigated. This is tested on experimental data from critical mass studies. It is shown that an always-safe, unreflected cylinder diameter of 20 cm results from this argument as does an always-safe moderation, $H/X = 2270$. The derived safe radius for a cylinder of infinite length, $\bar{R}$, is developed as a function of moderation H/X. The derived critical radius for a bare sphere, R_s^{bc}, is expressed as a function of both U^{235} assay and moderation. The multiplication factor, k, is developed for a system of right circular cylinders in line. In each case, k is expressed in terms of $\bar{\Omega}_f$, the average fractional solid angle subtended by a cylinder on an adjacent cylinder. Data are presented for critical systems showing how nearly the condition k = 1 is satisfied by critical data taken from studies conducted by the Union Carbide Nuclear Company. (auth)

41.
12615 Y-533
Carbide and Carbon Chemicals Corp. Y-12 Plant, Oak Ridge, Tenn.
AN EMPIRICAL STUDY OF SOME CRITICAL MASS DATA. C. L. Schuske and J. W. Morfitt. Dec. 6, 1949. Decl. Mar. 7, 1957. 23p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

A simple empirical equation has been found which relates the critical height of a water-enclosed stainless steel reactor at a given moderation to its diameter. Certain empirical constants appearing in the relation have a simple physical interpretation which succeeds to a limited extent in bridging the gap between the experimental results with finite cylindrical reactors and Grueling's ("Theory of Water-Boiler," LA-399, September 27, 1945), theoretical treatment which is limited to infinite cylinders or slabs. The report also discusses certain comparisons between theoretical and experimental results. (auth)

42.
9838 MonP-454
Clinton Labs., Oak Ridge, Tenn.
CRITICALITY STUDIES ON ENRICHED URANIUM–HEAVY WATER SYSTEMS. A. H. Snell. Dec. 15, 1947. Addendum: Aug. 1950. Decl. Mar. 7, 1957. 74p. Contract W-35-058-71. $0.45(OTS).

Addendum issued by Oak Ridge National Lab., Tenn.

The critical masses for systems consisting of U^{235} dissolved in D_2O have been studied for mean concentrations of 2.58, 5.17, and 10.35 grams of U^{235} per liter. The values obtained were respectively 1323, 930, and 869 grams U^{235} in roughly cylindrical geometry surrounded with D_2O reflector on all sides. The effects of various holes and thimbles penetrating the lattice and the reflector are given. The temperature coefficients were found to be respectively -0.85×10^{-3}, -0.59×10^{-3}, and -0.47×10^{-3} in terms of $\delta\kappa/\kappa$ per degree centigrade over the range 20-80°C. The effectiveness of a single control rod was measured in terms of distributed poison in the reactor and in terms of grams of U^{235}. The efficiency of utilization of the leakage neutrons in a ring of thorium rods in the cylindrical part of the reflector was measured. Neutron distributions are given as obtained by foil activation in the various modifications of the reactor. (auth)

43.
11827 IDO-16172
Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
MISCELLANEOUS SAFEGUARD CALCULATIONS. J. W. Webster and B. J. Garrick. May 26, 1954. Decl. Jan. 17, 1957. 10p. Contract AT(10-1)-205. $1.80(ph OTS); $1.80 (mf OTS).

Critical mass measurements were made for cylinders of U^{235} where the cylinder was bare, H_2O reflected, H_2O reflected with a Cd interface, and bare with a concentric Cd shell. The method of two-group, spherical geometry was used. (D.E.B.)

44.

13107 LWS-24712

California Research and Development Co., Livermore, Calif.

CRITICAL MASS STUDY FOR CYLINDRICAL GEOMETRY AS A FUNCTION OF RADIUS TO HEIGHT RATIO. Robert A. Weir. Jan. 20, 1953. Decl. Mar. 13, 1957. 6p. (CRD-R-30). $1.80(ph OTS); $1.80(mf OTS).

The results are presented of a study on the manner of variation of the critical mass of Pu and the peak-to-average power production ratio in a bare cylindrical reactor core. The particular case considered was that of a Pu, U, Fe, and Pb composite. Results indicate that increasing the ratio of critical radius to critical height from 0.5 to 1.0 would entail an increase in critical mass of 22%. (B.J.H.)

45.

13164 LA-1579

Los Alamos Scientific Lab., N. Mex.

TOPSY, A REMOTELY CONTROLLED MACHINE FOR THE STUDY OF CRITICAL ASSEMBLIES. R. H. White. June, 1953. Decl. Apr. 1, 1957. 34p. Contract W-7405-eng-36. $6.30(ph OTS); $3.00(mf OTS).

The construction, operation, and typical uses of Topsy, the versatile, remotely controlled, critical assembly machine at Pajarito are described. Section 1 covers the mechanical design of the machine, and the hydraulic and electrical operation of the various components. Section 2 describes how Topsy is used for investigation reacting metal assemblies. Procedures for establishing a delayed critical configuration and operation at delayed critical are illustrated for the U^{235}–U system. Also included are brief descriptions of U^{235}–Ni, Pu–U, and low density and concentration assemblies that have been made on the machine. (auth)

46.

12703 ORNL-2332

Oak Ridge National Lab., Tenn.

A CRITICALITY STUDY OF THE THOREX PILOT PLANT. O. O. Yarbro. Sept. 17, 1957. 29p. Contract W-7405-eng-26. $0.25(OTS).

A recent criticality study has indicated that the Thorex Pilot Plant is critically safe for the processing of thorium irradiated to less than 10,000 grams of U^{233} per ton. As the size of equipment exceeds the geometrically safe size and the total uranium inventory exceeds the minimum critical mass, minor equipment modifications and changes in operational procedures were necessary to provide safety under adverse operating conditions. (auth)

47.

14016 Y-881(Del.)

Oak Ridge National Lab., Y-12 Area, Tenn.

A GRAPHITE MODERATED CRITICAL ASSEMBLY—CA-4. E. L. Zimmerman. Dec. 7, 1952. Decl. with deletions Mar. 6, 1957. 72p. Contract W-7405-eng-26. $12.30 (ph OTS); $4.50(mf OTS).

An essentially bare graphite moderated critical assembly having dimensions of 51.0" × 44.11" and a critical mass of 52.48 kg of U^{235} was constructed at the ORNL Critical Experiments facility. Using a value for the buckling, B^2, of 0.0018628 cm^{-2} and an extrapolation distance of 2 cm, a multigroup calculation for the assembly gave an effective multiplication of 0.9912. The self-shielding factor due to lumping of the fuel in 0.01 in. discs was measured and found to be 0.94 ± 0.04 compared to a calculated value 0.95. Values of 0.15 ev for the mean fission energy and 27.66% for the fraction of fissions in the thermal group were calculated. Core removal type control rods were calibrated by the "rod-drop" method and by the observation of stable reactor periods. The form of the control rod calibrations curves indicate a contribution due to neutron streaming in the void formed by withdrawing the rod as well as the expected cosine squared distribution. A comparison using a flat strip of cadmium as a poison rod, which left essentially no void when removed, showed good agreement with a cosine squared sensitivity curve. The loss in reactivity as a function of the separation of sections of the assembly showed a loss of multiplication of 0.00635 due to a gap 0.3 in. wide. Bare In and Cd covered In foils were exposed in various parts of the assembly to observe the flux distribution both macroscopically and microscopically. Power distributions were observed by means of Al catcher foils in contact with the U. Comparison between fission rates of bare, Cd covered, and Cd–In covered fuel discs gave values of 2.1 and 3.4 compared to a calculated average value was 2.98. Danger coefficients for Na, Fe, Ni, and Mo were calculated using the multigroup neutron spectrum and the known cross section data. (auth)

1958

48.

8092 YAEC-57

Westinghouse Electric Corp. Atomic Power Dept., Pittsburgh.

ANALYSIS OF EXPERIMENTAL DATA ON REACTIVITY OF PLUTONIUM-BEARING FUEL RODS. W. H. Arnold, Jr. Feb. 25, 1958. 19p. For Yankee Atomic Electric Co. Contract AT(30-3-222, Subcontract No. 1. $3.30(ph OTS); $2.40(mf OTS).

A series of experiments were performed to determine the criticality of depleted uranium fuel rods containing 1% Pu. The data were analyzed and the results compared with hand calculations. Hand calculations gave quite good agreement with experiment, thus lending a degree of confidence to the predictions of the reactivity effects of plutonium in the first core of the Yankee Atomic Electric Company reactor. (auth)

49.

6966 LA-618(Del.)
Los Alamos Scientific Lab., N. Mex.
CRITICAL MASSES OF ENRICHED URANIUM HYDRIDES AND SOME RELATED MEASUREMENTS. C. P. Baker and M. G. Holloway. Feb. 3, 1947. Decl. with deletions Feb. 24, 1958. 40p. Contract W-7405-eng-36. $7.80 (ph OTS); $3.30(mf OTS).

The critical masses of enriched U–H mixtures under various conditions was measured, using two assembly structures embodying safety devices. A table of critical masses of UH_{10} in various tampers is given. The neutron density distribution inside WC and BeO tampers were measured with several detectors. In addition, a measurement was made of $\overline{\nu\sigma_R}/\overline{\nu\sigma_f}$ for the UH_{10} spectrum of neutrons. The mean life of neutrons in a UH_{10} assembly tamped with BeO and WC was measured by means of the Rossi method. (auth)

50.

10073 LA-2142
Los Alamos Scientific Lab., N. Mex.
PRELIMINARY CRITICAL EXPERIMENTS ON A MOCK-UP OF THE LOS ALAMOS MOLTEN PLUTONIUM REACTOR. H. G. Barkmann, D. M. Holm, R. M. Kiehn, and R. E. Peterson. June 1957. 29p. Contract W-7405-eng-36. $1.00(OTS).

A series of measurements of critical masses has been made for fast-spectrum plutonium assemblies similar to proposed designs for the Los Alamos Molten Plutonium Reactor Experiment (LAMPRE I). The effectiveness of various systems of reactivity control has been determined. Fission rate distributions and spatial flux variations were obtained for comparison with values computed according to the S_n method. (auth)

51.

963 A-4716
Clinton Engineer Works, Oak Ridge, Tenn.
CRITICAL MASS STUDIES, PART I. Clifford K. Beck, Dixon Callihan, and Raymond L. Murray. June 10, 1947. Decl. Mar. 12, 1957. 44p. $0.35(OTS).

Using 1108 one-inch cubes of mock-up UF_6, containing 54 kg of U^{235} made from UF_4 (95% isotopic concentration of U^{235}) and $(CF_2)n$, and similar hydrogenous cubes of $(CH_2)n$, in various proportions and under various conditions of accumulation, 26 assemblies were built to criticality and four other assemblies were built which did not become critical. From these experiments, at least some information was obtained on the effect on critical mass of amount of intermixed hydrogen, uniformity of hydrogen-uranium mixing, density, geometry, shielding or reflector materials around the accumulation. Methods of preparation and properties of the materials used, description of experimental arrangements and procedures, and statement of data obtained, are given. A brief summary is given of the pertinent results obtained in the experiments. (auth)

52.

17715
NEUTRON MULTIPLICATION IN SMALL SPHERES OF FISSIONABLE MATERIAL. P. J. Bendt and R. E. Peterson (Los Alamos Scientific Lab., N. Mex.). J. Appl. Phys. 29, 1271-7(1958) Sept.

Central-source neutron multiplication measurements have been made on small spheres of U^{233}, U^{235}, U^{238}, and Pu^{239}. Measurements were made on four sizes of spheres, varying approximately from 1 in. to 2.5 in. in diameter. In the case of U^{233}, only one sphere, 1.245 in. in diameter, was available. The three neutron sources, mock-fission, Po–Li, and Po–Be, were 0.4-in. diameter spheres. They have widely different neutron spectra. Multiplications were measured with a flat-response long counter, and with U^{235}, U^{238}, and Np^{237} spiral fission chambers, under conditions such that the error introduced by room-scattered neutrons was negligible. Results of the multiplication measurements are given. The measurements have been analyzed to obtain the leakage spectra from the spheres in terms of three velocity groups. The Np^{237} and U^{238} fission chambers serve as threshold detectors defining the lower energy of the second and third velocity groups, respectively. (auth)

53.

4429 AECU-3605
Los Alamos Scientific Lab., N. Mex.
GRAPHITE-MODERATED, GRAPHITE-REFLECTED CRITICAL ASSEMBLIES. Cleo C. Byers. [1957?]. 12p. Contract [W-7405-eng-36]. $3.30(ph OTS); $2.40(mf OTS).

Graphite-moderated, graphite reflected critical assemblies have been set up in the LASL Honeycomb remotely controlled machine. Information has been obtained on the critical masses of systems having C/Oy ratios of 6650 and 4093. A third system at a smaller ratio is planned. The reactivity contribution of channels through the core and reflector was determined. (auth)

54.

12890 AECD-4265
Carbide and Carbon Chemicals Co. (K-25 Plant), Oak Ridge, Tenn.
U-235 CRITICAL MASS DEPENDENCE ON MODERATION. A. D. Callihan, Hugh F. Henry, and R. L. Macklin. Sept. 22, 1952. Decl. with deletions June 3, 1958. 6p. Contract [W-7405-eng-26]. (KS-315(Del.)). $1.80(ph OTS); $1.80(mf OTS).

Based on experimental data and on conservative theoretical considerations, an estimate has been made of the relation of the minimum U-235 critical mass to the hydrogen moderation for uranium material at the U-235 assays for which experimental information is available. These estimates have been used in indicating maximum safe amounts for conditions where definite moderation limitations can be established. (auth)

55.
8757
CRITICAL MASS MEASUREMENTS FOR THE MERLIN RESEARCH REACTOR. D. R. Chick, K. Firth, M. Kerridge, and A. J. Salmon (Associated Electrical Industries, Ltd., Aldermaston, Eng.). Nature 181. 1171-3(1958) Apr. 26.

The Merlin research reactor is a light water-moderated reactor using highly enriched uranium fuel. A subcritical facility was built and is described for the experimental determination of the critical masses of the reactor before the reactor is made critical. The critical mass measurements for startup of the reactor are tabulated. (J.S.R.)

56.
3200 AERE-RP/P-1810
Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.
APPROACH TO CRITICAL EXPERIMENTS WITH SLIGHTLY ENRICHED URANIUM IN NATURAL WATER. V. S. Crocker, W. G. Davey, and K. R. E. Smith. July 1957. 28p.

57.
6550 ORNL-2367
Oak Ridge National Lab., Tenn.
CRITICAL MASS STUDIES. PART IX. AQUEOUS U^{235} SOLUTIONS. J. K. Fox, L. W. Gilley, and D. Callihan. Mar. 4, 1958. 57p. Contract W-7405-eng-26. $1.75(OTS).

Experiments were performed to determine the conditions under which aqueous solutions of U enriched to 93.2% in the U^{235} isotope can be made critical. The solutions, which had H: U^{235} atomic ratios varying between 27.1 to 74.6, were contained both in water-reflected and unreflected Al or stainless steel cylinders with and without Cd wrappings. The experiments varied from the use of a single vessel to interacting arrays of seven vessels. (auth)

58.
11786
SUBCADMIUM IN-PILE ETAS AND ABSORPTION CROSS SECTIONS OF U^{233}, Pu^{239}, AND Pu^{241}. E. R. Gaerttner, M. E. Jones, D. E. McMillan, J. B. Sampson, and T. M. Snyder (General Electric Co., Schenectady, N. Y.). Nuclear Sci. and Eng. 3, 758-71(1958) June.

Measurements of two independent types were made of the reactivity effect in a thermal test reactor of samples of U^{235}, U^{233}, Pu^{239}, and Pu^{241}. From these measurements average subcadmium values of eta ($\bar{\eta}$) relative to $\bar{\eta}$ of U^{235} are obtained independently of other knowledge of the average absorption cross sections. Average absorption cross sections are also obtained from the measurements. Values of ($\bar{\eta}$) for U^{233}, Pu^{239}, and Pu^{241} are respectively 2.231 ± 0.034, 1.927 ± 0.024, and 2.213 ± 0.07. The corresponding value of η (0.0253 ev) of Pu^{239} is found to be 2.025. A presentation of the method and results are given together with a comparison with previous work. (auth)

59.
4471 WAPD-P-695
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
ANALYSIS OF DCTF CRITICALITY AND REACTIVITY DATA. Ely M. Gelbard. Dec. 1955. 18p. Contract AT-11-1-GEN-14. $3.30(ph OTS); $2.40(mf OTS).

DCTF (Danger Coefficient Test Facility) critical heights were measured for six different concentrations, both with and without a water reflector. In addition, reactivities per inch of solution height have been determined for the same concentrations. This report summarizes results of an investigation of the experimental data, an investigation which had two main objectives. First, it was hoped that available information would suffice to determine lucite and water reflector savings for the given fuel concentrations. Second, the moments of the slowing down density in water were to be deduced from the critical height and reactivity measurements, through analysis of the critical equation. The results detailed were inconclusive, but suggestive. (auth)

60.
13634 TID-5345
Carbide and Carbon Chemicals Co. [K-25 Plant], Oak Ridge, Tenn.
VARIATION OF CERTAIN CRITICAL PARAMETERS WITH ASSAY. L. Geller. Nov. 20, 1952. Decl. June 3, 1958. 9p. Contract [W-7405-eng-26]. (KS-336). $1.80(ph OTS); $1.80(mf OTS).

New data from the 5% critical experiments were used to determine the variation of certain minimum critical parameters with the U^{235} assay. Curves are presented showing the minimum mass, volume, and cylinder diameter of U^{235} as functions of the assay. (W.D.M.)

61.
15018 A/CONF.15/P/593
Oak Ridge National Lab., Tenn.
EXPERIMENTAL AND THEORETICAL STUDIES OF UNREFLECTED AQUEOUS U^{235} CRITICAL ASSEMBLIES. R. Gwin, D. K. Trubey, and A. M. Weinberg. 20p. $0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

The empirical kernel method for treating bare critical systems is discussed and is utilized to predict the material buckling of aqueous U^{235} bare reactors in an effort to clarify the age discrepancy. These results are compared with experimental critical and kinetic data which are also presented. In the comparison, it is found that the uncertainty regarding the proper experimental extrapolation distance arises and prevents an unambiguous comparison. (auth)

62.

15276 K-1380(Pt.F)
Oak Ridge National Lab., Tenn.
CRITICAL MASS DATA APPLICABLE TO NUCLEAR SAFETY PROBLEMS. R. Gwin and J. T. Thomas. Pt. F [of] STUDIES IN NUCLEAR SAFETY. Lectures Presented at the Nuclear Safety Training School Conducted by Union Carbide Nuclear Company, June 3-14, 1957. 19p. Contract W-7405-eng-26.

Representative data obtained at several laboratories are summarized graphically. These include homogeneous solutions, solids, effects of reflectors, and mention of preliminary information on interacting systems. (auth)

63.

686 HW-51168
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
PROGRESS REPORT ON EXPERIMENTS TO DETERMINE INFINITE MULTIPLICATION FACTORS OF ENRICHED UO_3-H_2O MIXTURES. Harry E. Handler. July 1, 1957. Decl. Aug. 26, 1957. 8p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Experiments to determine maximum U^{235} enrichment for UO_3-H_2O and $UO_2(NO_3)_2-H_2O$ mixtures which will be subcritical independent of the H/U ratio and the volume are described. (T.R.H.)

64.

562 LA-2141
Los Alamos Scientific Lab., N. Mex.
BERYLLIUM-REFLECTED, GRAPHITE-MODERATED CRITICAL ASSEMBLIES. G. E. Hansen, J. C. Hoogterp, J. D. Orndoff, H. C. Paxton, P. G. Koontz, W. H. Roach, and D. P. Wood. July 1957. 61p. Contract W-7405-eng-36. $1.75(OTS).

Included are data on the properties of three sets of cylindrical Be-reflected, graphite-moderated critical assemblies that have been set up in the Honeycomb machine. The first set was primarily to establish characteristics as functions of C/U^{235} atomic ratio of a core with nearly constant reflector thickness. Fission rates were mapped with U^{235} foils, and reactivity contributions of some foreign materials were measured. The second set consisted of three assemblies with fixed cores to determine the effect of redistributing reflector from the ends to the cylindrical wall. Flux distributions in the basic fully-reflected assembly were mapped extensively with bare and with Cd-shielded foils of U^{235}, Au, and In. The third set was to establish the minimum-volume core at $C/U^{235} \sim 350$ that could be made critical with available Be. Influences of end reflector and of a Be island were checked. (auth)

65.

4428 AECU-3604
Los Alamos Scientific Lab., N. Mex.
CRITICAL ASSEMBLIES OF GRAPHITE AND ENRICHED URANIUM WITH BERYLLIUM REFLECTORS. G. E. Hansen, J. C. Hoogterp, J. D. Orndoff, and N. C. Paxton. 20p. Contract [W-7405-eng-36]. $4.80(ph OTS); $2.70 (mf OTS).

Data are given on properties of three sets of cylindrical Be reflected, graphite-moderated critical assemblies. The first set was primarily to establish characteristics as functions of C/Oy atomic ratio of core with nearly constant reflector thickness. Fission distributions were determined. The second set consisted of three assemblies with fixed core to determine the effect of redistributing reflector from the ends to the cylindrical wall. This series was done to provide the Los Alamos Scientific Laboratory's Theoretical Division checks for two-dimensional diffusion code. Flux distributions in the uniformly-reflected assembly were mapped extensively with bare and Cd-shielded foils of oralloy, Au, and In. The third set was to establish the minimum-volume core at C/Oy ~ 350 that could be made critical with available Be. Experimental critical data converted to equivalent spherical systems are compared with results of S_4 calculations. (auth)

66.

12893 LA-2203
Los Alamos Scientific Lab., N. Mex.
CRITICAL MASSES OF ORALLOY IN THIN REFLECTORS. G. E. Hansen, H. C. Paxton, D. P. Wood, K. W. Gallup, and R. H. White. Jan. 1958. 33p. Contract W-7405-eng-36. $6.30(ph OTS); $3.00(mf OTS).

Critical masses were measured for 5.25 in. diameter Oy cylinders in 0.5 and 1 in. thick reflectors of Be, graphite, Mg, Al, Ti, mild steel, Cu, W alloy, Tu, Ni, Co, Mo, Al_2O_3, Mo_2C, and polythene. These results were converted to the equivalent spherical critical masses of Oy and compared to yield consistent transport cross sections for the reflector materials. In addition, critical masses of Oy spheres in ~2 and ~4 in. thick spherical reflectors of W alloy, Fe, Ni, Ni-silver, Cu, Zn, Th, Be, BeO, C, and Tu were determined. (auth)

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67.

15017 A/CONF.15/P/592
Los Alamos Scientific Lab., N. Mex.
PROPERTIES OF ELEMENTARY FAST-NEUTRON CRITICAL ASSEMBLIES. G. E. Hansen. 15p. $0.50 (OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

Typical properties of elementary fast-neutron critical assemblies studied at the Los Alamos Scientific Laboratory are tabulated. The kinetic, spectral, and perturbative properties of Godiva, Topsy, and their Pu analogs were studied in detail. Results of critical measurements are supplemented by data from uranium-metal exponential columns. The influences of shape and material composition on critical size are discussed. The properties of spherical U and Pu assemblies are covered. (M.H.R.)

68.

9745 K-1019(4th Rev.(Del.))
Oak Ridge Gaseous Diffusion Plant, Tenn.
BASIC CRITICAL MASS INFORMATION AND ITS APPLICATION TO OAK RIDGE GASEOUS DIFFUSION PLANT DESIGN AND OPERATION. H. F. Henry, A. J. Mallett, and C. E. Newlon. Aug. 2, 1957. Decl. with deletions Mar. 13, 1958. 48p. Contract W-7405-eng-26. $1.50(OTS).

The current minimum experimental values of the basic criticality control parameters for U^{235} assays between 2% and approximately 90% are presented together with the basic criticality control methods currently in effect at ORGDP. The fundamental nuclear safety criteria remain essentially unchanged from previous editions of the report with the exception of the neutron interaction specifications, which have been extended considerably, and the approval, for the first time, of the limited use of water in cascade fire control activities. A chart of the organization for nuclear safety control at ORGDP and a glossary are also included. (auth)

69.

11384 AECD-4266
Carbide and Carbon Chemicals Co. [K-25 Plant], Oak Ridge, Tenn.
VARIATION OF CRITICAL PARAMETERS BETWEEN U-235 ASSAYS OF 4.9 PERCENT AND 93.4 PERCENT. H. F. Henry and C. E. Newlon. Oct. 23, 1953. Decl. with deletions June 3, 1958. 14p. Contract W-7405-eng-26. (KS-399(Del.)). $3.30(ph OTS); $2.40(mf OTS).

The minimum critical mass, volume, and cylinder diameter for U of 4.9% U^{235} assay were determined as 1.83 kg, 32.1 liters, and 10.7 in., respectively. Curves have also been prepared by applying the same constant correction to each minimum parameter curve that was used in establishing "always-safe" values for 93.4% assay material. These reduction curves may serve as possible criteria for establishing "always-safe" values over a wide range of U^{235} assays. (W.L.H.)

70.

3898 LA-1155(Del.)
Los Alamos Scientific Lab., N. Mex.
ORALLOY SHAPE FACTOR MEASUREMENTS. V. Josephson, R. W. Paine, Jr., and L. L. Woodward. Aug. 8, 1950. Decl. with deletions Dec. 17, 1957. 35p. Contract W-7405-eng-36. $6.30(ph OTS); $3.00(mf OTS).

Measurements were made at the Pajarito remote control laboratory to determine the effect of change of shape on system reactivity for oralloy cylinders. Systems tested include cylindrical configurations with various height-to-diameter ratios ranging from slabs to rods. Each system reactivity is referred to that of a sphere in the same tamper. Reactivity tests were made on bare (untamped) Oy configurations, as well as on systems in tuballoy tampers 1.12, 1.87, and 8.0 in. thick. The amount of reactivity change associated with a particular cylinder height-to-diameter ratio is found to be a function of tamper thickness, and is greatest for very thin tampers. (auth)

71.

15585 HW-55707(Del.)
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
NUCLEAR SAFETY IN PLUTONIUM METAL DISSOLUTION. N. Ketzlach. Apr. 11, 1958. Decl. with deletions May 21, 1958. 17p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Experimental evidence as well as theoretical considerations are presented which indicate that a plutonium metal–plutonium solution system can be more reactive than either one alone. More experiments as well as further theoretical development are required to achieve a better understanding of such systems. (auth)

72

4770 UCRL-4937
California, Univ., Livermore, Radiation Lab.
SPHERICAL AND CYLINDRICAL PLUTONIUM CRITICAL MASSES. Fred A. Kloverstrom. Sept. 1957. 18p. Contract W-7405-eng-48. $3.30(ph OTS); $2.40(mf OTS).

Experiments to determine critical masses of δ-phase plutonium cylinders of three diameters with thin metallic reflectors are reported. Critical reflector thickness measurements were made with two spherical Pu cores; the cylindrical and spherical data are combined to yield shape factors for the spheres for U^{238} and Be reflection. (auth)

73.

15023 A/CONF.15/P/598
Argonne National Lab., Lemont, Ill.
FAST NEUTRON POWER REACTOR STUDIES WITH ZPR-III. J. K. Long, W. B. Loewenstein, C. E. Branyan, G. S. Brunson, F. S. Kirn, D. Okrent, R. E. Rice, and F. W. Thalgott. 44p. $0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

Argonne's critical facility, ZPR-III, is briefly described. More than 30 different assemblies of widely varying geometries and compositions were studied in this facility, most of which were to aid in the development of fast power breeder reactors. A description of the ZPR-III assemblies, tables of central reactivity coefficients, prompt neutron data, fission distributions, multigroup constants, critical masses, and other data from these experiments are presented. (M.H.R.)

74.

1911 LA-1305
Los Alamos Scientific Lab., N. Mex.
ORALLOY CYLINDRICAL SHAPE FACTOR AND CRITICAL MASS MEASUREMENTS IN GRAPHITE, PARAFFIN, AND WATER TAMPERS. E. C. Mallary. Oct. 27, 1951. Decl. Mar. 26, 1957. 24p. Contract W-7405-eng-36. $4.80(ph OTS); $2.70(mf OTS).

Critical mass of an Oy (93.9%) sphere in graphite is given as a function of graphite tamper thickness. It was shown that ~20 in. of graphite is effectively infinite and that the ~17 in. used herein is ~98% infinite. Critical mass measurements of Oy (93.9%) spheres and cylinders of different diameters are given for ~17 in. graphite tamper, for effectively infinite paraffin tamper, and for effectively infinite water tamper. From these measurements cylindrical shape factors were computed. Both shape factors and critical masses are shown as functions of the critical height to diameter ratio of the cylinders. (auth)

75.

3915 Y-1023
Carbide and Carbon Chemicals Co. Y-12 Plant, Oak Ridge, Tenn.
MINIMUM CRITICAL MASS AND UNIFORM THERMAL NEUTRON CORE FLUX IN AN EXPERIMENTAL REACTOR. J. W. Morfitt. Dec. 1, 1953. Decl. Sept. 12, 1957. 167p. Contract W-7405-eng-26. $4.50(OTS).

The relationship between minimum critical mass and uniform thermal neutron core flux was experimentally investigated in a water-moderated water-reflected nuclear reactor employing U^{235} as fuel. The reactor vessel was a right circular cylinder of Al, 72 cm in diameter and 91.4 cm long. The core, 30.2 cm in diameter, was divided into 5 concentric regions by Al partitions and was surrounded by an effectively infinite water reflector on its lateral surface only. Varying concentrations of aqueous uranyl fluoride solution made from U containing 93.2% of the U^{235} isotope were added to the several regions to simulate a theoretical fuel distribution having a continuous concentration gradient. The theoretical fuel distribution was that given by a calculation method developed in a theoretical treatment of the problem by Goertzel, who demonstrated mathematically that the condition of minimum critical mass in a suitably chosen thermal reactor required that the thermal neutron flux be uniform everywhere in the core. The experimental results clearly establish the validity of the Goertzel theory. The experimentally determined critical height and mass, using the theoretically determined fuel loading, were within 2.5% of the corresponding calculated parameters. The importance function for the fuel was shown to be constant throughout the core as required for a minimum critical mass reactor. The measured thermal and non-thermal components of the neutron flux were in good agreement with theory, and the thermal flux, except for deviations produced by the stepwise approximation to the ideal fuel distribution, was uniform along a radius of the core. The longitudinal neutron flux behaved as expected; a measure of the Cd fraction showed the reactor to be essentially thermal, and in general, data obtained from the experimental reactor were compatible with the postulates and predictions of the Goertzel theory. An exploratory investigation to obtain experimental verification of a modification of the theory made by Goertzel to predict how the critical mass of fuel can be minimized in a reactor of less than optimum radius met with little success. Although some lowering of the critical mass was produced by the theoretically determined fuel distribution, a discrepancy of more than 35% was found to exist between theory and experiment. (auth)

76.

17405 TID-2505(Del.)(p.107-25)
Carbide and Carbon Chemicals Co. Y-12 Plant, Oak Ridge, Tenn.
MINIMUM CRITICAL MASS AND UNIFORM THERMAL-NEUTRON CORE FLUX IN AN EXPERIMENTAL REACTOR. John W. Morfitt. p.107-25 [of] NUCLEAR SCIENCE AND TECHNOLOGY. (EXTRACTS FROM REACTOR SCIENCE AND TECHNOLOGY. VOL. 4, ISSUES 1 TO 4, MARCH-DECEMBER 1954). 19p.

The relation between minimum critical mass and uniform thermal-neutron core flux has been experimentally investigated in a water-moderated, water-reflected nuclear reactor employing U^{235} as fuel. The theoretical fuel distribution was that given by Goertzel, who demonstrated mathematically that the condition of minimum critical mass in a suitably chosen thermal reactor required that the thermal-neutron flux be uniform everywhere in the core. The results clearly establish the validity of the Goertzel theory. The experimentally determined critical height and mass, using the theoretically determined fuel loading, were within 2.5% of the corresponding calculated parameters. The importance function for the fuel was shown to be constant throughout the core as required for a minimum-critical-mass reactor. The measured thermal and non-thermal components of the neutron flux were in good agreement with

theory. The longitudinal neutron flux behaved as expected, and the reactor was found to be essentially thermal as demanded by the Goertzel theory. An exploratory investigation to obtain experimental verification of a modification of the theory made by Goertzel to predict how the critical mass of fuel can be minimized in a reactor of less than optimum radius met with little success. Although some lowering of the critical mass occurred, a discrepancy of more than 35% was found to exist between theory and experiment. (auth)

77.

2121 LA-2023

Los Alamos Scientific Lab., N. Mex.

PRELIMINARY SURVEY OF URANIUM METAL EXPONENTIAL COLUMNS. J. J. Neuer, C. B. Stewart, G. A. Jarvis, R. N. Olcott, D. S. Young, and J. E. Sattizahn. Jan. 1956. Decl. Apr. 2, 1957. 44p. Contract W-7405-eng-36. $7.80(ph OTS); $3.30(mf OTS).

Reflected cylindrical exponential columns of bare and natural U were constructed of stacked plates of oralloy (93.4%) and tuballoy. Average concentrations of U^{235} ranging from 0.72% to 9.18% were investigated. Flux distributions, buckling values, and spectral indices were obtained. The indicated U^{235} concentration for infinite critical mass is about 5.5%. (auth)

78.

1139 LA-1958(Del.)

Los Alamos Scientific Lab., N. Mex.

CRITICAL MASSES OF FISSIONABLE METALS AS BASIC NUCLEAR SAFETY DATA. H. C. Paxton and Glen A. Graves. Apr. 1956. Decl. Sept. 7, 1956. 18p. Contract W-7405-Eng-36. $0.30(OTS).

Data on critical configurations of fissionable metals are summarized in a form emphasizing the influence of conditions commonly of concern in nuclear safety questions. Although the bulk of the data is derived from experiments with uranium metal enriched, usually, to about 90 wt. % U^{235} there are enough data for Pu and U^{233} to establish some general relations about their criticality relative to U^{235}. The specific information includes the influences exerted upon critical mass by various reflectors, by shape of the fissionable material, by variations in U^{235} concentration, by variations in material density, and by graphite and hydrogenous diluents. (auth)

79.

15123 A/CONF.15/P/2408

California. Univ., Livermore. Radiation Lab.

CRITICAL MEASUREMENTS AND CALCULATIONS FOR ENRICHED-URANIUM GRAPHITE-MODERATED SYSTEMS. H. L. Reynolds. 27p. (UCRL-5175). $0.50 (OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

The experimental results on a variety of enriched-uranium, graphite-moderated systems, both with and without reflectors are presented. Also included are the results of the multi-group transport, and one- and two-dimensional diffusion calculations used to interpret the data. In general the University of California Radiation Laboratory systems were bare or one-dimensional in order to simplify the analysis. Efforts were made to obtain systems as close as possible to an idealized system containing only a homogeneous mixture of moderator and fuel with all extraneous factors removed. The Los Alamos Scientific Laboratory experiments were carried out with more complex geometries approaching more closely potential reactor systems. All of the assemblies utilized heterogeneous arrangements of moderator and thin uranium foils. The bare systems range in carbon-to-uranium atomic ratios from 300/1 to 2500/1. These systems are not truly thermal and are in a range where critical mass is extremely sensitive to size or buckling. Experiments were performed for these systems to determine accurately the effects of extraneous factors such as room return, control-rod void spaces, nonhomogeneity of fuel loading, moderator block porosities and poison content. The reflectors include graphite and beryllium in one-, two-, and three-dimensional arrangements. The constants used in the multigroup calculations are presented in tabular form. Eighteen energy groups were used. The same constants were used in the transport and diffusion calculations to allow comparison of the two methods of calculation. (auth)

80.

8226 UCRL-4975

California. Univ., Livermore. Radiation Lab.

CRITICAL MASSES OF SPHERICAL SYSTEMS OF ORALLOY REFLECTED IN BERYLLIUM. H. Robert Ralston. Oct. 10, 1957. Decl. Feb. 26, 1958. 7p. Contract W-7405-eng-48. $1.80(ph OTS); $1.80(mf OTS).

The critical thickness of beryllium reflector was determined for oralloy (uranium, enriched to 93.17% U^{235}) spheres ranging from 10.765 kg to 32.654 kg. Four points which were determined by other experimenters were normalized to the data and are included in the curves. (auth)

81.

12087 AECD-4264

Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.

ESTIMATES OF MINIMUM CRITICAL VOLUMES. R. C. Rohr and H. F. Henry. Feb. 15, 1952. Decl. with deletions June 3, 1958. 11p. Contract [W-7405-eng-26]. (KS-267(Del.)). $3.30(ph OTS); $2.40(mf OTS).

A study was made of the available high-assay criticality data for materials of 93.4% and 95.3% U^{235}, and minimum critical volumes were estimated for several conditions. The data, estimated for aluminum and stainless steel containers, water and air reflected, and with and without shielding, are tabulated. (J.S.R.)

82.

2176 Y-829

Carbide and Carbon Chemicals Co. Y-12 Plant, Oak Ridge, Tenn.

EMPIRICAL STUDIES OF CRITICAL MASS DATA. PART II C. L. Schuske and J. W. Morfitt. Dec. 5, 1951. Decl. Mar. 16, 1957. 26p. Contract W-7405-eng-26. $4.80 (ph OTS); $2.70(mf OTS).

In a previous report a method of graph analysis was developed for obtaining the critical dimensions of semi-infinite U^{235} water reactors from measurements made on finite systems. The empirical treatment is applied here to analysis of experimental data. The cases considered are: (1) a comparison of critical conditions occurring in Al-walled and stainless steel-walled reactors, and (2) a comparison of critical conditions in isolated and interacting pairs of water-reflected cylinders at the moderation which minimizes the critical volume. (auth)

83.

11005 UCRL-5006

California. Univ., Livermore. Radiation Lab.

CRITICAL MEASUREMENTS ON INTERMEDIATE-ENERGY GRAPHITE-U^{235} SYSTEMS. J. E. Schwager, F. A. Kloverstrom, and W. S. Gilbert. Nov. 15, 1957. 40p. Contract W-7405-eng-48. $6.30(ph OTS); $3.00 (mf OTS).

A system for the measurement of near-homogeneous carbon-U^{235} critical masses is described. Cores are constructed with thin, enriched-uranium foils spaced between graphite blocks. Fuel density is variable by use of different foil thicknesses and spacings. Reactivity is controlled by boron rods; standard reactor instrumentation permits critical operation at low power. Results of critical measurements on unreflected systems having atomic C/U^{235} ratios of 670, 1380, and 2590 are given. Thin reflectors of graphite and beryllium were also used. Corrections for self-shielding in the fuel foils and systematic errors are described. (auth)

84.

10078 ORNL-2499

Oak Ridge National Lab., Tenn.

COMPARISON OF POOL-TYPE REACTOR CRITICAL EXPERIMENTS WITH TWO-GROUP AND THIRTY-GROUP CALCULATIONS. Ernest G. Silver. June 9, 1958. 82p. Contract W-7405-eng-26. $1.00(OTS).

This report contains a description of a series of two-group and multigroup calculations of the critical mass of two clean-geometry configurations of the BSR. It also contains a description of critical experiments that were done to determine the validity of the calculations. Comparison of the results indicates that the calculations described are capable of predicting the critical mass within about 2% of the measured critical mass under the favorable geometric conditions maintained in the present experiments. (auth)

85.

1642 KAPL-M-PPS-1

Knolls Atomic Power Lab., Schenectady, N. Y.

CRITICAL MASSES OF HOMOGENEOUS WATER-MODERATED REACTORS. P. P. Szydlik. Aug. 12, 1957. 76p. Contract [W-31-109-eng-52]. $13.80(ph OTS); $4.80(mf OTS).

The critical mass density of U^{235} has been determined as a function of buckling for one region zirconium-water and stainless steel-water cores with and without B^{10}. This functional relationship enables results to be applied to any type of geometry. Using reflector savings derived from multigroup calculations the above results were applied to a limited range of cylindrical cores with water reflectors. The critical mass as a function of the volume of stainless steel, the difference between critical masses hot and cold for a particular composition, and reactivity coefficients for the uranium mass are also shown. (auth)

86.

10086 WAPD-TM-100

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

CRITICAL EXPERIMENTS IN A URANIUM-ZIRCONIUM WATER-MODERATED CORE WITH PLATE FUEL ELEMENTS AND SLAB GEOMETRY. W. F. Vogelsang and R. G. Serenka. Dec. 1957. 42p. Contract AT-11-1-GEN-14. $1.25(OTS).

Experiments were performed on light-water-moderated, highly enriched uranium-zirconium-aluminum cores in slab geometry. In one case, light water was used as a reflector on all sides of the core. In the second, a metal and water reflector containing natural uranium-niobium-aluminum was symmetrically added to a thinner, enriched uranium-zirconium-aluminum slab core. Experimental flux plots and machine calculated flux plots agreed within the experimental and calculational uncertainties. Few-group slowing-down models combined with spatial diffusion theory were used to predict criticality of the assemblies to within 2% in the worst case. The four-group slowing-down model, incorporating the effects of changes in the thermal flux spectrum, gave the best results. (auth)

1959

87.

131

CRITICAL MEASUREMENTS ON UO_3-H_3PO_4 SOLUTIONS. J. C. Allred, P. J. Bendt, and R. E. Peterson (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 4, 498-500(1958) Sept.

Critical measurements were carried out with solutions of UO_3(93.5% U^{235}) dissolved in 4.3 M H_3PO_4. A brief description of these measurements, their results, and an evaluation of some calculations are presented. (A.C.)

88.

5125 CF-58-12-36
Oak Ridge National Lab., Tenn.
CRITICAL CONCENTRATION DATA FOR HRT-TYPE REACTORS MODERATED BY D_2O-H_2O MIXTURES. R. Chalkley. Dec. 15, 1958. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Studies were made concerning the smallest core diameter and the most appropriate D_2O-H_2O composition ratio which would permit criticality in an HRT-type reactor; the fuel solution should contain no more than 10 g U/liter at 280°C. For the present core diameter of 32 inches (two-region operation), it was found that an addition of 10% light water to the moderator would reduce the critical concentration from 9.2 to about 7.8 g U/liter at 280°C (concentrations in g total U/liter, based on U of 93.4% enrichment). The smallest core diameter for which the reactor would remain critical with less than 10 g U/liter is about 29 inches. This occurred with a moderator composition of about 85% D_2O. (auth)

89.

22208 CF-59-9-3
Oak Ridge National Lab., Tenn.
EUROCHEMIC ASSISTANCE: ANSWERS TO QUESTIONS ON CRITICALITY. E. D. Clayton. Sept. 11, 1959. 4p. $1.80(ph), $1.80(mf) OTS.

Answers to questions on the criticality of Pu and Pu solutions are presented. (W.L.H.)

90.

2087 UCRL-5255
California. Univ., Livermore. Radiation Lab.
CRITICAL-MASS DETERMINATIONS OF LEAD-REFLECTED SYSTEMS. Robert E. Donaldson and Wilbur K. Brown. June 9, 1958. Decl. Sept. 22, 1958. 10p. Contract W-7405-eng-48. $0.50(OTS).

A series of experimentally determined critical masses of cylindrical and spherical lead-reflected oralloy systems are presented. Critical masses are given for two oralloy core sizes in both cases and also for reflected and unreflected ends in the cylindrical case. Experimental methods are described and a photograph of the assembly machine is included. (auth)

91.

17458
REACTIVITY AND FLUX MEASUREMENTS IN HIGHLY REFLECTED U^{235}-BISMUTH-GRAPHITE CRITICAL EXPERIMENTS. T. C. Engelder, H. W. Giesler, and J. P. Farrarr (Babcock and Wilcox Co.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 168-9(1959) June.

The Liquid Metal Fuel Reactor Experiment research and development program includes a series of critical experiments on U^{235}—Bi—graphite cores, highly reflected by graphite. The critical assembly is described and parameters are given. In the first set of experiments, the uranium concentration was varied, keeping the side and end reflector 2 to 3 ft thick. The critical diameters for the concentrations are reported. Flux traverses, intra-cell thermal flux structure, and reflector and geometrical effects were measured. (W.D.M.)

92.

21043 ORNL-2143
Oak Ridge National Lab., Tenn.
CRITICAL MASS STUDIES. PART VIII. AQUEOUS SOLUTIONS OF U^{233}. J. K. Fox, L. W. Gilley, and E. R. Rohrer. Sept. 23, 1959. 23p. Contract W-7405-eng-26. $1.00(OTS).

A series of experiments were performed to establish the critical parameters of aqueous solutions of uranyl nitrate and uranyl fluoride in which the uranium contained 98.7% U^{233}. Solutions were made critical in both spherical and cylindrical geometries with paraffin or water as a neutron reflector and, in two instances, with no reflector. The U^{233} concentration varied from 30 to 600 g/liter. The minimum critical mass observed was 590 g of U^{233} in the solution having an $H:U^{233}$ atomic ratio of 419 occupying a 10.4-in.-diam sphere. The minimum measured volume was 3.66 liters in a 6.7-in. equilateral cylinder containing a solution with an $H:U^{233}$ atomic ratio of 30.4. Extrapolated source neutron multiplication data indicate that a 5-in.-diam cylinder can be made critical if reflected, but a 4-in.-diam cylinder would be subcritical at all moderations. It was also found that 2.02 kg of U^{233} in an unreflected 10-in. equilateral cylinder is critical with a solution having an $H:U^{233}$ atomic ratio of 154. An unreflected sphere 12.6 in. in diameter is critical with 1.14 kg of U^{233} in a solution with an $H:U^{233}$ ratio of 381. Extension of the data to geometries other than those used experimentally was made by an empirical calculation. (auth)

93.

20016 RFP-123
Dow Chemical Co. Rocky Flats Plant, Denver.
PLUTONIUM GRAPHITE ASSEMBLIES. A. Goodwin, Jr., and C. L. Schuske. Sept. 29, 1958. Decl. July 24, 1959. 25p. Contract AT(29-1)-1106. $3.30(ph), $2.40(mf) OTS.

Neutron multiplication measurements and theoretical calculations were made on cylindrical assemblies of graphite and plutonium disks. (auth)

94.

5925

CRITICAL ASSEMBLY OF URANIUM–BERYLLIUM OXIDE. C. Gourdon, J. Martelly, M. Sagot, and G. Wanner. Bull. inform. sci. et tech. No. 16, 9-17 (1958) Apr. (In French)

Sub-critical Laplacien measurements were made on uranium–beryllium oxide lattices by the classical exponential method and the "Neutrostat" method. The assembly used was an orthocylinder with a vertical axis. It was composed of 7.5 tons of BeO with 90 fuel channels with a cross section of 5×5 cm^2, arranged in a square lattice of 15 cm pitch. By charging the assembly with 700 kg of uranium, enriched to 1.36%, it was possible to attain criticality under quite favorable conditions. The assembly has no radial reflector and the axial reflectors were poisoned with cadmium to have a very small albedo and to avoid a perturbation of the neutron spectra. Cadmium tubes around each uranium rod were moved up or down to control the reactivity. The material buckling, photoneutron effect in BeO, and the temperature and pressure coefficients of reactivity were studied. (J.S.R.)

95.

15208 Y-1248

Union Carbide Nuclear Co. Y-12 Plant, Oak Ridge, Tenn.

CRITICAL ASSEMBLIES OF URANIUM METAL. R. Gwin and W. T. Mee. Mar. 26, 1959. 17p. Contract W-7405-eng-26. $0.75(OTS).

Data on critical assemblies of fissionable uranium metals are summarized. Empirical studies are made for specific reflectors and geometries to determine the feasibility of extrapolating these data for conditions concerning nuclear safety problems. Also included are the influences on critical systems by various reflectors, U-235 isotopic enrichment, density, and small metal pieces homogeneously distributed in water. (auth)

96.

16103 AERE-R/R-2703

United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.

CRITICAL ASSEMBLIES WITH HEAVY WATER SOLUTIONS OF URANYL FLUORIDE (H.A.Z.E.L.). PART 2. PHYSICS. J. R. Harrison, M. F. Smith, W. G. Clarke, A. M. Mills, and J. A. Dyson. Nov. 1958. 67p. $1.96 (BIS).

Details are given of a series of critical assemblies and flux measurements undertaken in HAZEL using 46% U^{235} as UO_2F_2 in D_2O in cylindrical geometry. The fissile concentrations investigated were in the range 1939 to 6722 D/U_F, and in the geometry considered (2 inch diameter steel cylinder), the critical heights were in the range 70 to 200 cms. The minimum critical mass was 1.768 kg of U^{235} occurring at a concentration of 4687 D/U_F. (auth)

97.

9431 BNL-483(p.59-60)

Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.

AGE DETERMINATION FROM CETR CRITICAL ASSEMBLY DATA. Clifford J. Heindl. p.59-60 [of] THORIUM–U^{233} SYMPOSIUM, SPONSORED BY THE UNITED STATES ATOMIC ENERGY COMMISSION AT BROOKHAVEN NATIONAL LABORATORY, JANUARY 9-10, 1958. 2p.

An assembly consisting of 14.5% by volume Th, 85% polyethylene tape containing U^{235} oxide fuel, 24% Al, and 53% light water was brought to criticality at a series of seven different water heights by changing its radius. At each critical height, small perturbations were made in water level in both increasing and decreasing directions, and resultant periods measured. The average rate of change of reactivity with water height obtained at each level is listed. (W.L.H.)

98.

16920 K-1019(5th Rev.)

Oak Ridge Gaseous Diffusion Plant, Tenn.

CRITICALITY DATA AND NUCLEAR SAFETY GUIDE APPLICABLE TO THE OAK RIDGE GASEOUS DIFFUSION PLANT. H. F. Henry, A. J. Mallett, C. E. Newlon, and W. A. Pryor. May 22, 1959. 53p. Contract W-7405-eng-26. $1.50(OTS).

The available minimum experimental and theoretical criticality information for U^{235} enrichments of 1% to ~90% are presented together with the fundamental nuclear safety control criteria currently in effect at ORGDP. The fundamental nuclear safety criteria remain essentially unchanged from the previous edition of the report with the exception of the extension of nuclearly safe variables under 5% U^{235} enrichment, the increase of the minimum U^{235} enrichment considered to be non-reactive from 0.71% to 0.90%, and a statement of a new mass-volume principle. Other additions include guides for computing a solid angle and applying nuclearly safe variables to uranium materials, other than metal, of intermediate densities. (auth)

99.

16107 KY-294

Union Carbide Nuclear Co. Paducah Plant, Ky.

A GENERALIZED AND CONSISTENT METHOD FOR CALCULATING THE CRITICAL MASS OF HOMOGENEOUS AQUEOUS URANYL FLUORIDE SOLUTIONS. O. W. Hermann. June 8, 1959. 22p. Contract W-7405-eng-26. $0.75(OTS).

A more encompassing, yet consistent and practical, method of calculating the critical mass of aqueous uranyl fluoride solutions was developed. Comparison of calculated results with experimentation gave a maximum of only 2.3% error in the effective multiplication factor and 9% error in the critical mass, with absolute averages of 0.7% and 3.4%, respectively, through the ranges from high moderation down to an H/U of 25 for 90% U^{235} assays or an H/U of 4 for 2% U^{235} assays. (auth)

100.

5475 HW-58049
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
NUCLEAR SAFETY IN PROCESSING LESS THAN 5.0% U-235 ENRICHED REACTOR FUELS. N[orman] Ketzlach. Dec. 11, 1958. 23p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

Processing of reactor fuels in which the initial enrichment is less than 5% U^{235} is considered. The critical parameters for heterogeneous systems of fuel in water or in uranium solutions as well as homogeneous solutions of fuel and water are examined. In addition, nuclear safety in the use of boron poisoning to increase safe batch sizes and in the use of safe vessel geometries is discussed. A cartridge-type dissolver system for fuel elements is described, and it is pointed out that experiments to determine the neutron reflecting properties of enriched uranium solutions surrounding vessels are justified. (J.R.D.)

101.

3973 BNL-489(p.275-90)
California. Univ., Livermore. Radiation Lab.
STUDIES OF ENRICHED URANIUM GRAPHITE REACTOR SYSTEMS. Albert J. Kirschbaum. Appendix: SELF SHIELDING. W. S. Gilbert. p.275-90 [of] PROCEEDINGS OF THE FRENCH-AMERICAN CONFERENCE ON GRAPHITE REACTORS, [HELD AT BROOKHAVEN NATIONAL LABORATORY], NOVEMBER 12 TO 15, 1957. 16p. (UCRL-4983-T).

A/CONF.15/P/2408 supersedes this information in this paper.

The results to date are presented from studies of essentially homogeneous enriched uranium (93.5% U^{235}) graphite systems. Critical configurations for bare and graphite or beryllium reflected cores are given for carbon to uranium atomic ratios of 600:1, 1200:1, and 2400:1. The results of experiments to determine the systematic errors are given. This allows reduction of the critical size data to idealized geometries for comparison with neutronic calculations. By use of a pulsed neutron source, data on the prompt neutron population relaxation time as a function of buckling were obtained. The experimental technique and resultant data are discussed. A comparison of the critical buckling and time behavior data with a simple modified Fermi age theory is made. This includes discussion of the prompt neutron lifetime effectiveness of the control and safety rod system, the bulk neutronic properties of the graphite, and self shielding effects of the uranium. (auth)

102.

17453
CRITICAL DIMENSIONS OF UNREFLECTED BeO SYSTEMS FUELED BY THIN ENRICHED URANIUM FOILS. F. A. Kloverstrom, R. M. Deck, and A. J. Reyenga (University of California, Livermore). Nuclear Sci. and Eng. 2, No. 1, Suppl., 162-3(1959) June.

A series of critical measurements on unreflected parallelepipeds built of enriched (93.2%) uranium foils and BeO blocks was made for five fuel densities. The over-all fuel density was variable by changing the foil thickness and/or spacing. The assembly consists of a horizontal steel diaphragm suspended above a vertically moving table. The major portion of an assembly is built on the table and the remainder on the diaphragm. The two parts are assembled remotely by lifting the table. The thermal neutron absorption cross section for BeO was found to be 12 ± 2 mb. Results obtained to date are summarized. (W.D.M.)

103.

22212 UCRL-5369(Pt.1)
California. Univ., Livermore. Lawrence Radiation Lab.
CRITICAL MEASUREMENTS ON NEAR-HOMOGENEOUS BeO-MODERATED, ORALLOY-FUELED SYSTEMS. Fred A. Kloverstrom, Richard M. R. Deck, and Arnold J. Reyenga. July 1, 1959. 13p. Contract W-7405-eng-48. $0.50(OTS).

A series of critical measurements on unreflected systems, fueled by thin Oy foils, is described. Fuel density is varied by use of different foil thicknesses and spacings between foils. Five fuel densities were used which correspond to atomic BeO/U^{236} ratios from 246 to 7660. For three of these ratios, the fuel foil thickness was varied to find effects of self-shielding and flux depression. (auth)

104.

17970 AECD-4285
Los Alamos Scientific Lab., N. Mex.
A URANIUM CRITICAL ASSEMBLY AT LOW U^{235} CONCENTRATION. G. A. Linenberger. [1956]. Decl. June 1, 1956. 14p. Contract [W-7405-eng-36]. $3.30 (ph), $2.40(mf) OTS.

A critical metal assembly is described that has a cylindrical core of uranium with an average U^{235} concentration of 16¼% and a 3 inch reflector of natural uranium. The critical mass was determined to be 692 ± 4 kg of core material, from which a value for the critical mass of a bare sphere, having the same concentration, is calculated. Measurements of the prompt neutron decay constant, reactivity contributions of several materials, and radial and axial variations of fission rates of U^{235}, U^{238}, and Np^{237}, as well as ratios of fission cross sections for these isotopes, are reported. (auth)

105.

15206 CF-59-4-120

Oak Ridge National Lab., Tenn.

PRELIMINARY REPORT ON 2% U^{235}-ENRICHED UF_4-$C_{25}H_{52}$ CRITICAL ASSEMBLIES. J. T. Mihalczo, J. J. Lynn, Dunlap Scott, and W. C. Connolly. Apr. 22, 1959. 42p. $7.80(ph), $3.30(mf) OTS.

A series of critical experiments with blocks of 2% U^{235}-enriched UF_4-$C_{25}H_{52}$ was initiated at the ORNL Critical Experiments Facility. Thus far assemblies with H:U^{235} atomic ratios of 195 and 294 were built in parallelepipedal and simulated cylindrical geometries, both reflected and unreflected. From the results the minimum critical masses for reflected spheres were determined to be 16.3 and 8.5 kg of U^{235} for fuel mixtures with H:U^{235} atomic ratios of 195 and 294, respectively. The minimum critical masses for unreflected spheres of these two fuel mixtures are 24.3 and 12.7 kg of U^{235}, respectively. (auth)

106.

10634 LAMS-2288

Los Alamos Scientific Lab., N. Mex.

PHYSICS OF INTERMEDIATE REACTORS. C. B. Mills and F. W. Brinkley. Jan. 1959. 49p. Contract W-7405-eng-36. $1.50(OTS).

The neutron diffusion approximation equations and multigroup averaged cross sections were correlated with a systematic transport approximation study of a wide spectrum of critical assemblies and the results tabulated. The small differences were corrected by neutron age considerations where significant with respect to experimental results. Initial results for temperature effects on reactivity of graphite reactors are indicated. (auth)

107.

17370

k_∞ MEASUREMENTS OF 2 PER CENT U^{235} ENRICHED UF_4 IN PARAFFIN. V. I. Neeley (General Electric Co., Richland, Wash.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 67-9(1959) June.

Values of k_∞ for low enriched materials are of interest in the processing and handling of power reactor fuels. An experiment to measure k_∞ in the cavity of the Physical Constants Testing Reactor is described. The value of k_∞ for 2% U^{235} enriched UF_4-paraffin systems at a hydrogen-to-U^{235} atomic ratio of 195 was found to be 1.216 ± 0.013. (W.D.M.)

108.

16104 CF-59-6-45

Oak Ridge National Lab., Tenn.

A COMPARISON OF ELEMENTARY CRITICALITY CALCULATIONS WITH EXPERIMENTAL RESULTS. C. W. Nestor, Jr. June 11, 1959. 16p. $3.30(ph), $2.40(mf) OTS.

Several experiments have been performed at ORNL with light water solutions of uranyl nitrate (highly enriched in either U^{233} or U^{235}) in an essentially bare sphere 27 inches in diameter. Results are presented of several calculations with elementary bare reactor theory and a discussion of the observed discrepancies between the calculated and experimental results. If the observed critical concentration is used in the calculations, the calculated effective multiplication constant is less than unity; thus a higher critical concentration would be predicted than is actually observed. (auth)

109.

19050 CF-59-7-66

Oak Ridge National Lab., Tenn.

MULTIGROUP DIFFUSION THEORY CALCULATIONS FOR RECENT CRITICAL EXPERIMENTS. C. W. Nestor, Jr. July 21, 1959. 13p. Contract [W-7405-eng-26]. $3.30(ph), $2.40(mf) OTS.

In connection with the program of the measurement of eta for U^{233}, several critical experiments were performed with light water solutions of uranyl nitrate in an essentially bare sphere 27 inches in diameter. Results of two multigroup-diffusion-theory calculations for the above experiments are presented. Assumed cross sections, material concentrations, detailed neutron balances, and a comparison with elementary theory are included. The agreement between the calculated and experimental multiplication constants is excellent for the multigroup calculation but only fair for the elementary calculation. The latter method overestimates the fast leakage so that the computed multiplication constant is less than that found experimentally. (auth)

110.

17435

A CORRELATION OF THE CRITICAL CONDITIONS FOR HOMOGENEOUS BARE REACTORS. Benjamin Pinkel and George B. W. Young (RAND Corp., Santa Monica, Calif.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 142-4(1959) June.

A method of correlating the critical conditions of bare homogeneous reactors is presented. This method is applied to the results of an 18 group analysis for a series of bare, cold, homogeneous reactors comprised of pure U^{235} and the moderators H_2O, D_2O, LiH, Be, BeO, and C over a wide range of moderator to uranium mole ratios. The properties contained in the correlation parameters used are the buckling, the transport

mean free path, the ratio of U^{235} volume to total reactor volume, the moderator thermal absorption cross section, the neutron energy degradation parameter, and the moderator to uranium mole ratio. (W.D.M.)

111.

9433 BNL-483(p.73-4)
Los Alamos Scientific Lab., N. Mex.
CRITICALITY DATA ON METAL U^{233} SPHERICAL SYSTEMS. E. A. Plassman and D. P. Wood. p.73-4 [of] THORIUM–U^{233} SYMPOSIUM, SPONSORED BY THE UNITED STATES ATOMIC ENERGY COMMISSION AT BROOKHAVEN NATIONAL LABORATORY, JANUARY 9-10, 1958. 2p.

Data consisting of critical reflector thicknesses for 2.41 kg and 10.0 kg U^{233} metal spheres are summarized. Six-group S_4 predictions are compared with observed results for the Oralloy-enriched U and U reflected cases. (W.L.H.)

112.

7624 UCRL-5349
California. Univ., Livermore. Lawrence Radiation Lab.
CRITICAL PARAMETERS OF SPHERICAL SYSTEMS OF ALPHA-PHASE PLUTONIUM REFLECTED BY BERYLLIUM. H. Robert Ralston. Sept. 10, 1958. Decl. Jan. 7, 1959. 9p. Contract W-7405-eng-48. $0.50(OTS).

The critical thicknesses of Be reflectors for α-phase Pu spheres of 2.472, 3.217, 3.933, 4.664, and 5.426 kg were found to be 32.0 ± 4.0, 21.0 ± 1.0, 13.0 ± 0.1, 8.17 ± 0.03, and 5.22 ± 0.02 cm, respectively, for Be density of 1.84 g/cc and Pu average density 19.25 g/cc. A description of measuring techniques and apparatus is given. (T.R.H.)

113.

17974 TID-3533
Technical Information Service Extension, AEC.
CRITICALITY STUDIES. A Literature Search. Raymond L. Scott, comp. Aug. 1959. 117p. $2.50 (OTS).

A compilation of 1122 references on criticality studies is presented including references to experimental and theoretical studies of conditions for criticality with various materials and configurations, as well as references on the safe handling of fissionable materials, both during critical experiments and in plants for the chemical and metallurgical processing of fissionable materials. (W.D.M.)

114.

128 ORNL-2201
Oak Ridge National Lab., Tenn.
TWO BERYLLIUM-MODERATED CRITICAL ASSEMBLIES. E. L. Zimmerman. Oct. 20, 1958. 66p. Contract W-7405-eng-26. $1.75(OTS).

Two unreflected critical assemblies using beryllium as the moderator and 93.4% enriched uranium metal as the fuel were built to provide a basis for the evaluation of certain reactor calculational procedures. Control and safety rods of the core-element-removal type were used in order that the final assemblies would not be complicated by strong absorber rods. In the first assembly (CA-1), which had outside dimensions of 21.0 × 21.0 × 23.3 in., the 0.01-in.-thick uranium disks were separated by 1-in.-thick blocks of beryllium, which gave a Be:U^{235} atomic ratio of 390 and a fuel loading of 18.08 kg of U^{235}. The extrapolated value of k_{eff} for the system was 1.0054. In the second assembly (CA-18), which had outside dimensions of 24.0 × 28.4 × 24.1 in., the fuel disks were separated by 4-in.-thick blocks of beryllium, which gave a Be:U^{235} ratio of 1560 and a fuel loading of 7.65 kg of U^{235}. For this assembly the extrapolated k_{eff} value was 1.0020. The observed uranium cadmium fractions in the two assemblies were 0.46 and 0.86, respectively. A number of multigroup calculations were made to evaluate the effects of various corrections and assumptions. It was concluded that the calculated neutron multiplication is very sensitive to the competition between leakage and slowing down at high energies, a range where fundamental data are uncertain. Without resolving the detailed neutron behaviors in this range, a reasonable selection of data within experimental uncertainties will give satisfactory values for such quantities as critical size. (auth)

1960

115.

19926 BAW-1173
Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va.
THORIUM URANIUM PHYSICS EXPERIMENTS (TUPE). Monthly Report for September 1959. R. C. Anderson, M. L. Batch, R. H. Lewis, N. L. Snidow, and W. M. Vannoy. 29p. Contract AT(11-1)766. OTS.

The cores to be studied are described, and the status of the project is given. The results of measurements on core 25√2B are discussed in terms of critical mass, perturbation of M/W, thermal disadvantage factor, buckling, and cadmium ratio experiments. (W.D.M.)

116.

22263 RFP-190

Dow Chemical Co. Rocky Flats Plant, Denver.

PLUTONIUM PLEXIGLAS ASSEMBLIES. PART II. G. H. Bidinger, C. L. Schuske, and D. F. Smith. Apr. 6, 1960. Changed from OFFICIAL USE ONLY May 19, 1960. 20p. Contract AT(29-1)-1106. OTS.

Neutron multiplication measurements were made on tamped and untamped cylindrical assemblies. The assemblies consisted of plutonium metal sheet moderated with Plexiglas. Experiments were performed to evaluate the effects of inhomogeneity. This work is a continuation of RFP-178. (auth)

117.

810 AERE-R/R-2731

United Kingdom Atomic Energy Authority. Atomic Energy Research Establishment, Harwell, Berks, England.

CRITICAL ASSEMBLIES WITH HEAVY WATER SOLUTIONS OF URANYL FLUORIDE (HAZEL). PART III. THEORETICAL ANALYSIS. C. Carter, P. K. H. Lang, and G. Myatt. July 1959. 15p. BIS.

Multigroup diffusion theory was used to analyze the critical size and neutron flux measurements obtained with uranyl fluoride–heavy water homogeneous critical assemblies. It is found that two-group theory is inadequate to explain the experimental data, but a reasonably good correlation can be made with a six-group theory. Alternatively, two-group theory can be adjusted to fit the data by using an artificially high slowing down area for heavy water. (auth)

118.

19691

Argentina. Comisión Nacional de Energía Atómica, Buenos Aires.

DISPOSITIVO PARA ENSAYOS CRITICOS (RA-1,5). Informe No. 11. (Critical Assembly (RA-1.5). Report No. 11). Carlos Domingo, Miguel Geiger, Velia Hoffmann de Geiger, and Jorge Sare. 1959. 19p.

The RA-1.5 critical assembly is briefly described. The procedures used in the testing of the safety apparatus and in the determination of the critical mass are given. The results of five critical tests are presented. (J.S.R.)

119.

16166 GAT-DM-769

Goodyear Atomic Corp., Portsmouth, Ohio.

SAFE GEOMETRIES AND MASS AT ASSAYS BELOW FIVE PER CENT U^{235}. J. L. Feuerbacher. May 18, 1959. 9p. OTS.

Investigations were made to determine the most reliable method of estimating safe geometries at assays below 5% U^{235}. Conservative methods of estimating safe mass at the assays considered are also examined. Data on safe geometrical and mass parameters are included. (J.R.D.)

120.

13126 HW-24454

Hanford Works, Richland, Wash.

PROPOSED METHOD FOR TREATING HYDROGEN DISPLACEMENT EFFECTS IN CRITICAL MASS MEASUREMENTS. Paul F. Gast. May 13, 1952. Decl. Feb. 24, 1960. 12p. OTS.

Analysis of data on the critical masses of Pu solutions revealed that significant increases in the critical mass occur when the H density is reduced by dilution with heavier nuclei. Recommendations for study of this problem are outlined. Calculations for various situations are included. (B.J.R.)

121.

6844 RFP-158

Dow Chemical Co. Rocky Flats Plant, Denver.

PLUTONIUM GRAPHITE ASSEMBLIES—PART II. A. Goodwin, Jr. and C. L. Schuske. Aug. 10, 1959. Decl. Dec. 9, 1959. 21p. Contract AT(29-1)-1106. OTS.

Neutron multiplication measurements were made on a number of cylindrical assemblies of Pu and graphite disks. S_n calculations were made on homogeneous mixtures of Pu and graphite with varying C/Pu ratios and varying reflector thickness. (auth)

122.

9166 ANL-6115

Argonne National Lab., Lemont, Ill.

POST CRITICALITY STUDIES ON THE TREAT REACTOR. Haig P. Iskenderian. Feb. 1960. 13p. Contract W-31-109-eng-38. OTS.

Criticality calculations for the TREAT reactor, assuming no slots and no test hole, and using a modified Fermi-Age Theory which allows for epithermal absorption and fission, indicated a critical radius of 59 cm. At the time of startup TREAT became critical at a radius of 67.8 cm. The results of spectrochemical and chemical tests indicated a high boron impurity (an average of 7.6 ppm) in the core graphite, and a maximum of about 1.8 wt.% H_2O in one sample from the permanent reflector graphite. The excess amount of boron impurity in the core graphite appears to be the major cause of the discrepancy. Values of the negative temperature coefficient of reactivity and of neutron lifetime were calculated for the critical reactor containing 7.6 ppm of boron, assuming no slots or test holes, and the results are found to be in fair agreement with measurements. The source of excess boron impurity was traced to the borated steel separators which were used during the baking of the fuel elements. (auth)

123.

12330

FAST REACTORS. Leonard J. Koch (Argonne National Lab., Lemont, Ill.) and Hugh C. Paxton (Los Alamos Scientific Lab., Albuquerque, N. Mex.). Ann. Rev. Nuclear Sci. 9, 437-72(1959).

A review of the United States, Russia, and United Kingdom fast reactor and fast critical assembly programs is given. Critical size, perturbation, and spectral data obtained from fast critical assemblies are treated. Coolant and heat transfer problems, control and kinetics transient accidents, and fuel cycles of fast reactors are discussed. (C.J.G.)

124.

16167 HW-24514(Del.)

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

CRITICAL MASS STUDIES OF PLUTONIUM SOLUTIONS. F. E. Kruesi, J. O. Erkman, and D. D. Lanning. May 19, 1952. Reissued in declassified form, Feb. 15, 1960. 82p. Contract W-31-109-Eng-52. OTS.

The chain reacting conditions for plutonium nitrate in water solution were examined experimentally for a variety of sizes of spheres and cylinders. The effects on the critical mass of the displacement of hydrogen and the addition of poisons to the fuel were measured in water tamped and bare reactors. The data reveal that the absorption cross-section of Pu^{240} is 925 ± 200 barns and the minimum critical mass of Pu^{239} in water is 510 g at a concentration of about 33 g/liter. (auth)

125.

7988

ENRICHED-URANIUM HYDRIDE CRITICAL ASSEMBLIES. G. A. Linenberger, J. D. Orndoff, and H. C. Paxton (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 7, 44-57(1960) Jan.

Assemblies considered consist of approximate spheres of enriched-uranium hydride composition (approximating UH_3) in 8-in. thick normal uranium and nickel reflectors and in a uranium reflector with nickel liner. Data are of the following types: (1) critical sizes, (2) values of Rossi alpha in the neighborhood of delayed critical, (3) activation rates of various internal neutron detectors, and (4) reactivity coefficients of a variety of elements. From the reactivity coefficients at various radial positions, changes in critical mass corresponding to small changes in composition and density are computed. (auth)

126.

20780 HW-63576(p.65-9)

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

PHYSICS OF NUCLEAR SAFETY. Criticality Measurements of Heterogeneous 3.1 Per Cent Enriched Uranium and Water Systems. R. C. Lloyd, E. D. Clayton, R. B. Smith, and V. I. Neeley. p.65-9 of NUCLEAR PHYSICS RESEARCH QUARTERLY REPORT [FOR] OCTOBER, NOVEMBER, DECEMBER 1959.

The program of critical approach and exponential measurements of 3.063 per cent enriched uranium rods in light water was continued. Critical masses, buckling values, and some measured extrapolation length values were reported for three rod diameters (0.300, 0.600, and 0.925 inches) in previous quarterly reports. Measurements were made with rods of 0.175-inch diameter by 23.5 inches in length. These rods were encased with 0.025-inch wall Lucite tubes for insertion into hexagonal lattice frameworks. Measurements were carried out in the same manner as described previously. All lattices were moderated and completely reflected with light water. (auth)

127.

14250 CF-60-4-24

Oak Ridge National Lab., Tenn.

COMPARISON OF k_∞ MEASUREMENTS IN A CRITICAL ASSEMBLY WITH k_∞ MEASUREMENTS IN THE PHYSICAL CONSTANTS TESTING REACTOR. J. T. Mihalczo. May 3, 1960. 34p. Contract W-7405-eng-26. OTS.

The infinite medium multiplication factor, k_∞, for a homogeneous mixture of 2% U^{235}-enriched UF_4 in paraffin, was determined from a series of critical experiments in which known changes in the buckling were made and the resulting stable periods measured. The value determined, using a two-group model for the nonleakage probability, was 1.197 ± 0.09. Within the quoted errors this value is in agreement with an earlier value of k_∞ = 1.216 ± 0.013 for the same material experimentally determined in the Physical Constants Testing Reactor. (auth)

128.

1103 LAMS-2288(Suppl. 1)

Los Alamos Scientific Lab., N. Mex.

PHYSICS OF INTERMEDIATE REACTORS. Supplement. C. B. Mills. Apr. 1959. 81p. Contract W-7405-eng-36. OTS.

Multigroup neutron diffusion and transport equations have been shown to correlate neutron cross sections and simple critical experiments for a wide variety of materials to approximately second order accuracy. This system was used for criticality survey work for the moderators H and C, giving critical radius for moderator to fissionable material atomic ratio and temperature. Critical radius dependence for D_2O, Be, and BeO was computed for U^{235} only. Neutron group averaged cross sections supplementing and correcting previous listings are tabulated. These are consistent with the literature. (auth)

129.

5775 HW-62031

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

A COMPARISON OF PCTR AND CRITICAL EXPERIMENT DETERMINATIONS OF k_∞ FOR TWO WEIGHT PER CENT URANIUM-235 ENRICHED UF_4 IN PARAFFIN. V. I. Neeley. Sept. 22, 1959. 28p. Contract AT(45-1)-1350. OTS.

The value of k_∞ for 2 wt.% U^{235} enriched UF_4 paraffin moderated at a H/U^{235} atomic ratio of 195 was determined in the PCTR. This material was returned to ORNL in order that critical experiments might be conducted to determine k_∞. The two values of k_∞ were then compared. The value of k_∞ as determined in the PCTR was dependent upon the cross-section values which were chosen. In this experiment "effective" cross-section values obtained by averaging the cross section over the Wigner-Wilkens spectrum of a similar type of mixture were used. The value of k_∞ determined in the PCTR was 1.216 ± 0.013. The value obtained from the work carried out at ORNL was 1.200 ± 0.011 from a one-group treatment of the critical experiments and 1.202 ± 0.012 from a two-group treatment of the work. A theoretical calculation of k_∞ gives a value of 1.23. This is in good agreement with the experimental value. Critical experiments done on this material at ORNL show that the minimum critical mass for a bare "square" cylinder contains approximately 28 kg of U^{235}. A calculated value of the amount of U^{235} in a just critical bare square cylinder was obtained using the experimental k_∞ value. Again there was good agreement with the experimental results. (auth)

130.

7987 LA-2085

Los Alamos Scientific Lab., N. Mex.

CRITICAL ASSEMBLY OF URANIUM METAL AT AN AVERAGE U^{235} CONCENTRATION OF 16-1/4%. J. J. Neuer, G. E. Hansen, G. A. Jarvis, G. A. Linenberger, and D. S. Young. Oct. 1956. Decl. Jan. 4, 1960. 28p. Contract W-7405-eng-36. OTS.

A uranium metal critical assembly consisted of a 15-in. diameter core with an average U^{235} content of 16¼%, surrounded by a 3-in. thick natural uranium reflector. The critical mass was 692 kg of core material. (auth)

131.

20782 PGR-97(R)

United Kingdom Atomic Energy Authority. Production Group. Risley, Lancs, England.

AN EMPIRICAL CORRELATION OF THE EXPERIMENTAL DATA ON HOMOGENEOUS CRITICAL ASSEMBLIES OF URANIUM AND HYDROGEN OF ALL ENRICHMENTS. B. G. Owen. Apr. 1960. 42p. BIS.

The original correlation on highly enriched uranium–hydrogen critical systems is extended to all enrichments. By using three empirical equations and the one-group buckling relations, the physical size of any uranium–hydrogen homogeneous critical assembly in simple geometry can be predicted. The predictions are compared with experiment where experimental data are available. The derived reflected spherical, infinite cylinder, and slab dimensions are shown. (auth)

132.

9179

REACTOR PHYSICS EXPERIMENTS USING SUBCRITICAL SYSTEMS. A. Ōyama and S. An (Univ. of Tokyo). J. Atomic Energy Soc. Japan 1, 436-46(1959). (In Japanese)

A review is presented on reactor physics experiments using subcritical systems. Experimental facilities and techniques are described, and the correlations of theory and experiments are discussed. Some of the interesting pulsed neutron experiments utilizing subcritical systems are also discussed. (auth)

133.

7986 LA-1671

Los Alamos Scientific Lab., N. Mex.

BARE CRITICAL ASSEMBLIES OF ORALLOY AT INTERMEDIATE CONCENTRATIONS OF U-235. H. C. Paxton, G. A. Linenberger, L. L. Lowry, R. N. Olcott, J. D. Orndoff, and J. E. Sattizahn. May 1954. Decl. Jan. 4, 1960. 46p. Contract W-7405-eng-36. OTS.

Bare cylindrical critical assemblies were constructed of stacked plates of 93.4% enriched and natural uranium. Proportions were adjusted to give average U^{235} concentrations of 53.6, 37.7, and 29.0%. Approximate relations between critical parameters and U^{235} concentration were obtained. (W.D.M.)

134.

16168 LAMS-2415

Los Alamos Scientific Lab., N. Mex.

CRITICAL DATA FOR NUCLEAR SAFETY GUIDANCE. H. C. Paxton, comp. Feb. 1960. 68p. Contract W-7405-eng-36. OTS.

A collection of criticality data is presented for the evaluation of nuclear safety problems. Various geometries are treated, e.g., spheres, cylinders, slabs, cubic arrays, annuli, etc. The isotopes of U^{233}, U^{235}, and Pu^{239} at various enrichments and in different solutions are considered. (W.D.M.)

135.

25020

HIGH TEMPERATURE CRITICAL SYSTEMS. Harry L. Reynolds (Univ. of California, Livermore). ARS (Am. Rocket Soc.) J. 30, 772-5(1960) Aug.

A facility called "Hot Box" was constructed at Jackass Flats, Nevada, to be used in studies of high-temperature critical assemblies of simple geometry, in order to determine whether calculation procedures are adequate for estimating the effect of temperature changes. Uranium foils of 2-mil thickness and moderator blocks are assembled at room temperature and raised to higher temperatures by hot gas; the critical mass at one particular temperature is determined. Data are presented for graphite-moderated bare-unreflected and reflected systems at temperatures up to 1200°F and should be useful for nuclear propulsion system calculations. (D.L.C.)

136.

9911 RFP-178

Dow Chemical Co. Rocky Flats Plant, Denver.

PLUTONIUM PLEXIGLAS ASSEMBLIES. C. L. Schuske, G. H. Bidinger, A. Goodwin, Jr., and D. F. Smith. Jan. 20, 1960. Decl. Feb. 16, 1960. 20p. Contract AT(29-1)-1106. OTS.

Neutron multiplication measurements were made on tamped and untamped cylindrical assemblies. The assemblies consisted of Pu metal sheet moderated with Plexiglas. Experiments were performed to evaluate the effects of inhomogeneity. (auth)

137.

16169 Y-839

Carbide and Carbon Chemicals Co. Y-12 Plant, Oak Ridge, Tenn.

EMPIRICAL STUDIES OF CRITICAL MASS DATA. PART III. C. L. Schuske and J. W. Morfitt. Jan. 16, 1952. Decl. Mar. 7, 1960. 13p. Contract W-7405-eng-26. OTS.

The critical mass curves of an equilateral cylindrical reactor as a function of moderation were calculated by graphical methods for H: U^{235} atomic ratios of 25 down to 1.4. The critical assemblies were built up of small cubes. These cubes were of two types, H-cubes and U-cubes. The H-cubes were small blocks of polyethylene ($CH_{1.92}$) approximately 1" on the edge. The U plastic cubes (U-cubes) contained a mixture of UF_4 (U^{235}) isotopic concentration of 9.3% and polytetrafluoroethylene ($CF_{2.02}$) pressed together to form a material having an over-all density of 4.73 g/cm^3. The infinite cylinder and slab dimensions as functions of moderation were approximated for the low values of H: U^{235} atomic ratio. (C.J.G.)

138.

819 NP-8000

Lockheed Nuclear Products, Marietta, Ga.

FLUX DISTRIBUTIONS AND CRITICALITY STUDIES FOR THE CRITICAL EXPERIMENT REACTOR. R. B. Shain. Sept. 1959. 32p. Contract AF33(600)-38947. (NR-64).

One- and two-dimensional flux distributions are given for the Critical Experiment Reactor using two-group diffusion theory. One-dimensional flux distributions are shown for the two- and four-region reactor models along with two-dimensional iso-flux distributions. The calculated critical loading was found to be 18.6 fuel elements and is 0.6% lower than the experimentally-determined critical loading value. Values of K_{eff} for a 30- and 32-element core, using a two-dimensional, two-group diffusion theory program, are 1.106 and 1.130, respectively. One- and two-dimensional methods for calculating reactivities of the CER agree within 1%. (auth)

139.

19927 BAW-1193

Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va.

THORIUM URANIUM PHYSICS EXPERIMENTS (TUPE). Monthly Report for February 1960. N. L. Snidow. 7p. Contract AT(11-1)766. OTS.

The status of the project is summarized, and the results of clean core measurements on Cores 15B and 15A are given. (W.D.M.)

140.

22261 AHSB(S) Handbook 1

United Kingdom Atomic Energy Authority. Authority Health and Safety Branch, Risley, Lancs, England.

HANDBOOK OF CRITICALITY DATA FOR PLANT DESIGNERS AND OPERATORS. 1960. 52p.

Criticality data are presented which are intended for use by chemical plant designers and operators. The assumptions and definitions are based on operations and accidents that might reasonably be considered possible in such plants. Graphs are presented of the four commonly encountered critical parameters: mass, volume, radius of an infinite cylinder, and thickness of an infinite slab. Curves are given for systems involving Pu^{239}, 30-93% U^{235}, 5-30% U^{235}, and less than 5% U^{235}. (W.D.M.)

1961

141.

16407 (LAMS-2489) CRITICAL MASSES OF COMPOSITES OF Oy AND Pu-239-240 IN FLATTOP GEOMETRY. D. M. Barton, William Bernard, and G. E. Hansen (Los Alamos Scientific Lab., N. Mex.). Dec. 1960. Contract W-7405-ENG-36. 18p.

Three critical plutonium-enriched uranium composites with plutonium of different Pu^{240} content are described. In these systems one gram of Pu^{240} is equivalent to ~0.63 grams of Pu^{239} and an analysis is presented which translates this datum to $\bar{\nu}(Pu^{240}, 2\ Mev) = 3.32 \pm 0.14$ neutrons emitted per Pu^{240} fission induced by a 2 Mev neutron. (auth)

142.

2068 Y-801(Del.)
Oak Ridge National Lab., Y-12 Area, Tenn.
CRITICAL MASS STUDIES. PART VI. Dixon Callihan, D. F. Cronin, J. K. Fox, J. W. Morfitt, E. R. Rohrer, and D. V. P. Williams. Aug. 8, 1951. Decl. with deletions June 17, 1960. 32p. Contract W-7405-eng-26. OTS.

Data are presented on the conditions under which arrays of enhanced and natural uranium slugs of the Hanford type can be made critical with light-water moderator, and composite reflectors of water, lead, and natural uranium. An effort was made to determine the minimum number of enhanced slugs that could be made critical under such conditions, to establish the nuclear safety of a lead-lined shipping container, and to obtain a qualitative comparison of the relative reflecting ability of lead, water, methacrylate plastic, and natural uranium. (auth)

143.

6605 A URANIUM-METAL EXPONENTIAL EXPERIMENT. Curtis G. Chezem (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 8, 652-69(1960) Dec.

Certain conflicts arising from previous measurements of neutron flux parameters in the equilibrium spectrum of natural uranium were resolved. The parameters which were investigated are given along with "best" values as measured in this work: Material buckling, $-0.0119 \pm 0.0005\ cm^{-2}$; Diffusion length, 9.17 ± 0.18 cm; U^{238}/U^{235} fission cross-section ratio, 239 ± 7; Pu^{239}/U^{235} fission cross-section ratio, 250 ± 16; Np^{237}/U^{235} fission cross-section ratio, 14.5 ± 0.5; and U^{238} inelastic scattering cross section, 2.00 ± 0.04 barns. The experiment was performed at the Pajarito critical assemblies facility utilizing two exponential columns of natural uranium, each 30.7-in. high, having diameters of 15 and 21 in. and excited by a small fast reactor. The system was outdoors, elevated some 11 ft above the ground level to reduce flux perturbations due to backscattering of neutrons. Perturbation corrected measurements in both columns made by several detection methods and with various source spectra agree to within experimental error and are consistent with calculated values. (auth)

144.

21396 (DP-532) HANDBOOK OF NUCLEAR SAFETY. Hugh K. Clark (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). Jan. 1961. Contract AT-(07-2)-1. 150p.

Discussions are given of the factors that determine a critical mass, consequences of attaining a critical mass, theory of chain reactions, and margins of safety. Critical and safe conditions are included for the fissionable materials, U-233, U-235, and plutonium, both as pure metals and when alloyed with other metals. Considerations are given for heterogeneously and homogeneously moderated systems, and interactions occurring between units in air and water. (B.O.G.)

145.

4715 (NAA-SR-5610) AN ENRICHED UO_2-ZrH CRITICAL ASSEMBLY. M. V. Davis, A. W. Thiele, and L. I. Moss. Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif. Nov. 1, 1960. 34p. Contract AT-11-1-GEN-8. OTS.

The SNAP Critical Assembly is a pseudospherical nuclear reactor with a fixed hydrogen moderator, 93.17% U^{235} fuel, and a beryllium and graphite reflector. The core is made up of segments of cold pressed ZrH with 8% by weight UO_2 powder. The assembly is constructed in two hemispheres with horizontal faces which are brought in contact. Instrumentation and reactivity, activation, and intrinsic behavior measurements are discussed. (W.D.M.)

146.

21894 CRITICAL EXPERIMENTS FOR THE PRELIMINARY DESIGN OF THE ARGONNE HIGH FLUX REACTOR, PART A. J. W. L. DeVilliers, Q. L. Baird, J. Juliano, C. N. Kelber, R. Kiyose, and K. E. Plumlee (Argonne National Lab., Ill). Trans. Am. Nuclear Soc., 4: No. 1, 109-10(June 1961).

147.

3302 (BAW-1185) LIQUID METAL FUEL REACTOR EXPERIMENT; CRITICAL EXPERIMENT REPORT OF EXPERIMENTAL DATA. T. C. Engelder, H. W. Giesler, J. P. Farrar, and J. H. Mortenson. Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va. Mar. 1960. Changed from OFFICIAL USE ONLY Oct. 18, 1960. 78p. Contract AT(30-1)-1940. OTS.

Results are reported of a series of critical experiments on U^{235}-bismuth-graphite cores, highly reflected by graphite, performed during the course of the Liquid Metal Fuel Reactor Experiment (LMFRE) research and development program. The experiment employs a large movable table assembly on which is erected a graphite honeycomb, with provision for changes in core composition and reflector thickness. Information on the physical arrangement, table and table drive system, control rods and drive system, neutron source and drive, and instrumentation is

provided. The critical dimensions of 12 assemblies of pseudocylindrical geometry are reported. These assemblies cover the range of parameters N_{25}/N_{Bi} = 600 to 1200 ppm, V_{Bi}/V_G = 0 and 0.5, and side and end reflectors from zero to 42 in. of graphite. Various small reactivity corrections, such as temperature, residual table gap, room reflections, rod guides and voids, and heterogeneities, were also measured. Extensive microscopic and macroscopic flux measurements were made using bare and cadmium-covered U^{235}, gold, and Dy–Al alloy foils and wires. The intracell flux distribution in three representative assemblies was measured, and flux ratios were computed for thermal-utilization calculations. Gross-flux traverses were made radially, axially, and diagonally, and over-all contour diagrams were plotted to obtain additional data on flux separability with which to check calculational methods. The important foil-correction factors were studied experimentally so that accurate flux shapes in two-region reactors could be reported. Subcadmium and epicadmium fission distributions were measured, and the radial fuel importance was studied in one assembly. Critical buckling measurements were limited by the small asymptotic region in the reflected assemblies, but sufficient information was obtained to determine the axial and radial reflector savings for N_{25}/N_{Bi} = 1200 ppm. It was also possible to construct two bare or near-bare assemblies, for which the critical buckling is reported. (auth)

148.

7018 (ORNL-3016(p.59-70)) SOLUTION EXPERIMENTS IN A FLUX-TRAP CRITICAL ASSEMBLY: PRELIMINARY STUDY FOR HIGH FLUX ISOTOPE REACTOR. J. K. Fox, L. W. Gilley, and D. W. Magnuson (Oak Ridge National Lab., Tenn.).

A study of a flux-trap critical assembly was made, using a solution of $UO_2(NO_3)_2$ enriched to 93.5 wt. % in U^{235} and dissolved in mixtures of D_2O and H_2O. The assembly was a preliminary mock-up of the HFIR, and the data were intended to aid in the establishment of design parameters for the HFIR. Critical parameters, relative flux distributions, flux ratios, and the effects of centrally located air- or aluminum-filled "voids" were obtained. (auth)

149.

2063 AERE-M-718

United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.

DENSITIES OF HYDROGEN AND URANIUM AND THEIR EFFECT UPON CRITICALITY. E. V. Garner and K. D. B. Johnson. July 1960. 10p. BIS.

The densities of hydrogen and uranium in crystalline solids and liquids which may occur during the processing of fissile materials are given. Criticality dependence upon density is discussed. (W.D.M.)

150.

6600 CRITICAL PLUTONIUM AND ENRICHED-URANIUM-METAL CYLINDERS OF EXTREME SHAPE. G. E. Hansen, H. C. Paxton, and D. P. Wood (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 8, 570-7 (1960) Dec.

Critical configurations were established with enriched uranium in the form of squat 15.0-in. dia. cylinders and elongated 3.24-in. dia. cylinders. These cores were reflected by depleted uranium, polyethylene, graphite, and water; the squat cylinder was unreflected and reflected by beryllium of various thicknesses. Critical systems of plutonium were squat 6.0-in. dia. cylinders and elongated 2.25-in. dia. cylinders reflected by normal uranium, graphite, water, and in one case, polyethylene. Observed critical heights and diameters were corrected to correspond to standard enriched-uranium and plutonium densities and concentrations. These are tabulated along with effective extrapolation distances. (auth)

151.

6602 CRITICAL MASSES OF ENRICHED-URANIUM CYLINDERS WITH MULTIPLE REFLECTORS OF MEDIUM-Z ELEMENTS. G. E. Hansen, D. P. Wood, and W. U. Geer (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 8, 588-94(1960) Dec.

Critical masses were measured for enriched-uranium-metal cylinders reflected on both ends and on one end only by multiple layers of two and three of the metals copper, iron, zinc, nickel, and stainless steel. For other measurements the core was partially moderated with graphite and with polyethylene so as to give the influence of decreased neutron energy upon reflector savings of the multiple reflectors. Critical mass values with composite reflectors are less than the simple averages of values for the elements alone. This reduction of critical mass, most pronounced for the Ni-Fe reflectors, is primarily because the self-shielding of the scattering resonances in medium-Z elements is appreciable when one-element reflectors are used, and is reduced when two or more of these elements are mixed in the reflectors. (auth)

152.

29881 (K-1335(Del.)) APPLICATION OF INTERACTION CRITERIA TO HETEROGENEOUS SYSTEMS. H. F. Henry, C. E. Newlon, and J. R. Knight (Oak Ridge Gaseous Diffusion Plant, Tenn.). June 4, 1957. Decl. with deletions July 24, 1961. Contract W-7405-eng-26. 21p.

Recent experimental criticality data with homogeneous and heterogeneous systems of interacting containers were used in evaluating an interaction principle for the safe storage and handling of dissimilar containers of fissionable materials. The experimental data which included slab and cylindrical geometries, U-235 assays of 93.2%, and H/U-235 atomic ratios from 0 to 330, and which extend below the useful range of a two-group theory previously

used to evaluate interaction experiments, indicate that the principle is valid over the wide range of criticality parameters considered, and that a homogeneous system of interacting containers is, in general, more highly reactive than any corresponding heterogeneous one. An analysis was also made of the safety of cylindrical storage units where criticality control is based upon mass rather than upon geometric limitations. Calculations using a two-group interaction theory indicate that, for containers meeting ORGDP safe interaction criteria, either uniform dilution or concentration of the fuel from an optimum H/U-235 ratio of about 600 will result in a smaller container separation being required. (auth)

153.

31489 (ANL-6401) CRITICAL STUDIES OF DILUTE CARBIDE FAST REACTOR CORE. ZPR-III Assembly 34. R. J. Hubert, J. K. Long, R. L. McVean, and J. M. Gasidlo (Argonne National Lab., Idaho Falls, Idaho). May 1961. Contract W-31-109-eng-38. 21p.

Critical studies were made with a simulated, large, dilute power reactor having uranium carbide as fuel. The uranium in the core was 30.7% enriched, and the atomic ratio of uranium to carbon was 0.946. The critical mass was 503.01 kg U^{235} and the critical volume 574.47 liters. Central reactivity coefficients, effective fission cross-section ratios, heterogeneity effects, reactivity worth of distributed materials, foil irradiations, and the average prompt neutron lifetime were measured. Multigroup calculations using the Yiftah, Okrent, and Moldauer cross-section set overestimated k for the critical configuration by 4.7%. (auth)

154.

7042 TWO PLUTONIUM-METAL CRITICAL ASSEMBLIES. G. A. Jarvis, G. A. Linenberger, J. D. Orndoff, and H. C. Paxton (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 8, 525-31(1960) Dec.

Two plutonium-metal critical assemblies were studied at the Pajarito site in Los Alamos. Part I describes Jezebel, the bare plutonium assembly, and gives its observed characteristics along with a few comparisons with enriched-uranium systems. Part II covers Popsy, a plutonium core in a thick normal-uranium reflector. As Popsy was relatively inflexible—intended only for a preliminary survey—its experimental program was much less complete than that of Jezebel. (auth)

155.

17476 (LA-2044(Del.)) PLUTONIUM-METAL CRITICAL ASSEMBLIES. G. A. Jarvis, G. A. Linenberger, and H. C. Paxton (Los Alamos Scientific Lab., N. Mex.). May 1956. Declassified with deletions May 13, 1960. Contract W-7405-Eng-36. 34p.

The two plutonium-metal critical assemblies that were studied at Pajarito Site are Jezebel, bare plutonium, and Popsy. A plutonium core in a thick normal uranium reflector. These assemblies and their properties are described. (auth)

156.

21895 CRITICAL EXPERIMENTS FOR THE PRELIMINARY DESIGN OF THE ARGONNE HIGH FLUX REACTOR, PART B. J. O. Juliano, C. N. Kelber, and K. E. Plumlee (Argonne National Lab., Ill.). Trans. Am. Nuclear Soc., 4: No. 1, 110-11(June 1961).

157.

2066 UCRL-5369(Pt.2)
California. Univ., Livermore. Lawrence Radiation Lab. CRITICAL MEASUREMENTS ON NEAR-HOMOGENEOUS, BeO-MODERATED, ORALLOY-FUELED SYSTEMS. Fred A. Kloverstrom and Donald E. Kraft. July 11, 1960. 18p. Contract W-7405-eng-48. OTS.

A series of BeO-moderated oralloy-fueled graphite-reflected reactors are described. Perturbations were introduced by inserting various materials into basic reactor configurations and observing their effect on the critical dimensions of the reactor. The reactors had fuel densities that corresponded to atomic BeO/U^{235} ratios of 247 and 493. (auth)

158.

6593 (ORNL-3016(p.71)) HOMOGENEOUS CRITICAL ASSEMBLIES OF 2% U^{235}-ENRICHED UF_4 IN PARAFFIN. J. T. Mihalczo and J. J. Lynn (Oak Ridge National Lab., Tenn.).

Additional critical experiments were performed with a homogeneous mixture of 2% U^{235}-enriched UF_4 and paraffin, the $H:U^{235}$ atomic ratio for these later experiments being 501. The results were combined with those of earlier experiments, in which $H:U^{235}$ ratios of 195, 294, and 404 were used, to present plots of minimum critical masses and volumes. (auth)

159.

6595 (ORNL-3016(p.73-6)) CRITICAL PARAMETERS OF BARE AND REFLECTED 93.4 wt. % U_{235}-ENRICHED URANIUM METAL SLABS. J. T. Mihalczo and J. J. Lynn (Oak Ridge National Lab., Tenn.).

The critical parameters of slabs of 93.4 wt. % U^{235}-enriched uranium metal were measured in a series of neutron multiplication experiments. The effect of Plexiglas as a neutron reflector was determined, and a limited number of measurements with graphite and beryllium reflectors were made. The data were extrapolated to give 2.4 in. as the thickness of an infinite, unreflected critical slab and 0.6 in. as the thickness of an infinite critical slab reflected by 6 in. of Plexiglas. (auth)

160.

18649 (CF-61-4-33) NEUTRON MULTIPLICATION EXPERIMENTS WITH ENRICHED URANIUM METAL IN SLAB GEOMETRY. J. T. Mihalczo and J. J. Lynn (Oak Ridge National Lab., Tenn.). Apr. 10, 1961. 32p.

Critical thicknesses of uranium metal slabs enriched to 93.15% in the U^{235} isotope were obtained by a technique involving source-neutron multiplication counting. Subcritical assemblies of metal slabs were constructed to within, in most cases, 95% of critical mass, and the resulting reciprocal multiplication curves extrapolated to critical thicknesses. Slab dimensions ranged from 5 × 5 in. to 25 × 25 in., and thicknesses of infinite slabs were extrapolated from the data. Plexiglas, in thicknesses from 0 to 6 in., beryllium, and AGOT graphite served as neutron reflectors. Previous work with uranium-Plexiglas lattices was extended to lattice densities of 0.33 and 0.50, the latter being the limiting density under the conditions of the experiment. (auth)

161.

15077 MINIMUM CRITICAL DIMENSIONS FOR WATER SOLUTIONS. C. B. Mills (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng., 9: 377-90(Mar. 1961).

By use of the Los Alamos transport code, a parametric set of criticality conditions for one-dimensional geometries of light water solutions of the fissionable materials U^{233}, U^{235}, and Pu^{239} was determined. Minimum critical dimensions for slabs and cylinders and critical radius, mass, and volume for spheres as a function of solution concentrations (kg/l) and H/X atomic ratio are shown for bare and light water reflected solutions. Results of experimental studies for critical dimensions are given to support the study. (auth)

162.

21487 ANALYTICAL AND EXPERIMENTAL RESULTS FOR A COMPACT, ZIRCONIUM HYDRIDE MODERATED CRITICAL ASSEMBLY. L. L. Moss, C. M. Podewelts, R. L. Randall, J. L. Shapiro, and R. E. Wimmer (Atomics International, Canoga Park, Calif.). Trans. Am. Nuclear Soc., 4: No. 1, 53-4(June 1961).

163.

20051 (LAMS-2537) CORRELATIONS OF EXPERIMENTAL AND THEORETICAL CRITICAL DATA. COMPARATIVE RELIABILITY, SAFETY FACTORS FOR CRITICALITY CONTROL. Hugh C. Paxton (Los Alamos Scientific Lab., N. Mex.). Mar. 1961. Contract W-7405-eng-36. 54p.

This report supplements LAMS-2415.

Correlations of computed and experimental critical data are discussed. The scope is limited to fissile systems that may be approximated by simple descriptions. The methods used to adjust data to uniform conditions are outlined. The requirements for computation where the values are intended to substitute for experimental data in nuclear safety guidance are described. Computational methods using DSN and multigroup diffusion techniques are discussed. Both computed and experimental data are given for the following: critical masses of bare spheres of uranium with various moderators, critical masses and volumes of homogeneous water-moderated uranium spheres, core-density exponents for water reflected-water moderated U^{235} or Pu^{239} spheres, critical diameters of infinite cylinders of homogeneous water-moderated uranium, critical thicknesses of infinite slabs of homogeneous water-moderated uranium, critical masses of delta-phase plutonium mixtures with water and Plexiglas, influence of Pu^{240} on critical mass of water-moderated plutonium spheres, critical diameters of infinitely-long cylinders and critical thickness of infinite slabs of homogeneous water-moderated Pu^{239}, critical volumes of U–C–H_2O systems, and critical masses of bare spheres of uranium diluted with other metals and graphite. (M.C.G.)

164.

16649 (P-1713(RAND)) A DISCUSSION OF THE CORRELATION OF CRITICAL CONDITIONS FOR BARE HOMOGENEOUS REACTORS. Benjamin Pinkel and George B. W. Young (RAND Corp., Santa Monica, Calif.). June 4, 1959. 25p.

For presentation at the American Nuclear Society Annual Meeting, Gatlinburg, Tennessee, June 15-17, 1959.

An investigation was conducted to determine whether simple parameters could be found which could correlate the critical conditions for bare homogeneous reactors. In connection with this, a study was made of the results of an 18-group analysis which covered a group of reactors using Be, BeO, C, D_2O, Li^7H, and H_2O as moderators, over a wide range of moderator-to-uranium (U-235) mole ratios. The correlation was applied to the results of several criticality experiments. (B.O.G.)

165.

6604 CRITICAL REFLECTOR THICKNESSES FOR SPHERICAL U^{233} AND Pu^{239} SYSTEMS. E. A. Plassmann and D. P. Wood (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 8, 615-20(1960) Dec.

The thickness of reflectors required for critical configurations with spheres of U^{233} and Pu^{239} were estimated from multiplication measurements of nearly critical assemblies. Reflector materials employed were uranium enriched in U^{235}, normal uranium, beryllium, and tungsten alloy. Correction of the experimental data was attempted to give "idealized" dimensions, i.e., a solid core in intimate contact with its reflector material. (auth)

166.

2065 LA-267
Los Alamos Scientific Lab., N. Mex.
MULTIPLICATION OF NEUTRONS IN SMALL SPHERES OF ACTIVE MATERIAL. C. Richman. Apr. 7, 1945. Decl. Aug. 2, 1960. 18p. OTS.

An experiment is described in which a source of neutrons is surrounded by a sphere of active material. The resulting counts when the sphere is in place and removed are compared. The ratio of the two rates is a measure of the reactive properties of the sphere. Results are compared with those obtained by other workers. (auth)

167.

4594 (WAPD-TM-244) NUCLEAR ANALYSIS OF THERMAL REFLECTED CYLINDRICAL HOMOGENEOUS CRITICAL ASSEMBLIES. G. P. Rutledge and P. A. Kantorczyk. Nov. 1960. 21p. Contract AT-11-1-GEN-14. OTS.

Twenty-three small, cylindrical, UO_2F_2 aqueous solution, water-reflected, critical assemblies were analyzed utilizing few-group diffusion theory with the one-dimensional code WANDA. The fast-group constants were obtained using the MUFT code which computes the flux spectrum in a material of prescribed composition and buckling. The cross sections are then averaged over this spectrum to obtain the MUFT constants. Maxwellian thermal spectrum constants for the slow group were obtained with the SOFOCATE code. Also, three large, $UO_2(NO_3)_2$ aqueous solution, cylindrical critical assemblies were analyzed for comparison. The purpose of the analysis was to make several direct comparisons of critical experiments with analytical models in order to determine whether these models can be utilized to calculate criticality for assemblies that are not complicated with heterogeneous structural materials. The general use of these analytical models for fuel handling criticality calculations was evaluated. (auth)

168.

21744 THE REACTIVITY (K_{eff}) OF THE TERNARY SYSTEM ZIRCONIUM-WATER-URANIUM-235. G. P. Rutledge and F. A. Dobbe (Westinghouse Electric Corp., Pittsburgh). Trans. Am. Nuclear Soc., 4: No. 1, 21-2 (June 1961).

169.

9667 (RFP-213) PLEXIGLAS REFLECTED ASSEMBLIES OF PLUTONIUM. C. L. Schuske, D. F. Smith, and C. L. Bell (Dow Chemical Co., Rocky Flats Plant, Denver). Jan. 10, 1961. 7p. Contract AT(29-1)-1106.

A compilation is presented of critical extrapolations of subcritical neutron multiplication measurements made on assemblies of plutonium metal tamped with Plexiglas. In addition to this compilation, recent data are reported on a 20- by 20-in. slab of metal tamped with Plexiglas. A simple empirical equation of the form $(\sqrt{A}-\sqrt{A_0})(h-h_0)=C$ was found which fits the data and predicts the infinite slab and cylinder dimensions. (auth)

170.

32670 (TID-3306) CRITICALITY. A Bibliography of Unclassified Literature. Raymond L. Scott, comp. (Office of Technical Information Extension, AEC). Mar. 1961. 179p.

A total of 1145 references to unclassified reports and published literature is presented on calculations of critical parameters for various reactor fuels and moderators, critical and exponential experiments, and nuclear safety criteria for processing, handling, and storage of fissionable materials. Author, subject, and report number-availability indexes are provided. (auth)

171.

26664 (NP-10247(p.218-22)) CRITICALITY STUDIES OF PROJECT PHOENIX. R. Shankar Singh (India. Atomic Energy Establishment, Trombay). p.218-22 of PROCEEDINGS OF THE SYMPOSIUM ON LOW ENERGY NUCLEAR PHYSICS, 1961, BOMBAY, FEBRUARY 27-MARCH 2, 1961.

A study is made of the criticality conditions at the fuel reprocessing plant of the Canada India Reactor. The various stages in the fuel processing operation, whose purpose is to concentrate and recover the Pu^{239} produced in the reactor, are described. The criticality aspects of each stage are considered. (T.F.H.)

172.

31491 20% ENRICHED UO_2-GRAPHITE FUEL DISKS FOR SEMI-HOMOGENEOUS CRITICAL ASSEMBLY. R. Ueda (Japan Atomic Energy Research Inst., Tokyo). E. Sugimoto, K. Watanabe, and T. Ogura. Atompraxis, 7: 337-43(Sept. 1961). (In English)

About 25500 fuel disks of 20% enriched UO_2-graphite for the Semi-Homogeneous Experimental Facility in JAERI were fabricated. In preparing these disks, satisfactory quality controlling with respect to homogeneity and accuracy was carried out. Special care was also taken to prevent the generation of radioactive dust, which resulted in success from the viewpoint of accountability and safety. These fuel disks were charged into the critical assembly at JAERI, and criticality was successfully attained with little deviation from the calculated value. (auth)

173.

6601 CRITICAL MASSES OF CYLINDERS OF PLUTONIUM DILUTED WITH OTHER METALS. D. P. Wood, C. C. Byers, and L. C. Osborn (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 8, 578-87(1960) Dec.

Critical masses of 6-in. dia. cylinders of plutonium diluted in various volume ratios with steel, aluminum, thorium, air, and depleted uranium were determined. The cores were in 0, 2, 4.5, or 7.5-in. thick uranium reflectors and in 2, 4.5, or 7.5-in. thick thorium reflectors. From these data, the equivalent spherical critical masses of plutonium were estimated. An appendix gives reflector savings of undiluted plutonium for a variety of materials. (auth)

174.

11912 (ANL-6212) SOME PHYSICS CALCULATIONS ON THE PERFORMANCE OF LARGE FAST BREEDER POWER REACTORS. S. Yiftah and D. Okrent (Argonne National Lab., Ill.). Dec. 1960. Contract W-31-109-eng-38. 32p.

Critical mass, spectrum, breeding ratio, and coolant removal coefficients were calculated for a series of large Pu–U^{238}-fueled sodium-cooled fast-breeder power reactors, using a new 16-group cross-section set based in part on recent microscopic cross-section measurements. The parameters studied include reactor size, plutonium isotopic content, and type of structural material. Reactors cooled with Pb–Bi eutectic and those containing U^{233}–Th fuel were examined. (auth)

1962

175.

28603 (AEC-tr-5180) DESCRIPTION OF THE RACHEL ATOMIC PILE. (Translation). P. Bourgeois, C. Clair, R. Comte, and J. J. Long (France. Commissariat à l'Energie Atomique, Paris). June 1961. Includes Paper: NEUTRON PROPERTIES OF THE RACHEL CRITICAL REACTOR. C. Clair. 16p.

A critical experiment with high-speed neutrons and utilizing Pu metal is described. The information is presented in sections on the pile, core, reflector, mechanisms, containers, instrumentation, and implantation. (J.R.D.)

176.

18689 (TID-3571) FAST NEUTRON PULSE REACTORS AND FAST NEUTRON CRITICAL ASSEMBLIES. A Literature Search. William E. Bost, comp. (Division of Technical Information Extension, AEC). Apr. 1962. 20p.

A bibliography on fast neutron pulse reactors and fast neutron critical assemblies is presented. The 122 references are to reports, journal articles, and other published literature. (M.C.G.)

177.

933 (HW-69273). CALCULATED CRITICAL PARAMETERS FOR SLIGHTLY ENRICHED URANIUM RODS IN LIGHT WATER. C. L. Brown (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Apr. 1961. Contract AT(45-1)-1350. 48p.

Critical parameters for slightly enriched uranium rods in light water are presented. The parameters include minimum critical mass, critical slab thickness, critical cylinder diameter, critical volume, and minimum critical mass per unit area. The variables are water-to-uranium volume ratio, uranium rod diameter, and U^{235} enrichment. The calculations are based on a semi-empirical method for calculating material buckling derived from a correlation of theory and experimental measurements. The calculated results at 2 and 3.063% U^{235} are within 3% of the experimental measurements. (auth)

178.

7792 A REVIEW OF CRITICALITY DATA OBTAINED BY EXPERIMENTAL METHODS. D. Callihan (Oak Ridge National Lab., Tenn.). p.139-72 of "Criticality Control in Chemical and Metallurgical Plant." Paris, European Nuclear Energy Agency, 1961. (In English)

Experimental data obtained for the purpose of determining criticality parameters of fissionable materials outside reactor environments are surveyed, with emphasis on U^{233}, U^{235}, and Pu^{239}. The data on U^{235} are mainly in the enrichment ranges <5 and >90%. The data on U^{233} and Pu^{239} are restricted to metallic units and aqueous solutions within a small range of chemical concentrations. The assemblies studied are assumed to be spherical. The operating conditions assumed are those expected in chemical and metallurgical processing plants. (T.F.H.)

179.

24407 REACTIVITY MEASUREMENTS IN THE AETR CRITICAL ASSEMBLY. S. G. Carpenter and V. E. Kistler (Atomics International, Canoga Park, Calif.). Trans. Am. Nuclear Soc., 5: No. 1, 68-9(June 1962).

180.

26528 NEUTRON PROPERTIES OF RACHEL CRITICAL ASSEMBLY. C. Clair (Commissariat à l'Énergie Atomique, France). p.321-5 of "Physics of Fast and Intermediate Reactors. Vol. I." Vienna, International Atomic Energy Agency, 1962. (In French)

The critical sizes were determined experimentally and compared with the calculated results. The effect of nonfissile structural materials was studied. (auth)

181.

18315 (HW-71666) NUCLEAR SAFETY AND CRITICALITY OF PLUTONIUM. E. D. Clayton and W. A. Reardon (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Nov. 1961. Contract AT(45-1)-1350. 86p.

The data and techniques which were used to evaluate the criticality problems in typical Pu processing plants are presented. Recommendations concerning the reliability of the data, and techniques of control are also discussed. (J.R.D.)

182.

18158 (HW-72586(p.54-65)) CRITICAL MASS EXPERIMENTS WITH PLUTONIUM–NITRATE SOLUTIONS. E. D. Clayton, W. A. Reardon, et al. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Critical mass experiments were continued with plutonium concentrations ranging from 36.3 g/l at 3.3M HNO_3 with a 4-in. concrete reflector to 238.0 g/l at 5.21M HNO_3 with a 1.0-in. paraffin reflector. The corrections for reflector savings and nitrate ion concentration were derived from the experiments, and the experiments reduced to equivalent, bare critical systems. Data for evaluating the reflector savings of stainless steel were obtained. The worth of stainless steel as a reflector in combinations of stainless steel and paraffin were estimated. A coefficient was derived relating the nitrate–ion concentration to the critical volume. (M.C.G.)

24416 CRITICAL MASS EXPERIMENTS WITH PLUTONIUM NITRATE SOLUTIONS. E. D. Clayton, R. C. Lloyd, and W. A. Reardon (General Electric Co., Richland, Wash.). Trans. Am. Nuclear Soc., 5: No. 1, 75-6(June 1962).

183.

2699 THE EFFECT OF CYLINDRICAL AIR VOIDS IN A HIGHLY ENRICHED WATER-MODERATED CORE. P. N. Cooper, W. M. Cooper, and K. Firth (Associated Electrical Industries ltd., Aldermaston, Berks, Eng.). J. Nuclear Energy, Pts. A & B. Reactor Sci. and Technol., 15: 86-90 (Oct. 1961).

Measurements were made in a light-water-moderated, highly-enriched, uranium-fueled, sub-critical facility of the variation in critical mass with diameter of a central air void. These measurements indicated that there would be no transient increase in reactivity if a central void were to collapse radially. Other measurements on the slow flooding of a 3 in. diameter central air void showed that there was a slight addition of reactivity in the early stages, reaching a maximum of about 0.1 per cent, but that a completely flooded void was much less reactive than the empty void. Two-group diffusion theory calculations for the initial and final states indicated that the results of the measurements were not affected by small changes in the enrichment of the uranium. (auth)

184.

12711 THE EFFECTS OF BERYLLIA REFLECTOR ELEMENTS ON A HIGHLY ENRICHED LIGHT-WATER MODERATED REACTOR. P. N. Cooper, W. M. Cooper, K. Firth, and A. J. Salmon (Associated Electrical Industries Ltd., Aldermaston, Berks, Eng.). Reactor Sci. and Technol., J. Nuclear Energy, Pts. A & B, 16: 43-4(Jan. 1962).

The performance and use of beryllium oxide reflector elements are described with respect to improvements made in a water moderated reactor utilizing the elements. Since beryllia does not absorb neutrons as markedly as water it may be used for filling the vacant spaces in the lattice plates, thereby giving the neutron currents more facility. Since beryllia is also a better reflector than water, there will also be a decrease in critical mass, hence, increasing the actual radiation intensities in the core. Results are given for a subcritical reactor showing the comparisons with and without beryllium oxide reflector elements. (N.W.R.)

185.

9572 (CRRP-1064) MEASUREMENTS MADE IN THE POOL TEST REACTOR DURING THE PERIOD FROM NOVEMBER 1957 TO JUNE 1958. D. S. Craig and G. L. Keech (Atomic Energy of Canada Ltd., Chalk River, Ont.). Nov. 1961. 57p. (AECL-1407)

A comparison was made of the measured and calculated critical masses for a water-reflected core, a graphite-reflected core, and a graphite-reflected core with an internal water thermal column. Measured values are given of the worth of the absorber rods in the first of these assemblies, and of one rod over a limited range in the second assembly. Values are also given for the void coefficients and $r\sqrt{T/T_0}$ for several positions in these assemblies. A determination of the effective delayed neutron fraction is discussed. (auth)

186.

21842 CRITICAL EXPERIMENT WITH GRAPHITE REFLECTOR IN THE VVRS "EWA" REACTOR. M. Czerniewski and P. Szulc (Inst. of Nuclear Research, Polish Academy of Sciences, [Warsaw]). Nukleonika, 7: 81-8(1962). (In Russian)

Experimental data concerning critical mass, control-rod worths, and reactivity effects of a fuel assembly and graphite expellers were obtained. (auth)

187.

24410 CRITICAL STUDIES WITH ZPR-III FOR THE ENRICO FERMI FAST REACTOR CORE B. Thomas A. Doyle (Atomic Power Development Associates, Inc., Detroit) and Allen L. Hess. Trans. Am. Nuclear Soc., 5: No. 1, 71(June 1962).

188.

1195 (UCRL-6504) **PRELIMINARY RESULTS OF HIGH-TEMPERATURE BARE U^{235}–C CRITICAL ASSEMBLY MEASUREMENTS.** Reinald G. Finke (California. Univ., Livermore. Lawrence Radiation Lab.). June 6, 1961. Contract W-7405-eng-48. 8p.

The influence of temperature on the critical buckling of bare graphite assemblies with various carbon-to-uranium-235 molar ratios has been measured. A range from 1185:1 to 21,690:1 was covered, for 45 to 1205°F. Preliminary results indicate that the fractional rate of change of critical buckling with core temperature varies monotonically with C/U^{235} ratio by a factor of five over the factor-of-eighteen range in gross C/U^{235} ratio. This quantity appears to approach asymptotically a value near 2%/100°F at very high C/U^{235} ratios. (auth)

189.

3856 (ORNL-3139 (p.131-2)) CRITICAL PARAMETERS OF SOLUTIONS OF U^{235}-ENRICHED URANYL NITRATE IN CYLINDRICAL CONTAINERS. J. K. Fox (Oak Ridge National Lab., Tenn.).

The critical dimensions of cylindrical Al vessels containing enriched solutions of $UO_2(NO_3)_2$ are determined. At 93.15 wt % U^{235}, the solvent is a D_2O-H_2O mixture containing 70.1 wt % D_2O. At 92.6 wt % U^{235}, the solvent is H_2O. The cylinder diameters vary from 24 to 51 cm, and the critical heights from 19 to 72 cm. The data are taken for unreflected, $B-D_2O$-reflected, and H_2O-reflected cylinders. (T.F.H.)

190.

11186 (ANL-6494) CRITICAL STUDIES OF A FAST REACTOR CORE CONTAINING DEPLETED URANIUM AND SODIUM AS DILUENTS. (APR-III Assembly 36). J. M. Gasidlo, J. K. Long, and R. L. McVean (Argonne National Lab., Idaho Falls, Idaho). Jan. 1962. Contract W-31-109-eng-38. 23p.

Critical studies were performed with a metallic, fast reactor core designed to investigate the effects of replacing highly absorbing U^{238} diluent with high-scattering, low-absorbing sodium diluent. The fuel was 15.7 wt % enriched U^{235} and the core contained 18.2 vol % sodium and 12.68 vol % stainless steel. The experimental program was designed to measure the effect of the material replacement on spectral indices, which consisted of the standard fission ratios, foil irradiations, and a large number of central reactivity coefficients. Other measurements included the Rossi-α, radial fission traverses, and edge reactivity worths of a few samples. (auth)

191.

2701 PLEXIGLAS- AND GRAPHITE-MODERATED PLUTONIUM ASSEMBLIES. A. Goodwin, Jr. and C. L. Schuske (Dow Chemical Co., Rock Flats Plant, Denver). J. Nuclear Energy, Pts. A & B. Reactor Sci. and Technol., 15: 120-9 (Oct. 1961). (RFP-202)

The critical masses of plutonium-Plexiglas and plutonium-graphite mixtures were determined. The experimental data were derived from sub-critical neutron multiplication measurements on cylindrical assemblies containing these mixtures. The atomic ratios of moderator-to-fuel ranged from H: Pu = 0 to ~100 and C: Pu = 0 to ~10. Theoretical values of critical masses were derived for water- and graphite-moderated systems of plutonium using 16-group transport and diffusion calculations. The atomic ratios of moderator-to-fuel ranged from H: Pu = 0 to ~1000 and Cu: P = 0 to ~35,000. (auth)

192.

26533 STATUS OF COMPUTATIONAL AND EXPERIMENTAL CORRELATIONS FOR LOS ALAMOS FAST-NEUTRON CRITICAL ASSEMBLIES. G. E. Hansen (Los Alamos Scientific Lab., N. Mex.). p.445-55 of "Physics of Fast and Intermediate Reactors. Vol. I." Vienna, International Atomic Energy Agency, 1962. (TID-13176). (In English)

Direct correlations among data are presented to indicate possibilities for correlations with computations. Sensitivities of computed spectra and critical sizes to neutron-transport models and arithmetic approximations are presented for typical assemblies to help establish computational detail. Comparisons between experiment and prediction include spectral indices, critical sizes, neutron lifetimes, and delayed-neutron fractions. (L.N.N.)

193.

32261 (LAMS-2744) REFLECTOR SAVINGS OF MODERATING MATERIALS ON LARGE DIAMETER U(93.2%) SLABS. G. E. Hansen, D. P. Wood, and B. Peña (Los Alamos Scientific Lab., N. Mex.). June 1962. Contract W-7405-eng-36. 27p.

Critical mass measurements of 15.0 and 21.0-inch diameter U(93.2%) cylinders unreflected and reflected on one and/or two faces by graphite and the hydrogenous materials water, polyethylene, paraffin, and lucite are reported. (auth)

194.

30016 (JAERI-1032) MEASUREMENTS AND CALCULATIONS FOR GRAPHITE-MODERATED ENRICHED-FUEL CRITICAL ASSEMBLY, SHE. K. Inoue, M. Iizumi, S. Kobayashi, and K. Sumita (Japan Atomic Energy Research Inst., Tokyo). 1962. 25p.

To understand the reactor physics in enriched-fuel, graphite-moderated thermal reactors theoretical and experimental investigations were performed. A critical assembly was used to obtain the criticality data. An approach

of criticality calculations was developed, and it is demonstrated that the procedure is feasible in the limited area such as in the case of enriched-fuel, graphite-moderated small thermal reactors. The agreement between the theoretical and experimental criticality data is good. The effect of chemical binding in the graphite is of interest in the thermal reactor. Results of theoretical calculations are also presented. (auth)

195.

9347 (JÜL-26-RE) UNTERKRITISCHE VERSUCHE FÜR DEN FRJ-1 (MERLIN-JÜLICH). (Subcritical Tests on FRJ-1 (Merlin-Jülich)). J. Jacquemin, H. Friedewold, and W. M. Cooper (Kernforschungsanlage, Jülich, Germany). Nov. 1961. 94p.

Subcritical tests were carried out for the FRJ-1 (Merlin-Jülich) in England. The critical mass was determined for the following nuclear arrangement: compact core; square, elongated, and U-shaped core; as well as a core with central air and water voids. For one series of core configurations, the efficiency of the coarse and fine control elements was determined. Further measurements referred to the determination of the temperature coefficients of the core and reflector, as well as to the distribution of the statistical weight and the void coefficients of reactivity in the compact core. The results of the measurements are given in tabular form. (tr-auth)

196.

30999 (JAERI-1034) CRITICAL EXPERIMENTS AND ANALYSIS OF HOMOGENEOUS HEAVY WATER SYSTEM. 20 PERCENT ENRICHED U–D_2O REFLECTOR. (Japan Atomic Energy Research Inst., Tokyo). Apr. 1962. 55p. (In Japanese)

An aqueous homogeneous critical facility was constructed. A series of critical experiments was performed on the system in which heavy water reflector surrounded heavy water solutions of uranyl sulfate with a U^{235} enrichment of 20%. Heavy water molecule to U^{235} atom ratio in the solutions ranged from 3600 to 800 depending on the core diameter. Space dependencies of thermal neutron spectra in these systems were studied by the integral method with Lu. Deviations of epithermal neutron spectra from 1/E distribution were also investigated by the cadmium ratio method with In, Au, Pd, and Co. In theoretical analysis of these systems, leakage of fast neutrons from the core and competition of the leakage with resonance absorption in the core are important factors. Therefore the resonance escape probability was defined rigorously and a multigroup model was applied. The group constants were determined by averaging over spectra which were calculated by the Greuling Goertzel approximation. Agreements between theoretical and experimental results are satisfactory except perturbation terms. Discrepancies in the effective multiplication factors do not exceed 1%. Spatial distributions of the thermal, in-resonance, and fast neutron fluxes are well reproduced by the present theory. An agreement of the theoretical cadmium ratios in the core with the experimental values indicates that the leakage of fast neutrons from the core is treated adequately. (auth)

197.

30137 (JAERI-1031) CRITICALITY CALCULATION OF THE JRR-3. Takumi Asaoka (Japan Atomic Energy Research Inst., Tokyo). Jan. 1962. 50p. (In Japanese)

A criticality analysis which is performed by using a new 3-group method is described. Comparison of the calculated results with earlier ones reveals that the values of the effective multiplication factor for the reactor coincide within less than 1 ~ 2%. Moreover, a comparison of the calculated effective multiplication factor value with the experimental one for the FR-2 shows that the effective multiplication factor for the JRR-3 can be estimated within an error of less than 1 ~ 2%. According to these considerations, allowing for an error of ±1.5% in the value of the effective multiplication factor calculated, it is determined that the JRR-3 will reach criticality when the heavy-water level is raised by $(2.05^{+0.35}_{-0.2})$m from the upper surface of the grid plate. In other words, the critical geometric buckling is expected to be $(4.8 \pm 0.5)m^{-2}$ and the critical mass of U^{235} in the core to be $(21.5^{+3}_{-2.5})$kg. (auth)

198.

24353 MEASUREMENTS OF REACTOR PARAMETERS IN SUBCRITICAL AND CRITICAL ASSEMBLIES: A REVIEW. I. Kaplan (Massachusetts Institute of Tech., Cambridge). Trans. Am. Nuclear Soc., 5: No. 1, 25(June 1962).

199.

1263 (CF-61-8-71) OAK RIDGE NATIONAL LABORATORY FAST BURST REACTOR: CRITICAL EXPERIMENTS AND CALCULATIONS. W. E. Kinney and J. T. Mihalczo (Oak Ridge National Lab., Tenn.). Aug. 24, 1961. Contract [W-7405-eng-26]. 27p.

Determinations of critical masses and void coefficients of reactivity on three unmoderated, unreflected assemblies and the critical masses of ten unmoderated, reflected assemblies are reported. Multigroup and one-group transport theory and perturbation theory calculations were performed to aid in predicting results of experiments with an unmoderated and unreflected research reactor capable of producing intense bursts of fast neutrons. A description of the Fast Burst Reactor is included along with a discussion of calculation methods. (J.R.D.)

200.

30001 (APAE-Memo-322) CORRELATION OF CRITICAL MASS DATA ON LIGHT WATER MODERATED, FULLY ENRICHED URANIUM, STAINLESS STEEL REACTORS. PART I. D. H. Lee (Alco Products, Inc., Schenectady, N. Y.). July 17, 1962. Contract AT(30-1)-2639. 52p.

Experimental data were collected on over 70 light water moderated, fully enriched uranium, stainless steel, critical cores. An equation for the critical mass of cores with a buckling of 0.007 cm^{-2} that is lineally dependent on stain-

less steel volume fraction and grams of B^{10} was compared with available critical experiments and found to yield reasonable results. A correlation method, relating buckling to $(\xi\Sigma_s/\Sigma_a)$ was found to fit the available experiments. (auth)

201.

27738 (HW-73116(p.101-7)) CRITICAL MASS EXPERIMENTS WITH PLUTONIUM NITRATE SOLUTIONS. R. C. Lloyd, E. D. Clayton, and W. A. Reardon (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Critical mass studies were made of plutonium nitrate solutions in a 14-in. dia stainless steel sphere. The Pu concentrations ranged from 30 to 231 g/l with acid molarities varying from 1.1 to 5.6 molar. The wall thickness of the spherical vessel is 0.044 in. Measurements were made with the vessel bare and reflected with ½ in. of paraffin, 1 in. of paraffin, 4 in. of concrete, and 10 in. of concrete. Four measurements were made with the sphere covered with 0.030-in. Cd sheet backed with reflector. Evaluations were made of the control rod worth at various concentrations. The data from the experiments are summarized. The additional experiments with paraffin reflectors were used to obtain improved estimates of the reflector savings of this material. Estimates for the reflector savings of the various reflectors were obtained by first correcting the critical volumes of the bare spheres for the effect of the stainless steel shell by 14 ml/mil of stainless steel; then by comparing equivalent spherical radii, a reflector savings for the vessel wall and added reflector could be calculated from the measured volumes. The equivalent sphere volume for those cases in which the vessel was subcritical when full was taken as the extrapolated critical volume; the equivalent sphere radius for a partially filled sphere was estimated on the basis of equal surface-to-volume ratios for equivalent systems. These reflector savings are given for three concentrations of Pu. Control rod worth was evaluated during the critical approaches by means of multiplication measurements. The control rod consists of a stainless steel tube which is 1 in. in OD with a 0.065-in. thick wall; this tube can be inserted directly into the solution to within about 2 in. of the sphere bottom. A plot of control rod worth versus Pu concentration is included. The worth varies from about 60 ml of solution at high concentrations to about 560 ml at low concentrations. The control rod worth in terms of Pu appears to have a nearly constant value of about 15 grams over the entire range of concentrations measured, essentially independent of the H:Pu ratio. (auth)

202.

3813 (ANL-6403) FAST REACTOR SHAPE FACTORS AND SHAPE-DEPENDENT VARIABLES. W. B. Loewenstein and G. W. Main (Argonne National Lab., Ill.). Nov. 1961. Contract W-31-109-eng-38. 29p.

Existing experimental data on the variation of reactivity with core geometry are reviewed. Four typical fast neutron systems are analyzed to predict: (1) the variation of critical mass with cylindrical core geometry (core and reflector composition are held fixed); (2) the reactivity worth of fuel at the radial core boundary as a function of cylindrical core geometry; and (3) the geometric variation of heat removal parameters; these include the ratio of: (a) Maximum power density to average power density in the core. (b) Maximum power density to average radial power density in the core. (c) Total reflector power to total core power. The absolute values of all of these parameters are determined by the core and reflector compositions of the four systems. These were chosen to simulate typical constituents of interest to reactor analysis. Two systems represent a typical fast reactor and a typical fast critical experiment. The other two systems represent compositional combinations of the two basic systems. The results of the analyses show that the significant geometric variation is in items (2) and (3b). Item (1) is almost constant for small variations near the optimum geometric configuration. Outside of this range, the variation of critical mass with core geometry is pronounced. A most significant result shows that the ratio of the spherical critical mass to the minimum cylindrical critical mass (shape factor), for fixed core and reflector composition, depends primarily on core composition. The composition of the thick reflector has a lesser effect on this ratio which was found to increase with core density. The two-dimensional calculations are interpreted and analyzed on the basis of one-dimensional concepts. Reflector savings are calculated for spherical and cylindrical systems. The more exact reflector savings determinations are compared with more approximate calculations. It is found that the approximate determinations are qualitatively correct and show correct trends. However, the more detailed and accurate analytical techniques are required for precision comparison between theory and experiment. An interesting correlation between critical mass and core surface area is demonstrated. It was found that, in the range of interest, the critical mass depends almost linearly upon the surface area. The same linear dependence approximates all the systems studied. (auth)

203.

24397 THE INFLUENCE OF CORE SHAPE ON FAST REACTOR CRITICALITY. W. B. Loewenstein and G. W. Main (Argonne National Lab., Ill.). Trans. Am. Nuclear Soc., 5: No. 1, 61-2(June 1962).

204.

26522 INTEGRAL EXPERIMENTS ON FAST SYSTEMS OF PLUTONIUM, URANIUM AND THORIUM. J. J. McEnhill and J. W. Weale (Atomic Weapons Research Establishment, Aldermaston, Berks, Eng.). p.253-62 of "Physics of Fast and Intermediate Reactors. Vol. I." Vienna, International Atomic Energy Agency, 1962. (In English)

Two types of integral experiments undertaken to provide experimental checks on nuclear data and methods of calculation are described. A cylindrical assembly of material having a point source of DD or DT neutrons at its center is discussed. The neutron flux and reaction rates in the cylin-

der material are measured as a function of position. Results are compared with calculations. Data for natural uranium and thorium are presented. A low-power experimental reactor of simple regular geometry and approximately homogeneous composition is described. Holes can be provided as required for the insertion of counters or perturbation samples. The core composition is adjusted by the addition of a moderator to provide a neutron spectrum that tests the nuclear data in the energy region of interest. Measurements of the neutron-energy spectrum and various reaction rates and perturbation cross sections as a function of position in the reactor are made, and the results are compared with computer calculations. Results are presented for a system consisting of a core of U^{235} and graphite surrounded by a natural-uranium reflector. (auth)

205.

27739 (HW-73116(p.108-20)) LIMITING CRITICAL CONCENTRATIONS FOR A PLUTONIUM NITRATE SOLUTION AND FOR A URANIUM-235 SOLUTION. R. H. Masterson, V. I. Neely, and T. J. Powell (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Experiments were conducted to determine the limiting concentration (the concentration for which k_∞ = unity) of aqueous plutonium nitrate and uranyl fluoride solutions. For the Pu experiment, the solutions were contained in stainless steel tanks. Because of the difficulty in evaluating the effect of the stainless steel on the measurements, the uncertainty in the measured limiting concentration is approximately ±1 g/l. The experimental data were corrected for the effects of the Pu^{240}, Pu^{241}, and nitrate present to give a value of 8.4 ± 1 g/l as the limiting concentration for Pu^{239} in an aqueous solution. Foil irradiations were made using Au, Cu, and Pu^{239} foils. Cd ratios of 4.83 for the Au and 14.89 for the Cu were obtained. The results of the Pu-U foil irradiations indicate an effective neutron temperature of about 390°K, whereas the physical temperature of the solution was 299°K. The 93% enriched UO_2F_2 measurement was made using containers of Al, which had no measurable effect on the results. The experimental data give a value of 12.94 ± 0.03 grams of U/l (12.05 ± 0.03 g/U^{235}/l) as the limiting concentration of the solution used in the measurement. Au foil irradiations gave a Cd ratio of 9.6 in the center of the system. (auth)

206.

26534 RECENT ADVANCES AND PROBLEMS IN THEORETICAL ANALYSES OF ZPR-III FAST CRITICAL ASSEMBLIES. D. Meneghetti (Argonne National Lab., Ill.). p.457-87 of "Physics of Fast and Intermediate Reactors. Vol. I." Vienna, International Atomic Energy Agency, 1962. (In English)

Theoretical analyses of critical assemblies studied experimentally on the Argonne Zero Power Facility, ZPR-III, are compared with measured data. Discrepancies between calculated and experimental values are examined with results from previous assemblies for systematic trends. Effects of modifications in the cross-sections and in the calculational procedures in reducing the discrepancies are discussed. Calculation of reactivity changes, flux variations, and detector-response variations resulting from the heterogeneities caused by various intra-drawer configurations of fuel and diluent plates within the drawers of the critical assemblies are compared with experimental data. Transport methods for such thin slab cell configurations are compared. Improvements in the calculational methods for these cells are suggested; a simple multi-group approximation method, amenable to hand calculation, is described. The sensitivities of reactivity and inhours as functions of asymptotic reactor periods are examined for fast-reactor compositions. Results of calculations of effective delayed-neutron fraction, inhours per cent k_{eff}, and prompt- and delayed-neutron worths are given for fast assemblies. (auth)

207.

2405 CRITICALITY OF LOW ENRICHMENT U^{235} IN HYDROGEN. John Mihalczo (Oak Ridge National Lab., Tenn.) and C. B. Mills. Nuclear Sci. and Eng., 11: 95-6 (Sept. 1961).

Six experiments were performed on the uranium density in paraffin and water at 0.5 to 3.1 gm/cm^3. The H : U^{235} atomic ratios varied from 200 to 1000. Calculations were performed, using two methods (NSA, 15: 15077). Results, represented tabularly and graphically, gave k_{eff} = 1.00 ± 0.01. Critical radii of unreflected spheres were calculated, and effects of an infinite paraffin reflector on the 2% enrichment calculations are given. (L.N.N.)

208.

19673 THE INFINITE NEUTRON MULTIPLICATION CONSTANT OF HOMOGENEOUS HYDROGEN-MODERATED 2.0 wt %-U^{235}-ENRICHED URANIUM. John T. Mihalczo (Oak Ridge National Lab., Tenn.) and Victor I. Neeley. Nuclear Sci. and Eng., 13: 6-11 (May 1962).

The infinite medium neutron multiplication factor, k_∞, of a mixture of 92.1 wt % UF_4 and 7.9 wt % paraffin was measured both in the Physical Constants Testing Reactor at the Hanford Atomic Products Operation and in critical experiments at the Oak Ridge National Laboratory. The density of the mixture is 4.5 g/cc and the U^{235} enrichment of the uranium is 2.0 wt %, resulting in an H : U^{235} atomic ratio of 195. The values of k_∞ from the two experiments are 1.216 ± 0.013 and 1.197 ± 0.015, respectively. In the analysis of the critical experiments a two group model was assumed for the nonleakage probability. The neutron age to thermal was determined from buckling perturbation measurements as 43.1 ± 3.4 cm^2. The critical buckling was measured to be (4344 ± 65) × 10^{-6} cm^{-2}, the bare extrapolation distance 2.7 ± 0.3 cm, and the fast fission factor 1.039 ± 0.004. Within the experimental error, the values of k_∞ from critical experiments at ORNL and from the PCTR at HAPO agree. (auth)

209.

761 DESCRIPTION AND ANALYSIS OF SOME "OVER-MODERATED" CRITICAL EXPERIMENTS FROM 68–550°F. G. H. Miley, R. D. Gillespie, J. W. Pollard, and J. A. Bistline (Knolls Atomic Power Lab., Schenectady, N. Y.). Trans. Am. Nuclear Soc., 4: No. 2, 299-300 (Nov. 1961).

210.

18316 CRITICALITY OF LOW ENRICHMENT URANIUM IN HYDROGEN. C. B. Mills and G. I. Bell (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng., 12: 469-73 (Apr. 1962).

Critical masses of homogeneous water-moderated assemblies containing low enrichment uranium are given. The calculations were made using the multigroup DSN code with eighteen energy groups. Effective absorption cross sections for U^{238} were computed with the infinite mass and narrow resonance approximations. The calculations were compared with various experiments and rather good agreement was found. The results are presented as a parametric survey for U^{235}/U atom ratios from 0.014 to 0.300 and for all H/U^{235} ratios for which criticality is possible. The decrease in critical radius with an infinite water reflector is also shown. A bare homogeneous system with $U^{235}/U < 0.010$ cannot be made critical at any H/U^{235} ratio. (auth)

211.

28441 REFLECTOR MODERATED REACTORS. C. B. Mills (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng., 13: 301-5 (Aug. 1962).

The complete spatial separation of moderator and uranium fuel bearing regions are shown by experiment to result in critical reactors with low critical mass and relatively uniform fissioning density. Studies of several of these experiments to establish the accuracy of a numerical method of calculation (SNG) for this class of problems show good correspondence between theory and experiment. This method is then used for a useful survey of critical mass and U^{235} atomic density as a function of geometry for the best moderators, D_2O and Be. (auth)

212.

6106 (NEPA-1100) COMPARATIVE CRITICAL CONDITIONS IN SIMPLE NUCLEAR REACTORS. A. O. Mooneyham (Fairchild Engine and Airplane Corp. NEPA Div., Oak Ridge, Tenn.). Aug. 1, 1949. Decl. July 18, 1961. 48p. Contract [W-33-08-ac-14801(16250)].

The physical and nuclear properties of simple critical reactors are compared. Those considered include homogeneous, gas-cooled, enriched, cylindrical, and room temperature reactors containing hydrogen, beryllium oxide, beryllium carbide, graphite, or iron. Comparisons are made of U^{235} investments, critical sizes and other parameters and their interpretations. (J.R.D.)

213.

26525 THE EPITHERMAL CRITICAL EXPERIMENTS. H. A. Morewitz and S. G. Carpenter (Atomics International, Canoga Park, Calif.). p.287-93 of "Physics of Fast and Intermediate Reactors. Vol. I." Vienna, International Atomic Energy Agency, 1962. (In English)

The present phase of the advanced-epithermal-thorium-reactor program consists of integral-reactor-physics experiments designed to provide neutron-cross-section information at 10 Mev to 1 kev range. Nine multi-region, slow-fast, pseudospherical critical assemblies of the honeycomb, split-table type are studied. Three assemblies were run. The outer driver-decoupler region drives an interior U^{233}-Th fueled spherical test region whose neutron-flux spectrum is successively degraded by increasing the graphite moderator to fuel ratio. A square-wave oscillator experiment defines the central reactivity worths of 40 small samples of different materials to 10^{-6} Δk for each assembly. Intercalibrated artificial neutron sources are oscillated to determine central neutron importance functions. The spectra are obtained by fission-counter measurements using calibrated foils of different thresholds and by a Li^6-solid-state-counter sandwich spectrometer. A digital computer routine is used to compile all measurements into a self-consistent library of spectrum-averaged cross sections. (auth)

214.

18622 (UCRL-6729) REACTIVITY EFFECTS OF VARIOUS REFLECTORS ON NEAR-HOMOGENEOUS, BeO-MODERATED, ORALLOY-FUELED SYSTEMS. John R. Morton, Frederick J. Shon, Thomas F. Weirich, and Loren L. Gardner (California. Univ., Livermore. Lawrence Radiation Lab.). Jan. 2, 1962. Contract W-7405-eng-48. 18p.

Measurements were made of the reactivity changes caused by reflectors of various materials on BeO-moderated, U^{235}-fueled subcritical systems with a moderator-to-fuel ratio of 247. The reflector materials included aluminum, steel, copper, brass, type 304 stainless steel, nickel, René 41, Hastelloy R-235, and BeO. The pulsed neutron technique was used. The data are tabulated in terms of the subcritical time constant as a function of reflector thickness and surface density. (auth)

215.

10807 (HW-66882) K_∞ OF THREE WEIGHT PER CENT U^{235} ENRICHED UO_3 AND $UO_2(NO_3)_2$ HYDROGENOUS SYSTEMS. V. I. Neeley, J. A. Berberet, and R. H. Masterson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Sept. 1961. Contract AT(45-1)-1350. 82p.

The value of the infinite multiplication constant (k_∞) was determined as a function of the hydrogen-to-uranium (H:U) atomic ratio for 3.64 weight per cent U^{235} enriched UO_3 and $UO_2(NO_3)_2$ hydrogen moderated homogeneous systems. The work was done to evaluate nuclear safety of and establish operational limits for slightly enriched homogeneous sys-

tems and was performed in the Hanford Physical Constants Testing Reactor. The amount of thermal neutron absorber, commonly referred to as poison, necessary to reduce the value of k_∞ of these systems to unity was also obtained directly from these studies. This value is of interest for processing reactor fuels in which soluble poisons would be used for criticality control. Curves of the value of k_∞ as a function of the H : U atomic ratio for 3.04 weight per cent U^{235} enriched UO_3 and $UO_2(NO_3)_2$ are shown. The curve of k_∞ versus H : U was determined by a least squares fit to the experimental data. Values of the amount of thermal poison necessary to reduce k_∞ to unity as a function of the H : U ratio are tabulated. (auth)

216.

13857 (HW-70310) MEASUREMENT OF MULTIPLICATION CONSTANT FOR SLIGHTLY ENRICHED HOMOGENEOUS UO_3–WATER MIXTURES AND MINIMUM ENRICHMENT FOR CRITICALITY. V. I. Neeley and H. E. Handler (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Aug. 21, 1961. Contract AT (45-1)-1350. 77p.

The minimum U^{235} enrichment required for criticality in an infinite homogeneous system of uranium trioxide (UO_3) and water has been determined from measurements of k_∞ in the Hanford Physical Constants Testing Reactor, (PCTR) This is the enrichment for which $k_\infty \leq$ unity in an aqueous homogeneous system. The experiments consisted of k_∞ measurements at U enrichments of 1.006, 1.070, and 1.159 weight per cent U^{235} for H-to-U atomic ratios in the range of 3.5–7.5. The minimum enrichment required for criticality in an infinitely large system with a homogeneous UO_3-water mixture was found to be $1.034^{+0.010}_{-0.009}$ wt % U^{235}. (auth)

217.

31987 (K-1550) THE EFFECT OF URANIUM DENSITY ON THE SAFE U-235 ENRICHMENT CRITERION. C. E. Newlon (Oak Ridge Gaseous Diffusion Plant, Tenn.). Oct. 11, 1962. Contract W-7405-eng-26. 20p.

A review of the density effect on the criticality of low enrichment U indicated that the nuclear homogeneity of fissionable materials, considered under the safe enrichment criterion of 0.95% U^{235}, should be established for nuclear safety. It appears that a condition of nuclear homogeneity will be maintained if the U density is less than or equal to 3.2 g U/cm^3, or for densities ranging from 3.2 g U/cm^3 up to a maximum of 18.9 g U/cm^3. One dimension of the U fuel is less than or equal to 0.02 in. (auth)

218.

12325 (PG-Report-97(2nd Ed.)) AN EMPIRICAL CORRELATION OF THE EXPERIMENTAL DATA ON HOMOGENEOUS CRITICAL ASSEMBLIES OF URANIUM AND HYDROGEN OF ALL ENRICHMENTS. B. G. Owen (United Kingdom Atomic Energy Authority. Production Group, Risley, Lancs, England). Nov. 1960. 37p.

The original correlation on highly enriched uranium–hydrogen critical systems is now extended to all enrichments. By using three empirical equations and the one-group buckling relations, the physical size of any uranium–hydrogen homogeneous critical assembly in simple geometry can be predicted. The predictions are compared with experiment where experimental data are available. The experimental data are of fissile solutions and intimate wax compacts using uranium compounds. The correlation, at present, must be confined to those conditions until more experimental data are available. The derived reflected spherical mass and volume is displayed in universal terms. The reflected infinite cylinder and slab dimensions are displayed graphically. (auth)

219.

24358 THE LOS ALAMOS FAST CRITICAL PROGRAM. Hugh C. Paxton (Los Alamos Scientific Lab., N. Mex.). Trans. Am. Nuclear Soc., 5: No. 1, 29-30 (June 1962).

220.

12699 THE CRITICALITY OF HOMOGENEOUS SOLUTIONS AND MIXTURES. I. THE CRITICALITY OF URANIUM-WATER SYSTEMS. M. A. Perks (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). Progr. in Nuclear Energy, Ser. IV, 3: 228-43(1960).

A comparison of two-group diffusion results with experimental data on spherical homogeneous uranium-water systems is made. Using extrapolation lengths derived from this comparison, predictions are made of the critical dimensions of infinite cylinders and slabs for all H/U ratios using the equal buckling conversion. (auth)

221.

12700 THE CRITICALITY OF HOMOGENEOUS SOLUTIONS AND MIXTURES. II. THE CRITICALITY OF PLUTONIUM-WATER SYSTEMS. M. A. Perks, F. R. Charlesworth, and D. E. J. Thornton (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). Progr. in Nuclear Energy, Ser. IV, 3: 244-52(1960).

A correlation of existing experimental data on the criticality of plutonium metal-water mixtures was made. The gap in data between effective plutonium densities of 0.1 g/cm^3 and 19.6 g/cm^3 was filled in on the basis of reasonable assumptions as to the shape of the curves. This derived basic data for spherical masses and volumes was extended by a combination of empirical methods and one-group buckling equations to the estimation of cylindrical and slab critical dimensions. Some suggestions for an experimental program to help fill in the uncertain parts of the curves are made. (auth)

222.

8372 (XDC-60-12-156) CORRELATION OF ZPR-III SPHERICAL FAST CRITICAL ASSEMBLIES. R. E. Reid and S. K. Breslauer (General Electric Co. Aircraft Nuclear Propulsion Dept., Cincinnati). Dec. 1960. Contracts AF 33(600)-38062 and AT(11-1)-171. 48p.

A theoretical correlation is presented of the experimental measurements made on Assembly 6F and Assembly 9A of the ZPR-III series of fast critical experiments. Both B. H. Duane's variational optimum formulation of transport theory and a standard formulation of diffusion theory were used in this correlation. Variational optimum transport theory calculations and diffusion theory calculations were made on critical mass, relative fission distributions, and U^{238} to U^{235} fission ratios. Methods were derived from variational optimum transport theory based on first-order perturbation theory to calculate the Rossi-alpha decay constant and central reactivity coefficients of materials and correlation calculations were made on these parameters. Both the transport theory and diffusion theory calculations were quite accurate for the prediction of critical mass, fission distribution, and fission ratios. The transport theory calculations were slightly more accurate, however. The correlation of the Rossi-alpha constant was quite good, while the prediction of the central reactivity coefficients was satisfactory for some materials and unsatisfactory for others. Careful analysis of all the results of the variational optimum transport theory calculations yielded considerable insight into the inaccuracies in the calculational procedure and nuclear data. This correlation study highlighted the applicability and accuracy of variational optimum transport theory for detailed fast critical experiment correlation. It also showed that diffusion theory provided adequate critical mass predictions for fast reactors in the size range of the ZPR-III assemblies. (auth)

223.

14242 (FTD-TT-61-124) STUDY OF CRITICAL PARAMETERS FOR REACTOR SYSTEMS (COLLECTION OF ARTICLES). Translated from p.3-117 of "Issledovaniya Kriticheskikh Parametrov Reaktornykh Sistemy" Sbornik Statei (A publication of the State Publishing House of Literature in the Field of Nuclear Science and Tech., Moscow, 1960). 182p.

A total of 9 papers is included, 5 of which separate abstracts were prepared. The remaining 4 papers which were previously abstracted in NSA included information on critical masses in Be, graphite, and spherical reactors, and critical masses of water mixtures of U and Pu compounds. (J.R.D.)

224.

9349 (RFP-248) INDUSTRIAL CRITICALITY MEASUREMENTS ON ENRICHED URANIUM AND PLUTONIUM. PART II. C. L. Schuske, C. L. Bell, G. H. Bidinger, and D. F. Smith (Dow Chemical Co. Rocky Flats Plant, Denver). Jan. 10, 1962. Contract AT(29-1)-1106. 20p.

A series of neutron multiplication measurements made on sub-critical systems containing enriched uranium and plutonium is presented. These measurements involve both aqueous and metal systems. (auth)

225.

5858 (XDC-57-6-129) ANALYSIS OF SM-I, SM-II-2 AND SM-II-5 CRITICAL EXPERIMENTS. Florence Silverstein (General Electric Co. Aircraft Nuclear Propulsion Dept., Cincinnati). Jan. 16, 1957. Decl. June 9, 1961. 42p.

Three configurations (SM-I, SM-II-2, SM-II-5) of the Solid Moderator Reactor were analyzed for the purpose of correlating theory with experiment on criticality, critical mass, gross radial power distribution, fine radial power distribution, and activation flux sensors. A method of determining the bare equivalent diameter is described for reactors having only a side reflector. Results of using first-order perturbation theory for determining the worth of off-center holes and boral strips are given. Satisfactory correlation between theory and experiment was obtained with ANP design procedures when the U^{238} in the fuel, a more recent value of $\nu = 2.46$, and improved thermal averaging of cross sections were taken into account. (auth)

226.

21258 CRITICALITY INVESTIGATIONS ON U^{235}-IMPREGNATED CELLULOSE SHEETS. W. Stumm and H. Groll. Atomwirtschaft, 7: 215-17(Apr. 1962). (In German)

The criticality of arrangements of fissionable material with various moderators cellulose is considered. The proposals of the International Parkinson Committee (ICP) essentially amount to the achievement of a self-limitation of the document situation in nuclear organizations of every kind by prescribing the exclusive use of writing paper containing a homogenous distribution of U^{235}. In the piling-up of U^{235} impregnated cellulose sheets in the form of NSP (Nuclear Standard Paper) it is necessary to distinguish between the criticality of the U^{235} embedded in the sheets acting as a moderator (Nuclear Criticality) and the conventional or bureaumetric criticality. Various safety considerations are to be observed in the case of the building up of critical masses of NSP. The use of blotting pads made of cadmium and gadolinium paper weights permits especially high piles. The NSP fuel element arises from the concentration of 300 NSP in up to 2 mm thick cladding, for which cardboard is used in place of zirconium. A subcritical assembly results from the piling up of such fuel elements in filing cabinets. This situation must be avoided because of the danger of a critical state through changes in the geometry, e.g., through tearing of the files. The same safety regulations apply to the store of useless fuel material (Archives). (auth)

227.

24413 CRITICAL EXPERIMENTS PERFORMED ON VERY SMALL DIAMETER PRESSURIZED WATER REACTOR CORE MOCKUPS. A. J. Svasek, S. N. Kemp, L. D. Walthousen, and W. J. McCool (Alco Products, Inc., Schenectady, N. Y.). Trans. Am. Nuclear Soc., 5: No. 1, 73-4 (June 1962).

228.

28548 (TID-7623(p.41-4)) FAST POWER REACTOR EXPERIMENTS IN ZPR-3 WITH METALLIC AND CERAMIC FUELS. F. W. Thalgott (Argonne National Lab., Ill.).

The fast power reactor critical facility, the Zero Power Reactor-III (ZPR-III), is a flexible facility for the study of the important reactor-physics characteristics of fast reactor assemblies. Experiments in ZPR-III with uranium oxide and uranium carbide power-breeder-reactor fuels are discussed. Critical masses obtained with the ZPR-III for these assemblies are compared with calculated results. The core compositions are given for the assemblies. (N.W.R.)

229.

10806 (DC-59-11-158) ANALYSIS OF LIVERMORE GRAPHITE MODERATED CRITICAL EXPERIMENT AND COMPARISON WITH EXPERIMENTAL DATA. E. R. White (General Electric Co. Aircraft Nuclear Propulsion Dept., Cincinnati). Oct. 20, 1959. Contract AT(11-1)-171. 7p.

Results of nuclear analysis on a graphite moderated, critical experiment are presented. It was found that critical masses in this system could be calculated at elevated temperatures with significant correlation with experimental data. (auth)

230.

934 (LAMS-2579) CRITICAL MASS MEASUREMENTS OF Oy AND Pu CORES IN SPHERICAL ALUMINUM REFLECTORS. D. P. Wood and B. Pena (Los Alamos Scientific Lab., N. Mex.). June 1961. Contract W-7405-Eng-36. 12p.

The thicknesses of spherical aluminum reflectors required to bring approximately 38 kg oralloy and 11 kg plutonium spheres to delayed critical were determined. The critical specifications are given. (auth)

1963

231.

19551 (ANL-6690) CRITICAL STUDIES OF URANIUM-STEEL AND URANIUM-STEEL-SODIUM FAST REACTOR CORES. (ZPR-III ASSEMBLIES 32 and 33). P. I. Amundson, W. Gemmell, J. K. Long, and R. L. McVean (Argonne National Lab., Idaho Falls, Idaho). Feb. 1963. Contract W-31-109-eng-38. 26p.

Critical studies of two fast reactor cores are described: one contains uranium and steel; the other contains uranium, steel, and sodium. Experimental results are given for fission ratio, central and edge reactivity coefficients, fuel bunching, average prompt neutron lifetime, and distributed worth measurements. (auth)

232.

40280 (UCRL-6901) ROOM TEMPERATURE CRITICAL MEASUREMENTS ON THORIUM-LOADED, GRAPHITE-MODERATED, ORALLOY-FUELED SYSTEMS. G. M. Benson and R. H. Fox (California. Univ., Livermore. Lawrence Radiation Lab.). June 30, 1962. Contract W-7405-eng-48. 29p.

The comparison of uranium-235 fueled, graphite-moderated critical assemblies with and without thorium is critically studied. The assembly for the experiment consisted of a lattice of thin oralloy fuel foils regularly spaced between 0.5- and 1-inch-thick graphite moderator blocks stacked in horizontal planes. Additional foils of thorium were procured having the same dimensions as the oralloy foils. The experimental procedure was simply to restack some of the previously reported uranium-235-graphite assemblies with the addition of thorium foils at regular intervals and to determine the new critical height. Thus, part of the fast leakage of neutrons in the uranium-235-graphite core is replaced with resonance absorption by the thorium. Comparison shows that the thermal spectra agree to ½%. 3% less neutrons reach thermal in the thorium-loaded case. In this case, the flux is higher in the intermediate energy range because of the reduced leakage. As a result, uranium absorption is increased slightly. (N.W.R.)

233.

4294 (ANL-6629) CORE A CRITICAL STUDIES FOR THE ENRICO FERMI ATOMIC POWER PLANT ON ZPR-III. C. E. Branyan (Argonne National Lab., Ill.). Oct. 1962. Contract W-31-109-eng-38. 110p.

A critical studies program for the Enrico Fermi Atomic Power Plant was run with the ZPR-III fast critical facility. The objectives of this program included determination of the U enrichment required for criticality, the effect of minor variations in core and blanket composition, reactivity coefficients, control and safety rod characteristics, power distribution, spectral indices, and the reactivity worth and wave shape of the oscillator rod. The experimental program was separated into two phases. The first phase in-

volved investigations of a clean assembly, which was a simplified and homogenized core and blanket geometry constructed for ease of experimental manipulation and analysis. The second phase involved experiments on the engineered, or as-designed, core. This assembly included such engineering details as control and safety rod channels, core end gaps, and a precise reconstruction of the core outline. This provided detailed information on worths of control rods and fuel subassemblies, power distribution, and the effect of variations in core and end-gap dimensions. The application of critical experiment data to the determination of the Enrico Fermi reactor characteristics has established the U^{235} enrichment for the fuel alloy, worths of fuel subassemblies, and the B^{10} enrichment for the control and safety rods. In addition, material-substitution experiments and fuel-worth measurements have provided the parametric data for the determination of the net temperature coefficient of the Enrico Fermi reactor. (auth)

234.

41723 (CEA-2274) ALECTO, EXPERIENCE DE CRITICALITE SUR UNE SOLUTION DE PLUTONIUM. RESULTATS EXPERIMENTAUX, CUVE No. 1 (ϕ = 324 mm). ALECTO—Criticality Experiment on a Plutonium Solution. Experimental Results. Vessel Number 1 (ϕ = 324 mm). J. Bruna, J. P. Brunet, R. Caizergues, C. Clouet D'Orval, J. Kremser, J. Leclerc, and P. Verriere (France. Commissariat à l'Energie Atomique. Centre d'Etudes Nucléaires, Saclay). 1963. 38p.

ALECTO is a critical experiment intended for the neutronic study of homogeneous aqueous multiplying media. It essentially consists of a cylindrical tank, reflected or not, where can be made critical a solution of fissionable material fed into the tank from a geometrically subcritical storage. The studies effected on this assembly concern on one hand the determination of critical masses, and on the other hand the nuclear parameters used in neutron calculations. The container tested in the first series of experiments described is a cylindrical tank, 324 mm diameter with a convex bottom, water reflected on the sides and on the interior. The minimum critical mass of this tank was determined and was found to be $M_{c\,min}$ = 845 ± 7 g. The decay constant of prompt neutrons as a function of reactivity was determined by the pulsed neutron technique. At the critical state, it was found to be α_c = 73 ± 6 sec^{-1}. Furthermore, from the study of this tank, were derived a number of safety regulations for plutonium solutions. (auth)

235.

28453 (TID-7650(p.88-127)) EPITHERMAL THORIUM REACTOR PHYSICS. S. G. Carpenter (Atomics International. Div. of North American Aviation, Inc., Canoga Park).

The physics of U^{233}–Th epithermal systems was studied. The Epithermal Critical Assembly is described, and the composition and sizes of 9 cores are listed. The calculations of both predicted and experimental data were made u ing a multiregion, multigroup diffusion code. Reactivity measurements, neutron importance, spectral measurements, and criticality analysis are discussed. (M.C.G.)

236.

11459 (HW-SA-2432(Rev.)) CRITICAL MASS EXPERIMENTS WITH PLUTONIUM NITRATE SOLUTIONS. E. D. Clayton, R. C. Lloyd, and W. A. Reardon (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). May 23, 1962. Contract [AT(45-1)-1350]. 22p.

Criticality data obtained with a 14-in.-dia. stainless steel sphere filled with plutonium nitrate of various compositions and with various states of neutron reflection are presented. The data on the paraffin-reflected spheres were corrected for reflector savings and the experiments reduced to equivalent critical systems. The data were also corrected for acid molarity. (J.R.D.)

237.

38440 (TID-7653(Pt.I)(p.2-7)) CRITICALITY EXPERIMENTS AND CALCULATIONS FOR HIGH-TEMPERATURE, GRAPHITE-MODERATED ASSEMBLIES WITH SIMPLE GEOMETRY. A. G. Cole and R. G. Finke (California. Univ., Livermore. Lawrence Radiation Lab.).

The effect of temperature on reactivity of a bare, graphite-moderated, enriched-uranium assemblies over carbon-to-uranium-235 atomic ratio range, 583:1 to 21,690:1, and with critical temperatures ranging between 46 and 1110°F was experimentally determined. Buckling calculations were done for each assembly as critical at room temperature, and as overstacked to be critical at an elevated temperature. Calculations were carried out with the one-dimensional, neutron-diffusion code, 9-ZOOM with 18 energy groups. The difference between k_{eff} values for a high-temperature experiment, using room-temperature measurements and densities of both, provides a measure of the experimental excess reactivity at room temperature to be overcome by elevating the temperature. Since the effect of elevating the temperature is to decrease reactivity, a calculated k_{eff} for the high-temperature critical assembly lower than that for the room temperature critical assembly indicates an over-calculation of the temperature effect. The difference between such k_{eff} values, divided by the above experimental reactivity effect, provides a measure of the extent of the over-calculation. Since this contribution is a small fraction of the experimental temperature effect, the over-calculation must be due to nuclear parameters such as the neglect of Doppler broadening of resonance levels and the inadequacy of the free gas moderator model. (N.W.R.)

238.

3959 (TID-17240) PRECISE CRITICALITY DETERMINATIONS IN THE SOLID HOMOGENEOUS ASSEMBLY. F. Feiner, S. Weinstein, W. C. Oaks, and K. V. Cooper (Knolls Atomic Power Lab., Schenectady, N. Y.). [1962]. Contract [W-31-109-eng-52]. 11p.

To accommodate large metal fractions, but still retain the advantages of homogeneity, the Solid Homogeneous Assembly (SHA) was designed with the metallic elements and fuel as very fine powders dispersed in paraffin. A series

of reproducibility checks performed with the assembly demonstrated that rearranging or replacing major segments of the core leads to reactivity changes of less than 10¢. In addition to critical size determinations preliminary flux distribution and reactivity coefficient measurements were made. The calculated criticalities using the KARE and KLAG programs were compared with the experimental ones. It was found that calculations run approximately 2% high fairly consistently. (M.C.G.)

3743 PRECISE CRITICALITY DETERMINATION IN THE SOLID HOMOGENEOUS ASSEMBLY. F. Feiner, S. Weinstein, W. C. Oaks, and K. V. Cooper (General Electric Co., Schenectady, N. Y.). Trans. Am. Nucl. Soc., 5: 344-5(Nov. 1962).

239.

2159 (UCRL-6980) HIGH-TEMPERATURE BARE BeO CRITICAL EXPERIMENTS: GENERAL DESCRIPTION AND PRELIMINARY RESULTS. Reinald G. Finke (California. Univ., Livermore. Lawrence Radiation Lab.). June 29, 1962. Contract W-7405-eng-48. 23p.

Five high-temperature bare BeO critical experiments were conducted for three different gross BeO/U^{235} molar ratios: 280:1, 550:1, and 1100:1. For each ratio, at least two critical configurations were determined. Their corresponding critical temperatures ranged up to 1115°F. The buckling temperature coefficient at 500°F varies directly with BeO/U^{235} ratio from 0.26%/100°F for the lower ratio to 0.60%/100°F for the higher ratio. The contribution of thermal expansion to the buckling temperature coefficient was examined for different BeO-block arrangements in the graphite supporting box. A variation by a factor of three for different block positions is calculated. Within experimental error, the nuclear components of the BeO buckling temperature coefficients are the same as, and show the same dependence on moderator-to-uranium ratio as, those previously reported for graphite at C/U^{235} ratios about 2.3 times the BeO/U^{235} ratios. (auth)

240.

3960 (UCRL-6673) A COOLED-REFLECTOR, HOT-BeO-CORE CRITICAL EXPERIMENT. R. G. Finke, E. Goldberg, H. L. Reynolds, and G. J. Grammens (California. Univ., Livermore. Lawrence Radiation Lab.). Oct. 27, 1961. Contract W-7405-eng-48. 29p.

A series of critical experiments to investigate temperature-reactivity effects and temperature-power distribution interactions with a cooled graphite reflector on two sides of a BeO core is described. It was found that for a core with a gross BeO/U^{235} ratio of 556 to 1, the reactivity was not dependent on the graphite reflector temperature. The experiments were performed at the HOT BOX facility. (J.R.D.)

241.

4255 (UCRL-6329) SUMMARY REPORT ON A HIGH-TEMPERATURE BERYLLIUM-OXIDE CRITICAL EXPERIMENT. R. Finke, E. Goldberg, H. Reynolds, and G. Grammens (California. Univ., Livermore. Lawrence Radiation Lab.). Mar. 21, 1961. Contract W-7405-eng-48. 36p.

An experimental verification of multigroup calculations of critical dimensions as a function of temperature of a beryllium-oxide-moderated, enriched-uranium-fueled assembly was made using Hot Box, a high-temperature critical facility. A decrease in buckling of 2.5% was required to compensate for an increase in core temperature from 90 to 955°F for a bare equivalent core with an effective BeO density of 2.86 g/cm^3 and a gross BeO/U^{235} molar ratio of about 550 to 1. The reactivity change was within 5% of the change predicted by the Zoom multigroup, multiregion, one-dimensional diffusion code for an equivalent core configuration. An overall average temperature coefficient of reactivity of $-\$0.48/100°F was measured for the temperature range studied, and a nuclear temperature coefficient of $-\$0.34/100°F was deduced. (auth)

242.

18492 (ANL-6650) CRITICAL EXPERIMENTS ON THE JUGGERNAUT REACTOR. J. R. Folkrod and D. P. Moon (Argonne National Lab., Ill.). Dec. 1962. Contract W-31-109-eng-38. 36p.

The experiments are described in sections on approach to critical, control system evaluation, measurements of fuel worth, measurements of reactivity and other coefficients, flux measurements, power measurements, and measurements of fuel plate, control rod, and graphite temperatures. (J.R.D.)

243.

26895 (ORNL-3359) PRELIMINARY SOLUTION CRITICAL EXPERIMENTS FOR THE HIGH-FLUX ISOTOPE REACTOR. J. K. Fox, L. W. Gilley, and D. W. Magnuson (Oak Ridge National Lab., Tenn.). June 12, 1963. Contract W-7405-eng-26. 85p.

The design of the High-Flux Isotope Reactor (HFIR) was supported by a series of preliminary experiments performed at the Oak Ridge Critical Experiments Facility in 1960. The experiments yielded results describing directly some of the expected performance characteristics of the reactor and strengthened the calculational methods used in its design. The critical assembly, like the reactor, was of a flux-trap type in which a central 6-in.-dia column of H_2O was surrounded by an annulus of fissile material and, in turn, by an annular neutron reflector. The fuel region contained a solution of enriched uranyl nitrate in a mixture of H_2O and D_2O and the reflector was a composite of two annuli, the inner one of D_2O surrounded by one of H_2O. In most experiments the ends of the assembly were reflected by H_2O. Important results evaluate the absolute thermal-neutron flux to be expected in the design reactor and describe the flux distributions within

this type of assembly. It was also observed that the cadmium ratio along the axis of the assembly was about 100, showing that a highly thermal-neutron flux was truly developed in the trap. It was shown that reduction of the hydrogen density in the central water column to about 60% of its normal value increased the reactivity about 6% and that further hydrogen density reduction decreased the reactivity as the effect of the loss of neutron moderation dominated the effect of the increased coupling across the central column. These considerations are of importance to the safety of the reactor. Additional experiments gave values of the usual critical dimensions and explored the effects on both the dimensions and the flux distributions of changing the concentration of the uranyl nitrate solution, of changing the composition of the solvent, and of adding neutron-absorbing materials to the D_2O reflector. These changes were made to alter the neutron properties of the fuel solution over a range including those expected in the reactor itself. (auth)

244.

12768 (ORNL-3360(p.35-6)) CRITICAL EXPERIMENTS WITH MIXTURES OF AQUEOUS U^{235}-ENRICHED URANYL NITRATE SOLUTIONS AND BOROSILICATE GLASS RINGS. J. K. Fox and J. T. Thomas (Oak Ridge National Lab., Tenn.).

Several critical experiments were performed with mixtures of aqueous 92.6% U^{235}-enriched uranyl nitrate and borosilicate glass Raschig rings to provide information for nuclear safety applications. The mixtures were contained in 20-, 30-, and 48-in.-ID Al or stainless steel cylinders, and two concentrations of uranyl nitrate solutions were used, one containing 415 g of U/l and the other 279 g of U/l. The glass rings varied in size, the largest having a 1.52-in. ID, a 1.85-in. OD, and a 1.89-in. length. Their B content was 0.5 to 5.7 wt %. It was found that for solutions having a U concentration ≤415 g/liter, k_∞ will be less than 1 if the glass rings contain 5.7 wt % natural B and occupy 24.1 vol % of the mixture. For solutions having a U concentration of ≤279 g/liter, the corresponding values are 4.0 wt % natural B and 24.1 vol % glass. (auth)

245.

3740 CRITICAL EXPERIMENTS AND ANALYSIS ON SMALL, LOW ENRICHED, UO_2 CORES WITH DISCRETE BURNABLE POISON. O. Gallar and S. Visner (Combustion Engineering, Inc., Windsor, Conn.). Trans. Am. Nucl. Soc., 5: 341-2(Nov. 1962).

246.

3741 DESCRIPTION AND ANALYSIS OF CRITICAL EXPERIMENTS WITH ANNULAR MULTITUBE SUPERHEATER FUEL ELEMENTS. M. F. Gormley, C. O. Dechand, P. G. Klann, and S. Visner (Combustion Engineering, Inc., Windsor, Conn.). Trans. Am. Nucl. Soc., 5: 342-3(Nov. 1962).

247.

15431 ANALYSIS OF THE ENRICHED URANIUM-HEAVY WATER HOMOGENEOUS SYSTEM. Jitsuya Hirota, Satoru Katsuragi, Hideo Kuroi, Reiko Koinuma, and Yukio Ishiguro (Japan Atomic Energy Research Inst., Tokyo). Nippon Genshiryoku Gakkaishi, 5: 14-22(Jan. 1963). (In English)

A series of critical experiments was conducted on a system in which heavy water solutions of uranyl sulfate with a U^{235} enrichment of about 20% are surrounded by a heavy water reflector. The deuterium to U^{235} atomic ratios in the solutions ranged from 7,200 to 1,600. In theoretical analysis of the system, leakage of fast neutrons from the core and competition of the leakage with resonance absorption in the core are rather important. Therefore, the resonance escape probability was defined rigorously and a multi-group model was applied to fast and epithermal neutrons. The group constants were determined by averaging over a spectrum which was calculated by the Greuling-Goertzel approximation. The agreement between the theoretical and experimental results are satisfactory except those for perturbation terms. The discrepancies in the effective multiplication factors do not exceed 1%. The spatial distributions of thermal and In-resonance neutron fluxes are well reproduced by the present theory. Agreement of the theoretical cadmium ratios with experimental values indicates that the effects due to the leakage of fast neutrons from the core are treated adequately. (auth)

248.

42383 OPERATION OF A CRITICAL MOCK-UP FOR VVR-M. D. M. Kaminker, K. A. Konoplev, Yu. V. Petrov, and R. G. Pikulik (Ioffe Inst. of Physics and Tech., Leningrad). Vienna, International Atomic Energy Agency, 1963, Preprint SM-42/84, 20p. (CONF-133-31). (In Russian)

From International Atomic Energy Agency Symposium of Exponential and Critical Experiments with Calculations, Amsterdam, Sept. 1963.

Critical mock-up operations of the VVR-M reactor with a beryllium reflector and a thermal neutron flux 3×10^{14} n/cm^2 sec at 10 Mw(t) is described. A theoretical evaluation is given of the core configuration and its measurements. Calculations were made for various core configurations, and the results were compared with experimental data for minimum critical masses. Data are also given on safety techniques employed during critical operations. (tr-auth)

249.

19033 (NAA-SR-Memo-8195) CRITICALITY STUDY 15 w/o U–Zr ALLOY. N. Ketzlach (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). Feb. 4, 1963. Contract [AT(11-1)-Gen-8]. 53p.

Maximum safe batch sizes were determined for all process steps involved in the melting and casting operations, as well as in the subsequent fuel handling procedures, of 15 wt % U–85 wt % Zr fuel rods. Criteria for hydrided as well as unhydrided rods are presented. A method is presented for the determination of an equivalent fuel rod di-

ameter so that criticality data, developed for long rods can be used in establishing nuclear safety criteria for fuel rods of any length. Packaging criteria are presented for the safe storage and transport of the fuel materials in 30 × 30 × 30 in. birdcages as well as in 55-gal. drums. The criteria developed can be used to determine the maximum safe number of storage units in an array when each unit contains no more than 45% of the critical quantity of fuel in spherical geometry under optimum conditions of water moderation and reflection. This is independent of fuel composition. Thirty-three figures are given that have been used in the development of this criteria. (auth)

250.

30754 (NAA-SR-Memo-8195(Rev.)) CRITICALITY STUDY. 15 W/O U–Zr ALLOY. Norman Ketzlach (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). June 10, 1963. Contract [AT(11-1)-Gen-8]. 53p.

Maximum safe batch sizes were determined for all process steps involved in the melting and casting operations, as well as in the subsequent fuel handling procedures, of 15 wt.% U–85 wt.% Zr fuel rods. Criteria for hydrided as well as unhydrided rods are presented. A method is presented for the determination of an equivalent fuel rod diameter so that criticality data developed for long rods can be used in establishing nuclear safety criteria for fuel rods of any length. Packaging criteria are presented for the safe storage and transport of the fuel materials in 30 × 30 × 30 in. birdcages as well as in 55-gallon drums. The criteria developed can be used to determine the maximum safe number of storage units in an array when each unit contains no more than 45% of the critical quantity of fuel in spherical geometry under optimum conditions of water moderation and reflection. This is independent of fuel composition. Thirty-three figures are given that were used in the development of this criteria. (auth)

251.

42222 CRITICAL EXPERIMENTS AS A MEANS OF EXPOSING THE PHYSICAL CHARACTERISTICS OF A REACTOR. V. A. Kuznetsov. Vienna, International Atomic Energy Agency, 1963, Preprint SM-42/86, 13p. (CONF-133-33). (In Russian)

From International Atomic Energy Agency Symposium of Exponential and Critical Experiments with Calculations, Amsterdam, Sept. 1963.

Criticality studies were made of intermediate reactors in order to determine physical properties of certain reactor types. An analysis was made of experimental data on reflector efficiency, heterogeneity effects, reactivity coefficients, and neutron lifetimes, and ideas are forwarded on the feasibility and perspectives of experimental data utilization in developing reactor theory and calculation methods. (tr-auth)

252.

31081 (HW-77311(p.40-7)) CRITICAL EXPERIMENTS WITH PLUTONIUM NITRATE SOLUTIONS. R. C. Lloyd, E. D. Clayton, C. R. Richey, and S. R. Bierman (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Critical experiments were conducted with plutonium nitrate solutions in an 11.5-inch diameter stainless steel sphere. The measured volume of the 11.5-inch sphere was 12.95 liters; the vessel wall thickness was 0.049 inch. Criticality in the water reflected sphere was studied as a function of nitrate concentration. Plutonium concentrations were in the range of 51 to 431 g Pu/liter, with low acid molarity (less than 2 in most cases). The data obtained yield a curve of critical concentration versus nitrate, which may then be used to estimate the critical concentration for a homogenous plutonium water mixture in the absence of nitrate, or to evaluate the minimum mass of plutonium in the sphere. Experiments were also conducted to determine the effect of the stainless steel vessel wall on the criticality of the water reflected unit. Pulsed neutron source experiments were conducted concurrently with critical mass measurements. Based on the measured neutron lifetimes and the prompt decay rates observed at various solution levels in the 11.5-inch sphere, effective multiplication constants were determined as a function of the plutonium concentration and solution volume in the sphere. One of the purposes of the pulsed neutron source experiments is to examine the feasibility of using this technique for k_{eff} measurements on in-plant equipment. Multigroup diffusion theory calculations were used to compute criticality for some of the solutions as used in the experiments, with a comparison between theory and experiment being given in terms of k_{eff}. (auth)

253.

29826 (ANL-6713) CRITICAL STUDIES OF A SMALL URANIUM CARBIDE-FUELED REACTOR WITH A BERYLLIUM REFLECTOR. (ZPR-III ASSEMBLY 40). R. L. McVean, P. I. Amundson, G. S. Brunson, and J. M. Gasidlo (Argonne National Lab., Idaho Falls, Idaho). Apr. 1963. Contract W-31-109-eng-38. 19p.

A small, lightweight, uranium carbide-fueled reactor with a beryllium reflector surrounding the core was mocked up as Assembly 40. It was determined that the presence of beryllium in the axial and radial reflectors did not endanger the safe loading and operation of the critical assembly. The actual experiment consisted of determination of the critical mass, measurement of the reactivity coefficients for a large number of fissile and nonfissile materials, the performance of radial and axial fission traverses, and measurement of central fission ratios. The effectiveness of the radial beryllium reflector as a control mechanism was determined and the Rossi-alpha was measured. (auth)

254.

41724 (HW-77089) THE LIMITING CRITICAL CONCENTRATIONS FOR Pu^{239} AND U^{235} IN AQUEOUS SOLUTIONS. R. H. Masterson, J. D. White, and T. J. Powell (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 27, 1963. Contract AT(45-1)-1350. 30p.

Measurements were made with plutonium nitrate solutions and uranyl fluoride solutions (93.15% enriched UO_2F_2) to determine the concentration at which k_∞ equals unity (the limiting critical concentration) for each of these solutions. The limiting critical concentration for Pu^{239} in an aqueous solution occurred at an H : Pu ratio of 3392 ± 100; this limiting ratio corresponded to a concentration of 8.0 ± 0.3 g Pu^{239}/liter for the solutions used in the experiments. In conjunction with this plutonium measurement, the limiting critical concentration for U^{235} in an aqueous solution was measured; the result (12.05 ± 0.03 g U^{235}/liter) was compared to a similar result reported by the Oak Ridge National Laboratory. (auth)

255.

26057 EXPERIMENTS AND ANALYSIS OF WATER-REFLECTED, UNDERMODERATED ZIRCONIUM-HYDRIDE CRITICAL ASSEMBLIES. PART I. EXPERIMENTS. L. I. Moss (Atomics International, Canoga Park, Calif.). Trans. Am. Nucl. Soc., 6: 49-50(June 1963).

256.

20826 (UCRL-7262) COUPLING EFFECTS OF HAFNIUM CONTROL RODS IN A NEAR-HOMOGENEOUS, BeO-MODERATED, ORALLOY-FUELED SYSTEM. J. M. Piowaty, John R. Morton, III, and Loren L. Gardner (California. Univ., Livermore. Lawrence Radiation Lab.). Feb. 27, 1963. Contract W-7405-eng-48. 10p.

Two series of measurements were made to examine the coupling effects of Hf control rods. From two to five rods were used. The measurements were performed on a BeO-moderated, U^{235}-fueled system with a BeO/U^{235} molar ratio of 247. The pulsed neutron technique was used. (auth)

257.

32895 (HW-SA-2999) IDEALIZED PLUTONIUM DISSOLVERS AND THE "ALWAYS SAFE" CONDITIONS. W. A. Reardon and F. R. Czerniejewski (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). [May 1, 1963]. Contract [AT(45-1)-1350]. 11p. (CONF-39-21)

From American Nuclear Society Meeting, Salt Lake City, Utah, July 1963.

A study was made to determine if some combination of plutonium metal and solution possesses a smaller critical mass than either the fully reflected metal system or a fully dissolved system. Primary results of the calculations are shown in a figure with the critical total mass (metal plus solution) presented as a function of the critical volume (metal plus solution) for various assumed central masses of plutonium metal. The "always safe" envelope is shown as a tangent curve and the lower portion of the homogeneous curve. Critical total mass was also plotted against the critical solution concentration. It is shown that there are situations where the critical mass is smaller than the total mass when partially dissolved. (M.C.G.)

258.

41731 (HW-77871(p.46-53)) SUBCRITICAL EXPERIMENTS WITH PuO_2–POLYSTYRENE COMPACTS. C. R. Richey, J. D.White, and E. D. Clayton (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Criticality measurements on concentrated Pu solutions were simulated using PuO_2–polystyrene cubes with a H-to-Pu ratio of 15. Neutron multiplication measurements performed by arranging configurations of the cubes on halves of a split table that were then moved together gave the critical masses of the nonreflected and Lucite-reflected assemblies. An evaluation of control and safety rods, both poison insertion and fuel removal, showed the desirability of fuel removal type rods. The neutron spectrum for the PuO_2–polystyrene compacts was strongly epithermal. (D.C.W.)

259.

16941 (RFP-63) CRITICALITY MEASUREMENTS ON PLUTONIUM METAL PRELIMINARY TO THE DESIGN OF A MELTING CRUCIBLE. C. L. Schuske, M. G. Arthur, and D. F. Smith (Dow Chemical Co. Rocky Flats Plant, Denver). June 1, 1956. Contract AT(29-1)-1106. 12p.

A series of neutron multiplication measurements were made on assemblies of Pu metal (density 15.8 g/cm^3) in order that a safe melting crucible could be designed for charges of 10 kg or more. Suitable ingot shapes and masses that could be rolled into sheets were determined. Final charge limits must be determined by making in situ measurements with the actual production furnace when the furnace and crucible are completed. (H.G.G.)

260.

30749 (HW-76553(Rev.)) CRITICAL MASS CONTROL SPECIFICATION HOOD 9A. R. J. Sloat (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). May 31, 1963. Contract AT(45-1)-1350. 7p.

Nuclear safety in hood 9-A depends on controlling the mass and geometry of the plutonium compounds not confined by the processing equipment. It is specified that plutonium concentrations in plutonium oxalate slurries and filter cakes allowed to accumulate in volumes in excess of 4 liters should be limited to a maximum of 1.0g Pu/cc in the process equipment. It was concluded that the critical mass of nonconfined plutonium may exceed 7 kg for dry, unmoderated plutonium compounds. Moderation of a reflected 2.5-kg plutonium mass could make it critical in an active volume of about 8 liters. (M.C.G.)

261.

15433 NEUTRON SPECTRA AND BREEDING POTENTIAL IN SHR. Akira Tachibana, Shigeru Yasukawa, and Takeji Ise (Japan Atomic Energy Research Inst., Tokyo). Nippon Genshiryoku Gakkaishi, 5: 51-9(Jan. 1963). (In Japanese)

Fast and thermal neutron spectra in the core of semi-homogeneous reactor were calculated at atom ratios 2,000 to 10,000 for carbon/U^{233} and moderator temperature of 1,200 to 1,800°K. With the obtained spectra and η-233 values measured directly in experiments, effective η-f values and breeding ratios, which are weighted averages of entire core spectra, were obtained. The results are very similar with those of Chernick, et al., of BNL and the breeding ratios were over 1.15 even including Xe^{135} and Sm^{149} poisoning at neutron flux $10^{14}n/cm^2/sec$. It is concluded that solid fuel thermal breeders have possibilities to be realized, depending on the designs which should be very elaborated. (auth)

262.

19034 (ORNL-TM-499) CRITICAL MASS STUDIES. PART XIII. BOROSILICATE GLASS RASCHIG RINGS IN AQUEOUS URANYL NITRATE SOLUTIONS. J. T. Thomas, J. K. Fox, and E. B. Johnson (Oak Ridge National Lab., Tenn.). Feb. 6, 1963. Contract [W-7405-eng-26]. 11p.

The effect of commercially available borosilicate glass Raschig rings on the criticality of aqueous uranyl nitrate solutions enriched in U^{235} was investigated. The natural-B content of the glass varied from 0.5 to 5.7 wt %, and the volume of the vessel occupied by the glass ranged from 20.9 to 30%. Results from exponential experiments, using a critical layer of solution above the column of solution-ring mixture as a neutron source, provided estimates of the material buckling of the mixture as a function of solution concentration, B content of the glass, and the glass volume present. It was shown, for example, that the buckling is negative (i.e., $k_\infty < 1$) if glass containing 4 wt % B occupies more than 22% of the mixture volume, whereas the same concentration of glass containing only 0.5 wt % of B results in positive values of the buckling except for solutions more dilute than about 72 g of U per liter ($H : U^{235} \cong 380$). (auth)

263.

3744 EXPERIMENTAL AND ANALYTICAL RESULTS FOR A ZIRCONIUM/URANIUM UNPOISONED CRITICAL SLAB ASSEMBLY. James R. Tomonto (General Electric Co., Schenectady, N. Y.). Trans. Am. Nucl. Soc., 5: 345-6 (Nov. 1962).

1964

264.

28959 (ANL-6733) A TWO-ZONE FAST CRITICAL EXPERIMENT (ZPR-III ASSEMBLY 42). P. I. Amundson, R. L. McVean, and J. K. Long (Argonne National Lab., Idaho Falls, Idaho). Jan. 1964. Contract W-31-109-eng-38. 36p.

The study was conducted to determine what properties of a previously studied dilute power reactor could be duplicated in a smaller two-zone assembly. Reactivity measurements performed included determinations of plate-orientation worth, homogeneity corrections, reactor-segment worths, central reactivity coefficients, and radial worth distributions of axial columns of core materials. Spectral index determinations included: central fission ratios, nuclear track emulsion measurements, fission counter traverses, Rossi-alpha measurements, sodium activation, and natural and enriched uranium foil measurements. (auth)

265.

44925 (ANL-6923(p.14-45)) LIQUID-METAL-COOLED REACTORS. (Argonne National Lab., Ill.).

Studies of Assembly 45 in ZPR-III are described. Data on the critical material and physical parameters of ZPR-IX Assembly 4 are tabulated along with data on the central worth coefficients on several metallic and ceramic materials and separated W isotope samples. Investigations are reported on the properties of U–Pu–fissium alloys, Ti–V jacket materials, radiation effects on Nb–Zr-clad U–Pu–fissium alloys, fabrication and properties of fuel elements for zero-power reactors, fabrication of Doppler test elements, development of Pu–U–C fuels, corrosion of Croloy in high-temperature air, reactions of Cd–Mg–Zn alloys with U_3O_8 at 600°C, distribution of Pu between Mg–Zn alloys and molten chloride salts, head-end treatment of refractory fuels, and sodium coolant chemistry. Other work is reported on development and operation of EBR-II and FARET. (J.R.D.)

266.

13234 (CONF-187-35) BUCKLINGS, DISADVANTAGE FACTORS, AND δ^{28} MEASUREMENTS IN SOME UNDERMODERATED SLIGHTLY ENRICHED CORES. Q. L. Baird and A. R. Boynton (Argonne National Lab., Ill.). [1963]. Contract [W-31-109-eng-38]. 5p.

From American Nuclear Society Meeting, New York, Nov. 1963.

The critical U^{235} mass for light-water-moderated, fully-reflected, slightly-enriched, oxide-fuelled cores with H/U^{235} atom ratios of 1 to 5 were determined. Bucklings were obtained and compared with theoretical values. Calculations indicate that the buckling is lowered by about 3% for Al-clad vs unclad fuel. The U^{238}/U^{235} fission ratio was also measured and compared with theory. The disad-

vantage factor and the epicadmium/subcadmium U^{235} fission ratio were measured by the integral technique. The disadvantage factors disagree with calculations by Pennington. (D.C.W.)

267.

1229 (CEA-2309) PROSERPINE MASSES CRITIQUES EN URANIUM 235 ET ETALONS DE REACTIVITE. (PROSERPINE—Critical Masses with Uranium-235 and Reactivity Standards). J. Bertrand, J. P. Brunet, R. Caizergues, C. Clouet d'Orval, J. Kremser, and P. Verriere (France. Commissariat à l'Énergie Atomique. Centre d'Études Nucléaires, Saclay). 1963. 43p.

The critical mass for the PROSERPINE reactor was measured as a function of the solution height in the tank, with various solution concentrations. A reactivity standard was then obtained by a boron poisoning technique. The relation to the reactivity of control rod displacement, self-shielding of two control rods, and rise of the solution level over the critical level was determined. (auth)

268.

38304 (A/CONF.28/P/362) NEUTRON PHYSICAL CHARACTERISTICS OF U + Be AND U + BeO SYSTEMS. I. I. Bondarenko, V. P. Garin, R. K. Goncharov, et al. (U.S.S.R.). May 1964. 22p.

For near-homogeneous systems with highly enriched uranium, critical parameters, neutron spectra, reflector effects, dynamics, and control-component effectiveness at various positions were investigated experimentally for varying fissionable material concentrations and geometries. Multi-group methods of calculation for these assemblies were corrected on the basis of the experimental data. For heterogeneous lattices using natural and slightly enriched uranium, experiments included the investigation of the critical lattice parameters (K_∞ and its cofactors) and the space-energy distribution of the neutrons, using a mechanical selector. Contribution to K_∞ from (n,2n) and (γ,n) reactions on beryllium is assessed. For pure moderators, experimental results for neutron ages, square diffusion lengths, and diffusion constants are given. Calculation results showing the importance of the (n,2n) process in neutron multiplication are presented. (M.J.T.)

269.

37866 (A/CONF.28/P/84) CRITICALITY STUDIES. Denis Breton, Christian Clouet d'Orval, and Pierre Lecorche (France. Commissariat à l'Énergie Atomique, Paris). May 1964. 15p.

Critical masses of plutonium and uranium solutions in cylinders, annular vessels, and plates were measured under various conditions of concentration, reflection, and interaction. Physical measurements such as spectra and reactor noise were also carried out. Various computation methods (diffusion, transport, and Monte Carlo) were developed, and results compared with experiment. The studies were applied to safety design and control of industrial plants. (M.J.T.)

270.

42915 (CEA-R-2453) EXPERIENCE DE CRITICITE SUR UNE SOLUTION DE POLUTONIUM ET D'URANIUM ENRICHI A 90%. RESULTATS EXPERIMENTAUX ET CALCULS CONCERNANT LA CUVE N' 2 (ϕ = 300). (Criticality Experiment on a Solution of Plutonium and of 90% Enriched Uranium. Experimental Results and Calculations on Tank Number 2 (ϕ = 300 mm)). Jean G. Bruna, Jean-Paul Brunet, Christian Clouet D'Orval, and Jacques Kremser (France. Commissariat à l'Énergie Atomique. Centre d'Études Nucléaires, Saclay). June 1964. 26p.

Experiments on plutonium and 90% enriched uranium solutions were made in the ALECTO reactor with a tank of external diameter 300 mm. Various geometries were tested for variable concentrations of fissionable salts. The critical mass was studied as a function of the concentration in various reflector conditions (water, concrete, wood) and the experimental values were compared with calculated values. The effects of cadmium as a reflector and of the stainless steel tank were also studied. Measurements of β/τ, ratio of the effective fraction of delayed neutrons to the average lifetime of the neutrons in the reactor, were carried out. (auth)

271.

42914 (CEA-R-2452) PROSERPINE— PLUTONIUM-239. PROSERPINE— URANIUM-235. COMPARAISON DE RESULTATS EXPERIMENTAUX. (PROSERPINE—Plutonium-239. PROSERPINE—Uranium-235. Comparison of Experimental Results). Jean-Paul Brunet, Robert Caizergues, Christain Clouet D'Orval, Jacques Kremser, Jean Moret-Bailly, and Philippe Verriere (France. Commissariat à l'Énergie Atomique. Centre d'Études Nucléaires, Saclay). 1962. 86p.

The PROSERPINE homogeneous reactor consists of a tank, 25 cm dia and 30 cm high, surrounded by a composite reflector made of beryllium oxide and graphite. In this tank plutonium or 90% enriched uranium solutions, the fissile substances being in the form of a dissolved salt, can be made critical. In varying the concentration of the solution, critical masses were studied as a function of the level of the liquid in the tank. The minimum critical mass is 256 ± 2 gm for plutonium and 409 ± 3 gm for uranium-235. In the range of the critical concentrations studied, the neutronic properties of fissionable solutions of plutonium and enriched uranium were compared for identical geometries. (auth)

272.

16736 (HW-79413) DETERMINATION OF THE RELATIVE WORTH OF PLUTONIUM ATOMS TO U^{235} ATOMS IN HOMOGENEOUS UO_3–WATER MIXTURES. L. L. Carter (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 30, 1963. Contract AT(45-1)-1350. 11p.

The limiting enrichment of uranium for criticality and the limiting atomic ratios of Pu-to-natural uranium con-

taining 0, 5, 10, and 15% Pu^{240} were determined by Monte Carlo techniques for UO_3–water mixtures. The Monte Carlo results predict an infinite multiplication factor of unity for a 1.001 ± 0.026 wt % enriched UO_3–H_2O mixture. This compares favorably to the experimentally determined limiting critical enrichment of 1.034 ± 0.01 wt % U^{235} in the uranium. The Monte Carlo and multigroup calculations indicate that natural uranium UO_3–H_2O mixtures with a plutonium-to-uranium atomic ratio of less than 0.00155 ± 0.00012 should have a k_∞ of less than unity when no Pu^{240} is present. The relative worth of plutonium atoms to U^{235} atoms from a reactivity standpoint with no Pu^{240} present is 1.90 ± 0.14. This ratio changes rapidly with the percent of Pu^{240} that is present in the Pu. (P.C.H.)

273.

42936 (TID-7050(p.1148-1218)) HIGHLY ENRICHED CLEAN CRITICAL EXPERIMENTS AT AMBIENT TEMPERATURES. D. R. Connors (Westinghouse Electric Corp. Bettis Atomic Power Lab., Pittsburgh).

Experiment and theory are compared for critical assemblies of a simplified geometry, which is clean of an axial inhomogeneity due, for example, to the partial withdrawal of the safety rods. Results are given for clean critical slab experiments for small-bundle box geometry and repeating plane geometry for metal-water systems, and for plastic-moderated critical experiments. (R.E.U.)

274.

30963 AN ANALYSIS OF 23 ZPR-III FAST-REACTOR CRITICAL EXPERIMENTS. William G. Davey (Argonne National Lab., Idaho Falls, Idaho). Nucl. Sci. Eng., 19: 259-73(July 1964).

Over fifty fast critical assemblies were studied in the Zero Power Reactor-III (ZPR-III). All of these assemblies were fueled with ^{235}U and reflected with depleted uranium; the core volumes ranged from 2 to 660 liters and the critical masses ranged from 25 to 580 kg of ^{235}U. The experimental characteristics of a representative group of 23 of these assemblies in which oxide, carbide, and metallic fuels were simulated were compared with calculated values. The parameters studied were critical size, central fission ratios, prompt-neutron lifetimes and the reactivity effects of substitution of various materials at the reactor center. (auth)

275.

37867 (A/CONF.28/P/381) THE CRITICAL PARAMETERS OF AQUEOUS SOLUTIONS OF $UO_2(NO_3)_2$ AND NUCLEAR SAFETY. B. G. Dubovskii, A. V. Kamaev, V. V. Orlov, G. M. Vladykov, V. N. Gurin, F. M. Kuznetsov, V. P. Kochergin, I. P. Markelov, G. A. Popov, and V. J. Sviridenko (U.S.S.R.). May 1964. 18p.

Experimental and theoretical investigations on critical parameters of uranium salt aqueous solutions with 90, 10, or 5 per cent enrichment are described. These investigations were carried out for determination of nuclear safety conditions and for studies of homogeneous reactor physics. Critical volumes of the solutions were determined at various uranium concentrations in the range of 30–450 g/l. Extrapolation lengths and effective additions of various reflectors were determined during experiments. The experimental results obtained for cylinder and rectangular cores by transformation of geometrical parameters using the extrapolation length and effective addition of light water reflector, were recalculated for cores having shapes of a sphere, infinite cylinder, and infinite flat layers. (auth)

276.

18613 CRITICAL PARAMETERS OF AQUEOUS URANYL NITRATE SOLUTIONS. V. G. Dubovskii, A. V. Kamaev, F. M. Kuznetsov, G. M. Vladykov, G. A. Popov, and Yu. D. Palamarchuk. At. Energ. (USSR), 16: 21-5(Jan. 1964). (In Russian)

The critical volumes of aqueous solutions of $UO_2(NO_3)_2$ solutions containing 90% enriched ^{235}U were determined on arrays containing spheres, cylinders, and rectangular parallelepipeds, assembled with and without water reflectors. The walls were made of 1 to 1.5 and 3-mm thick stainless steel sheets; the U concentration of the solution was varied from 36 to 460 g/l, corresponding to a variation of the ratio of the H to the U nuclei, ρ_H/ρ_{U-235}, from 780 to 50. The experiments without reflector were carried out in cells, the concrete walls of which were at least 2 m from the assembly. The thickness of the reflecting water layer was at least 20 cm. No corrections were made for the temperature of the solution which varied from 15 to 21°C. An excess of 13.6 g of HNO_3 was added for each 100 g of U per liter of solution. By transforming the geometric parameters, the dimensions of the system consisting of a sphere, cylinder, and an infinite flat slab were obtained. For the aqueous $UO_2(NO_3)_2$ solution the following critical parameters were obtained: critical volume, 8.4 liter; critical mass, 0.85 kg of ^{235}U; diameter of the infinite cylinder, 16.7 cm; and thickness of the infinite slab, 6.9 cm. (TTT)

277.

44968 (AAEC/E-123) BUCKLING AND INTEGRAL SPECTRUM MEASUREMENTS IN U235/BeO SUBCRITICAL ASSEMBLIES. P. Duerden, D. B. McCulloch, and E. Brittliff (Australia. Atomic Energy Commission Research Establishment, Lucas Heights, New South, Wales). July 1964. 34p.

The materials buckling of four BeO-moderated U-235/aluminum alloy fueled systems having BeO/U-235 ratios of 1465, 2930, 5860, and 8790 were measured by the exponential method. Relative fission rates of U-235, U-233, and Pu-239 were also measured in the equilibrium spectrum region of the same assemblies. The experiments are described in detail, and the results compared with the predictions of a simple spectrum model (Westcott) pending analysis using more complex calculational models. (auth)

Critical Experiments 278-285

278.

24865 (EURAEC-992) PLUTONIUM RECYCLING. Quarterly Report No. 13 (Suppl.). (France. Commissariat a l'Energie Atomique, Paris). 1963. Work Performed under United States–Euratom Joint Research and Development Program. Contract 037-60-12 RDF. 14p. (A translation).

A summary is presented of studies of: natural U and Pu–U fuels; graphite thermal columns; critical parameters; and operations in the French assemblies Aquilon, Marius, and Minerve. (T.F.H.)

279.

40213 (LA-1356(Del.)) PRECISION CRITICAL-MASS DETERMINATIONS FOR ORALLOY AND PLUTONIUM IN SPHERICAL TUBALLOY TAMPERS. G. E. Hansen and D. P. Wood (Los Alamos Scientific Lab., N. Mex.). Feb. 1, 1952. Contract [W-7405-eng-36]. 19p.

The results of precision delayed-critical-mass measurements on high density and high ^{235}U concentration oralloy cores close tamped in spherical tuballoy shells are given. The six critical-mass points obtained allow a reliable M_c vs Tu tamper thickness curve to be plotted. The results for Oy (93.9%), p = 18.75 gms/cm^3 are:

Tu thickness (in):	0	0.695	1.76	3.525	3.925	9.0
Oy critical mass (kg):	51.9	36.2	26.5	20.5	19.75	17.35

In addition, the M_c of Pu in a Tu tamper thickness of 4.603th in. was measured and found to be ~6.28 kg. (auth)

280.

26873 EXPONENTIAL AND CRITICAL EXPERIMENTS. VOL. I. Proceedings of the Symposium held in Amsterdam, Netherlands, 2-6 September 1963. (International Atomic Energy Agency, Vienna). Proceedings Series. Feb. 1964. 483p. (STI/PUB/79(Vol.I)); (CONF-133). $10.00; Sch 210,-; 60s.stg; F.fr. 40,-; DM 35,-(IAEA).

Separate abstracts were prepared for 10 of the 21 papers presented; 11 were previously included in NSA (17: 42220, 42221, 42222, 42224, 42225, 42236, 42238, 42382, 42385, 42386, and 42387). (R.E.U.)

281.

26881 EXPONENTIAL AND CRITICAL EXPERIMENTS. VOL. II. Proceedings of the Symposium held in Amsterdam, Netherlands, 2-6 September 1963. (International Atomic Energy Agency, Vienna). Proceedings Series. Feb. 1964. 619p. (STI/PUB/79(Vol.II)); CONF-133). $12.00; Sch 252,-; 72s.stg; F.fr 48,-; DM 42,-(IAEA).

282.

33026 EXPONENTIAL AND CRITICAL EXPERIMENTS. VOL. III. Proceedings of the Symposium held in Amsterdam, Netherlands, 2-6 September 1963. (International Atomic Energy Agency, Vienna). Proceedings Series. Apr. 1964. 485p. (STI/PUB/79(Vol.III)); (CONF-133). $10.00; Sch 210,-; 60s.stg; F.fr. 40,-; DM 35,-(IAEA).

Separate abstracts were prepared for 16 of the 25 papers included; 9 were previously abstracted for NSA (17: 42006, 42219, 42223, 42229, 42230, 42233, and 42235; 18: 1236 and 7898). (R.E.U.)

283.

38332 (A/CONF.28/P/878) EXPERIMENTAL STUDY OF CRITICAL SYSTEMS WITH Li^7H AND $ZrH_{1.6}$ MODERATORS. V. A. Kuznetsov, Y. A. Prokhorov, I. I. Zakharkin, I. E. Somov, and V. G. Maltsev (U.S.S.R.). May 1964. 18p.

Critical parameters were determined experimentally and corrections were considered for lithium-hydride and zirconium-hydride moderators on the PF-4 zero-power assembly. Neutron energy spectra were measured in the critical assembly with zirconium hydride moderator; from given flux distributions, it is concluded that neutron density can not be described in terms of a Maxwell distribution. (M.J.T.)

284.

36781 (HW-81659(p.26-32)) AN ANALYSIS OF URANIUM FUELED–LIGHT WATER MODERATED CRITICAL EXPERIMENTS. R. D. Liikala, J. R. Worden, and W. A. Reardon (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A detailed analysis of five uranyl nitrate critical solution experiments and six 2.70-wt % enriched-uranium-dioxide critical experiments has been made. The critical solution experiments were analyzed to determine the reliability of the uranium cross section data utilized, thereby eliminating one of the potential uncertainties in the analysis of the heterogeneous experiments. The description of the methods utilized in these analyses together with the results are described. (auth)

285.

3112 CRITICAL EXPERIMENTS AND CALCULATIONS WITH ANNULAR CYLINDERS OF U(93.2) METAL. J. T. Mihalczo (Oak Ridge National Lab., Tenn.). Trans. Am. Nucl. Soc., 6: 217-18(Nov. 1963).

286.

5967 (ORNL-3499(Vol.I)(p.49-50)) PROMPT-NEUTRON DECAY CONSTANT VS BUCKLING FOR HYDROGEN-MODERATED 3% U^{235}-ENRICHED URANIUM. J. T. Mihalczo (Oak Ridge National Lab., Tenn.).

Prompt-neutron decay constants were measured as a function of buckling for unreflected parallelepipeds of a homogeneous mixture of 8 wt % paraffin and 92 wt % UF_4 in which the U^{235} enrichment was 3.00%. The decay constants were determined by the pulsed-neutron technique, and the buckling values were computed using the measured extrapolation distance for one size and the variation of the extrapolation distance that was obtained from S_n transport theory calculations. (auth)

287.

34494 (CEA-R-2495) EXPERIENCES DE CRITICITE SUR DES CYLINDRES ANNULAIRES CONTENANT DES SOLUTIONS DE PLUTONIUM. (Criticality Experiments with Annular Cylinders Containing Plutonium Solutions). Michel Molbert, André Sauve, Michel Houelle, and Edouard Deilgat (France. Commissariat à l'Énergie Atomique. Centre d'Études Nucléaires, Saclay). June 1964. 59p.

Preliminary results are given from criticality analyses of annular cylinders containing plutonium solutions. Results are plotted for cylinders with light water reflectors, cadmium screens, and plutonium concentrations of 42.3 to 104.1 g/l. (R.E.U.)

288.

1284 (LA-2284) NUCLEAR PROPULSION TEST REACTOR KIWI-A NEUTRONIC STUDIES. J. D. Orndoff and J. C. Hoogterp (Los Alamos Scientific Lab., N. Mex.). Sept. 1958. Contract W-7405-eng-36. 53p.

Critical conditions were obtained for a series of uranium–graphite fueled assemblies in the C/Oy range 150 to 500 with thick graphite reflectors and Be or D_2O islands. Perturbations were studied in order to find the effect on reactivity of geometrical changes and engineering features. Information from these studies was used to develop a mockup of an optimum reactor geometry for the first rocket reactor test device, Kiwi-A. Studies on a zero power mockup (Zepo-A) helped establish its design details. A cold critical check was made of the Kiwi-A reactor. (auth)

289.

40212 (LA-1209) MEASUREMENTS ON UNTAMPED ORALLOY ASSEMBLY. J. D. Orndoff and H. C. Paxton (Los Alamos Scientific Lab., N. Mex.). Feb. 8, 1951. Contract [W-7405-eng-36]. 33p.

The characteristics of an untamped pseudospherical Oy assembly (about 98% critical) were surveyed. Rossi α values were determined at two different neutron multiplications. relative fission rates in Oy and Tu were measured as a function of radius, and reactivity contributions per gm-atom of Oy, Pu, Tu, Pb, Ag, Cu and ^{10}B were obtained at various positions. Results are considerably less reliable than corresponding data for assemblies operable at delayed critical. (auth)

290.

24349 (LAMS-3067) LOS ALAMOS CRITICAL-MASS DATA. H. C. Paxton (Los Alamos Scientific Lab., N. Mex.). Apr. 1964. Contract W-7405-eng-36. 61p.

Tabulated are critical masses of simple systems, which have been measured at Los Alamos through the year 1963. (auth)

291.

30958 (TID-7028) CRITICAL DIMENSIONS OF SYSTEMS CONTAINING U^{235}, Pu^{239} AND U^{233}. H. C. Paxton, J. T. Thomas, Dixon Callihan, and E. B. Johnson (Los Alamos Scientific Lab., N. Mex. and Oak Ridge National Lab., Tenn.). June 1964. Contract W-7405-eng-36. 150p.

A compilation of critical data obtained from experiments performed since 1945 is presented. The report supplements the Nuclear Safety Guide [TID-7016(Rev.1)] and shows the bases of the recommendations that appear in the Guide. Conversions to standard geometries is shown. Results are given for single units and multiple-unit arrays. 130 references. [Table 4, p.34—the last set of data applies to an unreflected 16.2-in. diameter cylinder.] (R.E.U.)

292.

26248 STUDY OF CRITICAL ASSEMBLIES WITH BERYLLIUM MODERATOR. N. N. Ponomarev-Stepnoĭ and S. S. Lomakin. At. Energ. (USSR), 16: 228-33(Mar. 1964). (In Russian)

Be-containing systems must be further investigated because the currently available data do not agree with the theoretical calculations. The relationship between the neutrons originating from the Be(n,2n) reaction and from fission was examined by means of U–Be critical assemblies, built from 100 × 50 × 10 mm thick Be plates with a density of 1.8 g/cm^3 and 100 × 100 × 0.5 mm fuel plates, consisting of U_3O_8 with a ^{235}U content of 1.34 g per element, and fluorothene. The slowing-down length was determined up to energy levels at which the transition to the thermal neutron spectrum and the fast neutron multiplication factor must be taken into account because of the fission and the Be(n,2n) reaction. The multigroup system of the neutron-physical constants used in the calculations yielded satisfactory results. The experimentally obtained value of τ = 90.1 cm^2, including the thermalization range from 1.44 to 0.2 ev, allowed the slowing-down process of neutrons in the Be to be defined more exactly. A value of γ = 1.12 was obtained, assuming the threshold value of the Be(n,2n) reaction is 1.85 Mev. The effect of this reaction must be taken into consideration when calculating for a Be-containing system. (TTT)

293.

25324 (K-1609) URANIUM DENSITY OF SYSTEMS INVOLVING MUTUALLY INSOLUBLE MATERIALS. Karl E. Rapp and R. L. Smitherman (Oak Ridge Gaseous Diffusion Plant, Tenn.). June 9, 1964. Contract W-7405-eng-26. 18p.

Critical masses, as they apply to the establishment of safe geometries, are altered by the involvement of hydrogen as a moderator. Theoretically, uranium compounds and hydrogen compounds which exhibit mutual insolubility should make proportional contributions to the composite uranium density of any system wherein they are combined. The uranium densities of four such systems were determined experimentally and the results obtained support the validity of the theory. Certain fundamental considerations were examined, which are potential sources of error in the application of these physical principles. (auth)

294.

28990 HOMOGENEOUS CRITICAL EXPERIMENTS WITH PLUTONIUM/PLASTIC MIXTURES IN THE INTERMEDIATE ENERGY RANGE. C. R. Richey, J. D. White, R. C. Lloyd, and E. D. Clayton (General Electric Co., Richland, Wash.). Trans. Am. Nucl. Soc., 7: 81(June 1964).

295.

11398 (HW-79054(p.54-63)) CRITICAL EXPERIMENTS WITH PuO_2–POLYSTYRENE COMPACTS. C. R. Richey, J. D. White, E. D. Clayton, S. R. Bierman, K. L. Garlid, and D. R. Skeen (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

The critical dimensions and Pu masses for arrays of PuO_2–polystyrene cubes with a Pu concentration of 1.14 g/cm^3 and a H/Pu atomic ratio of 15 were determined using an approach to criticality technique with a split table. Lucite reflectors were used in most measurements. Temperatures in the core regions were also measured, and neutron lifetimes in the arrays were obtained by noise analysis and pulsed neutron experiments. Additionally, temperature coefficients were measured, and flux traverses were carried out. (D.C.W.)

296.

22850 (HW-80020(p.31-7)) CRITICAL EXPERIMENTS WITH PuO_2–POLYSTYRENE COMPACTS. C. R. Richey, J. D. White, R. C. Lloyd, and E. D. Clayton (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

The compacts used in the experiments have a Pu concentration of 1.12 g/cm^3 and a H/Pu atom ratio of 15. The density of the polystyrene is 0.931 g/cm^3. The effects on the criticality of arrays of the rubberized plastic coating on each compact and of the fuel-removal safety and control rods were determined. Comparative reflector savings for water, Lucite, and a 3% enriched uranyl nitrate solution were measured. The effect of core density change on the criticality of a Lucite reflected prism was also investigated. (D.C.W.)

297.

36783 (HW-81659(p.45-8)) CRITICAL EXPERIMENTS WITH PuO_2–POLYSTYRENE COMPACTS. C. R. Richey, J. D. White, R. C. Lloyd, and E. D. Clayton (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Critical experiments on PuO_2–polystyrene compacts with a H/Pu atomic ratio of 15 are described. The Pu is 2.2% ^{240}Pu. Effects of a lucite reflector are taken into account. (T.F.H.)

298.

9061 PARAMETRIC SURVEY OF CRITICAL SIZES. William H. Roach (Los Alamos Scientific Lab., N. Mex.). Progr. Nucl. Energy, Ser. IV, 5: 505-33(1963).

A survey of critical sizes of simple geometric shapes is presented from experiment and calculation. An extension to computational surveys is given from the DSN computing scheme with a standard set of multigroup input parameters. 47 references. (R.E.U.)

299.

807 (LAMS-2955) CRITICAL DIMENSIONS OF URANIUM(93.5)–GRAPHITE–WATER SPHERES, CYLINDERS, AND SLABS. William R. Stratton (Los Alamos Scientific Lab., N. Mex.). May 1962. Contract W-7405-eng-36. 36p.

A parametric study of the critical dimensions of U(93.5)–graphite–water spheres, slabs, and cylinders is presented. The data were obtained by application of the LASL DSN code and include five different loadings of uranium in graphite; the range of moderation as caused by water is complete. Both unreflected and water-reflected systems are considered. (auth)

300.

38651 INVESTIGATIONS ON THE ZR-1 CRITICAL ASSEMBLY. F. Szabo, Z. Gyimesi, T. Barta, L. Turi, and J. Vigassy (Central Research Inst. for Physics, Budapest). Rev. Phys., Acad. Rep. Populaire Roumaine, 6: Suppl., 403-14(1961).

A summary is presented of the results of measurements made on the ZR-1 critical assembly. The purpose of the measurements was to provide a basis for research on organic moderators. Critical masses for various H/^{235}U ratios were determined directly. Flux distributions were measured in various lattices, and the temperature dependence of reactivity in various lattices was investigated. (T.R.H.)

301.

33013 (A/CONF.28/P/265) FAST CRITICAL EXPERIMENTS AND THEIR ANALYSIS. F. W. Thalgott (Argonne National Lab., Idaho Falls, Idaho), J. K. Long, W. G. Davey, W. Y. Kato, S. G. Carpenter, H. A. Morewitz, and G. H. Best. 15p.

Prepared for the United Nations Third International Conference on the Peaceful Uses of Atomic Energy, 1964.

A review of Pu- and Th-cycle fast critical assemblies is presented. Operations of ZPR-III, ZPR-VI, the Epithermal Critical Experimental Laboratory facility, and LAMPRE-I are reported. (T.F.H.)

302.

1210 (KAPL-M-SW-3) CATALOGUE OF SHA EXPERIMENTS AND CALCULATIONS. S. Weinstein, R. Bobone, and F. Feiner (Knolls Atomic Power Lab., Schenectady, N. Y.). July 2, 1963. Contract W-31-109-eng-2. 49p.

The results of experiments performed to determine the critical configurations of a solid homogeneous assembly with an assortment of reflectors and internal neutron absorbers are presented. The configurations were represented in few group neutron diffusion models, and eigenvalue calculations were performed. The experiments and calculations are compared, and the effects of the nuclear data and computational model are discussed. (auth)

303.

28982 NEUTRON-SPECTRUM MEASUREMENTS IN A (UF_4 + PARAFFIN) MULTIPLYING ASSEMBLY. J. C. Young, J. M. Neill, and J. R. Beyster (General Atomics Div., General Dynamics Corp., San Diego, Calif.). Trans. Am. Nucl. Soc., 7: 71-2(June 1964).

1965

304.

25685 (HW-84369(p.25-33)) U–Al–POLYETHYLENE EXPERIMENT IN THE PCTR. D. G. Albertson (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

The first of a series of experiments which simulate homogeneous, light-water-moderated U and Pu systems was completed in the PCTR. The principal components for the experiment were U–Al (33 wt % U, enriched to 93.0 wt % ^{235}U), pure polyethylene, and borated polyethylene (0.8 wt % B). A determination was made of the B concentration necessary to reduce the value of the infinite multiplication factor (k_∞) to unity. The atomic ratio of H to ^{235}U was 214; the critical value of the atomic ratio of ^{10}B to ^{235}U was determined to be 0.119. (auth)

305.

45522 (AE-195) EXPERIMENTAL STUDIES ON ASSEMBLIES 1 AND 2 OF THE FAST REACTOR FRO. PART 1. Andersson, T. L.; Hellstrand, E.; Londen, S-O.; Tiren, L. I. (Aktiebolaget Atomenergi, Stockholm (Sweden)). Aug. 1965. 48p. Dep. mn.

FRO is a fast zero power reactor built for experiments in reactor physics. It is a split table machine containing vertical fuel elements. 120 kg of ^{235}U are available as fuel, which is fabricated into metallic plates of 20% enrichment. The control system comprises 5 spring-loaded safety elements and 3 + 1 elements for startup operations and power control. The reactor went critical in February 1964. The first assemblies studied were made up of undiluted fuel into a cylindrical and a spherical core, respectively, surrounded by a reflector made of copper. Some experiments made on these systems are described. Primarily, critical mass determinations, flux distribution measurements and studies of the conversion ratio are dealt with. The measured quantities were compared with theoretical predictions using various transport theory programs (DSN, TDC) and cross section sets. The experimental results show that the neutron spectrum in the copper reflector is softer than predicted, but apart from this discrepancy agreement with theory was generally obtained. (auth)

306.

35604 (ANL-7045(p.1-23)) LIQUID-METAL COOLED REACTORS. (Argonne National Lab., Ill.).

Operation and maintenance of EBR-II are described, and design and development of FARET control drives, instrumentation, and static vessels are discussed. Investigations of Na-air reactions are described. FARET reactivity using carbide-fueled cores was investigated. Doppler experiments with ZPR-3 Assembly 45 are reported. Assembly #6 was reloaded in the ZPR-9 facility. Data on critical properties of assembly 6 are included. Design work on ZPPR is summarized. Development of refractory alloys for use in O_2-contaminated Na is discussed along with progress in fast reactor fuel processing. (J.R.D.)

307.

47989 (ANL-7090, pp 1-29) LIQUID-METAL-COOLED REACTORS. (Argonne National Lab., Ill.).

Progress on the reactor development program is reported. Various aspects of the following reactor programs are discussed; EBR-II, FARET, ZPR-6, ZPR-9, and ZPPR. Doppler measurements and the fat-man effect are given for the ZPR-6. Critical masses of the ZPR-9 assembly according to reflector thickness are given. Further developments in fast reactor fuels are discussed. (M.O.W.)

308.

25246 (HW-84369(p.53-4)) UNREFLECTED CRITICAL MASS OF Pu^{238} METAL. L. L. Carter (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

A series of Monte Carlo calculations was carried out to determine the bare spherical critical masses of ^{238}Pu and ^{239}Pu in metallic form. A summary of the Monte Carlo results is included. (J.R.D.)

309.

13596 (RPI-328-20(p.1-8)) **MULTIPLYING MEDIUM NEUTRON SPECTRUM MEASUREMENTS. R. J. Cerbone, R. E. Slovacek, R. R. Fullwood, and E. R. Gaerttner (Rensselaer Polytechnic Inst., Troy, N. Y.).**

Reduction and analysis of data obtained in measurements on a cube of homogeneous ^{235}U fuel–zirconium oxide–paraffin mixtures are reported. The data were collected at the center and at the surface using the Poole technique to obtain the steady-state spectrum, and the pulsed linac-phased chopper technique was used and to measure the asymptotic spectrum. (J.R.D.)

310.

18735 (HW-79572) STUDY OF THE CHAIN REACTION IN PLUTONIUM. E. D. Clayton (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Nov. 8, 1963. Contract AT(45-1)-1350. 39p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

Several hundred critical mass determinations were made of plutonium solutions in a Critical Mass Laboratory. This facility is especially designed and equipped to handle plutonium. A brief discussion is given of the experimental facility, its special features, and of the critical assemblies used in these experiments. Factors affecting criticality are presented and critical data given for plutonium; a few simple comparisons are made between plutonium and uranium. The results of Monte Carlo calculations of k, η, f for U and Pu in water systems are presented. Experimental methods of criticality measurement are discussed. Nuclear safety considerations and the need for critical mass data are briefly reviewed. New critical data on an intermediate spectrum critical assembly consisting of a homogeneous mixture of plutonium dioxide in plastic are presented. (auth)

311.

16437 (HW-83603) PROGRESS IN PLUTONIUM CRITICAL MASS PHYSICS. A Summary of Hanford Laboratories Achievements in These Programs Under General Electric, 1961–1964. E. D. Clayton (Hanford Atomic Products Operation, Richland, Wash.). Oct. 1964. Contract AT(45-1)-1350. 42p. Dep.; $2.00(cy), 1(mn) CFSTI.

Technical accomplishments and relevant activities performed in the Hanford Plutonium Critical Mass Laboratory since 1961 are reviewed. Several hundred critical-mass determinations have been made with plutonium in a Critical Mass Laboratory. This facility was specially designed and equipped to handle plutonium. Nearly three-fourths of the experiments were performed with plutonium nitrate solutions; the others were performed with PuO_2-plastic mixtures (solids) using a remote split-table machine. There have been no unusual incidents involving any of the critical assemblies during the conduct of these experiments. (auth)

312.

47967 PLUTONIUM CRITICALITY EXPERIMENTS. Clayton, E. D. (Battelle-Northwest, Hanford, Wash.). Phys. Today, 18: 46-8; 50-2(Sept. 1965).

For both reactor design and safety in nuclear-fuel processing, the masses, concentrations, and dimensions of critical plutonium systems must be known. Behavior is similar to that of uranium, but different enough that extrapolation is difficult and experiments are necessary. These experiments have examined criticality in water solutions and plutonium assemblies. The reasons for plutonium study, critical mass, volume, and size, criticality in ^{239}Pu and ^{235}U, plutonium isotopes, critical experiments, and safety are discussed. (J.F.P.)

313.

42077 (ANL–7014) AN ANALYSIS OF SIX AETR FAST CRITICAL EXPERIMENTS. Davey, William G. (Argonne National Lab., Idaho Falls, Idaho). Mar. 1965. Contract W-31-109-eng-38. 23p. Dep. mn; CFSTI, $1.00 cy, $0.50 mn.

Atomics International has constructed a series of multi-region fast-thermal critical assemblies as part of the Advanced Epithermal Thorium Reactor (AETR) program to obtain data on the physics of ^{233}U-Th-fueled reactors. In each case, the central test region was of a composition appropriate to a full-scale core. The physics data obtained in six cores, where the test regions simulated the cores of fairly large fast reactors, were analyzed using a cross-section set previously used in an analysis of ZPR-III fast critical assemblies. These data are central fission ratios, relative reactivities of certain materials, and the relative importance of neutron sources of different energies. Trends are identified in the differences between calculated and measured data, and these are compared with similar trends found in the ZPR-III assemblies. (auth)

314.

1839 (NAA-SR-7892) THERMAL NEUTRON SPECTRUM EFFECTS ON CALCULATIONS FOR GRAPHITE LATTICES. F. L. Fillmore (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). Oct. 15, 1964. Contract AT(11-1)-Gen-8. 27p. Dep.; $2.00(cy), 1(mn) OTS.

The effect of the thermal neutron spectrum on the multiplication constant in calculations for graphite-moderated lattices is examined. The point in question is not the theory of calculating the spectrum but rather how to employ spectrum results once they are obtained. The procedure for utilizing experimental flux measurements is also examined, and formulas are developed consistent with those of a commonly employed flux-averaging procedure for lattices using theoretical flux results. The average neutron velocity for each region of the lattice cell appears explicitly in these formulas. Results of calculations are given for 6 lattice spacings for a typical uranium-graphite lattice. Flux measurements are avilable for 3 of these spacings. Two essentially different calculational methods are employed. One of these uses the same neutron spectrum in all regions of the lattice cell while the other uses a different spectrum in each region. The latter method gives a harder spectrum in the fuel than in the moderator. Comparison of reactivity differences between these methods shows that taking into account the spectral hardening in the fuel reduces the calculated reac-

tivity by nearly 2 percent for the close lattice spacings. This difference would be greater for blacker fuel elements such as those employed in typical sodium-graphite reactors. The need for an accurate treatment of the thermal neutron spectrum is demonstrated. In particular, it is important to take into account the additional spectral hardening in the fuel relative to that in the moderator. The free-gas graphite scattering kernel with an effective carbon mass that is dependent on the graphite temperature, and perhaps on the lattice spacing, is adequate for this spectral hardening. (auth)

315.

8601 (NAA-SR-Memo-7891) MULTIGROUP CALCULATION METHODS WITH APPLICATION TO THE SGR CRITICAL EXPERIMENT. F. L. Fillmore (Atomics International. Div. of North American Aviation, Inc., Canoga Park. Calif.). Sept. 1, 1964. Contract AT-11-1-GEN-8. 69p. Dep.(mn); $3.00(cy), 2(mn) OTS.

The adequacy of methods for treating certain features of multigroup calculations in graphite-moderated lattices was investigated. The features of most concern are: (1) thermal neutron spectrum effects arising from the separation of fuel and moderator; (2) resonance capture and fission; and (3) fast fission. Methods applicable to homogeneous or nearly homogeneous systems find extensive use in practice. These methods are included in the survey in order to assess their reliability and to obtain results that can be compared with results obtained from other more complex procedures more appropriate for lattices. The calculations are made for a set of critical assemblies so that experimental results are also available for comparison. An additional purpose of this report is to present a reasonably detailed description of the multigroup methods being considered and to indicate points where improvements appear to be warranted. Although certain improvements may be difficult to implement, a better understanding of the specific problem areas cannot help but contribute to progress. It appears that one of the most significant deficiencies in methods that employ a homogeneous model in treating non-thermal groups is a large error in the treatment of fast fission. The defect in reactivity is about 4 percent for the lattices considered. This can be partially corrected by means of a diffusion theory cell calculation, but transport theory is needed in order to obtain good results. Another deficiency is encountered in methods that homogenize the fuel and moderator in the calculation of the thermal neutron spectrum. It was found that some of the homogenizing methods examined introduced in some cases errors of more than 2 percent in calculated reactivities. This error is of opposite sign to that of fast fission. Shortcomings in the treatment of resonance capture and fission were not found to contribute large errors in the lattices studied. This is reasonable because all of these lattices are well thermalized. Reactivity values obtained with a flux-weighted, multigroup procedure are in good agreement with measurements. (auth)

316.

19354 (ANL-6866) EXPERIMENTAL INVESTIGATIONS OF SECONDARY CRITICAL CONFIGURATIONS (ZPR-III ASSEMBLIES 27 AND 28). W. Gemmell, J. K. Long, and W. P. Keeney (Argonne National Lab., Ill.). Nov. 1964. Contract W-31-109-eng-38. 36p. Dep.; $2.00(cy), 1(mn) CFSTI.

Two configurations of a fast reactor core following a postulated accidental meltdown were investigated. The configurations were chosen to represent nonuniform distributions of fuel in which calculations of flux and reactivity worth could be compared with experimental values. The first configuration represented a case in which the center of a core had melted, and the top and center portions had collapsed into a dense fuel mass in the bottom of the reactor. The dense region is then surrounded by an annulus with the normal core composition, and this in turn is surrounded by the normal blanket. The second configuration represented a core in which vaporizing sodium caused the expulsion of all material from the axis of the core, and the annulus had melted and collapsed into a dense ring of fuel. If these situations were encountered in an actual meltdown, a secondary critical or supercritical configuration of fuel could occur. The history of this secondary configuration would be governed by the spatial relationships of the fuel worth, power distribution, and degree of criticality. The relationship of the power and fuel-worth distributions would determine if further motion of the fuel would result in an autocatalytic configuration. The experiments in the Zero Power Reactor III (ZPR-III) provided fission-rate and reactivity-worth distributions, which may be used to evaluate calculational methods designed to describe the history of possible accident configurations. (auth)

317.

42037 (BNWL-95, pp 4-17) REACTOR ANALYSIS. COMPARISON OF MEASURED AND CALCULATED SPECTRA AND AVERAGE REACTION RATES. Harris, R. A.; Reardon, W. A. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

Measured thermal-neutron spectra for several homogeneous, light water and polyethylene moderated systems, were compared to spectra calculated by theoretical methods presently in use. Average thermal group cross sections (chosen as cross sections for neutrons between 0 and .683 ev) were obtained by integrating energy-dependent cross sections over the measured spectra. These were then compared with the corresponding parameters obtained using the theoretical spectra. (M.O.W.)

318.

31510 EXPERIMENTAL AND THEORETICAL STUDIES OF HEAVY-WATER HOMOGENEOUS TWO-REGION SYSTEM. Jitsuya Hirota, Satoru Katsuragi, Reiko Saito, and Takehiko Yasuno (Japan Atomic Energy Research Inst., Tokyo). J. Nucl. Sci. Technol. (Tokyo), 2: 132-40(Apr. 1965).

A series of critical experiments and theoretical analyses was made on two-region systems in which a heavy water solution of uranyl sulfate with ^{235}U enrichment of about 20% is surrounded by heavy water, poisoned with B or mixed with ThO_2. D-to-^{235}U atomic ratios in the core solutions ranged from 1000 to 4500, depending on the core diameter and the blanket concentration. The theoretical effective

multiplication factor is decreased by treating the leakage of fast neutrons from the core rigorously and is increased by using spatial-dependent effective cross sections. These treatments are justified by the agreement found between the calculated and measured values for the Cd ratio and the thermal flux distribution. The disagreement seen in the effective multiplication factor between the theoretical and experimental values may be attributed to uncertainty in the resonance integral of ^{235}U. The discrepancy in the effective multiplication factors does not exceed 1%, when the accepted value of $\nu I_f - I_a$ is reduced to a smaller value. (auth)

319.

38141 CRITICAL MASS MEASUREMENTS FOR VARIOUS FUEL CONFIGURATIONS IN THE LASL D_2O REFLECTED CAVITY REACTOR. Jarvis, George A.; Byers, Cleo C. (Los Alamos Scientific Lab., N. Mex.). New York, American Inst. of Aeronautics and Astronautics, 1965, Preprint AIAA Paper No. 65-555, 14p. $1.00. (CONF-650610-2). ORINS

From American Inst. of Aeronautics and Astronautics, Propulsion Joint Specialists Conference, Colorado Springs.

The gaseous core cavity reactor is receiving continuing study as a possible basis for advanced performance nuclear propulsion systems. For this reason the Los Alamos Scientific Laboratory is operating a small cavity reactor, on a low priority basis, to accumulate information on some of the more important parameters for this type of reactor system. The LASL cavity reactor consists of a cylindrical cavity 40 inches in diameter and 40 inches in length, surrounded by a 20 inch thick D_2O reflector. For the initial fuel loading, 93.1 ^{235}U metal foils lined the surface of the cavity. The critical mass for this configuration was 5.7 kilograms of enriched uranium. Heating of the reflector results in increased critical mass. The reflector D_2O was heated through a range of about 40 degrees centigrade above room temperature. Critical mass as a function of reflector temperature is given. Measurements to examine consequences of a gaseous fuel shift were made. The fuel penalty as a function of orifice diameter is also given. (J.F.P.)

320.

3014 CRITICAL DIMENSIONS OF AQUEOUS UO_2F_2 SOLUTIONS CONTAINING 4.9% URANIUM-235-ENRICHED URANIUM. E. B. Johnson and D. F. Cronin (Oak Ridge National Lab., Tenn.). Trans. Am. Nucl. Soc., 7: 301-3 (Nov. 1964).

321.

44112 (CONF-651103-4) APPLICATIONS OF NUCLEAR SAFETY CRITERIA TO PLANT OPERATIONS. Ketzlach, Norman (Atomics International, Canoga Park, Calif.). [1964]. Contract AT(11-1)-Gen-8. 26p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

For Presentation at IAEA Symposium on Criticality Control of Fissile Materials, Stockholm.

Criticality parameters, developed by the use of conservative calculational methods, are presented for uranium carbide fuels. The method of analysis is outlined, together with the application of the criteria developed to plant processing conditions. A method is also presented for applying uranium metal-water data to other uranium fuels when no other criticality data are available. (D.C.W.)

322.

29773 CRITICAL EXPERIMENTS IN SUPPORT OF THE FIRST NERVA REACTOR. W. P. Kovacik, C. E. Barksdale, W. D. Rankin, and W. E. Freidhof (Westinghouse Astronuclear Lab., Pittsburgh). Trans. Am. Nucl. Soc., 8: 266-7(May 1965).

323.

19363 HOMOGENEOUS CRITICAL ASSEMBLY WITH PROFILED (SHAPED) FUEL LOADING. A. K. Krasin and E. I. Inyutin. At. Energ. (USSR), 18: 175-7(Feb. 1965). (In Russian)

A critical assembly scheme for a six-zone zero-power cylindrical reactor fueled with $UO_2(NO_3)_2$ acidified with nitric acid is described. Various concentration fuel delivered through polyethylene tubes maintained the same solution level in each zone. The radial neutron flux distribution in the active zone center was measured using indium tracers. The reactor reached criticality at the active zone level 39.6 ± 0.1 cm which corresponds to the critical load of 3250 ± 30 g ^{235}U. The diagram of radial neutron flux measurements shows the presence of bursts in the thermal neutron distributions which is characteristic of the multizone systems. The coefficient of non-uniform fission distribution along the radius is equal to 1.19. The profiled fuel loading law leading to a constant mean energy release along the reactor radius was estimated on the basis of the obtained data and on the assumption of weak thermal neutron field variations. The diagram of thermal neutron flux distribution and energy release for the given conditions is included. (R.V.J.)

324.

29669 PATHFINDER CRITICALITY PREDICTIONS AND MEASUREMENTS. Patrick Lacy (Allis-Chalmers Mfg. Co., Bethesda, Md.), John Fisher, Richard Kern, and Charles E. Larson. Trans. Am. Nucl. Soc., 8: 108-9(May 1965).

325.

25686 (NAA-SR-Memo-9977) CRITICAL EXPERIMENTS WITH DRIVEN UC FUEL IN A GRAPHITE MODERATED ASSEMBLY. D. W. Latham (Atomics International, Canoga Park, Calif.). Sept. 28, 1964. Contract [AT(11-1)-Gen-8]. 86p. Dep.(mn); $3.00(cy), 2(mn) CFSTI.

Critical experiments were performed with graphite moderated two-region cores consisting of a central test region fueled with 24 uranium carbide elements and a driver region fueled with uranium metal elements. The uranium carbide elements consisted of eighteen fuel pins within a stainless steel process tube with sodium filling the interstices to mock-up the coolant. Three triangular lattice pitches were studied. Within each lattice pitch, variations in proposed control channel materials and locations were studied. Results are given for two-region critical masses, values of k_{eff} for subcritical loadings of the 24 uranium carbide fuel elements, and first ring and peripheral driver element worths. Methods and results are presented in detail in an effort to implement the analytical techniques used in design calculations for advanced sodium graphite reactor cores. (auth)

326.

38255 THE EFFECTS OF HOT HYDROGEN PROPELLANT ON THE CRITICAL MASS OF GASEOUS NUCLEAR ROCKET CAVITY REACTORS. Latham, Thomas S. (United Aircraft Research Labs., East Hartford, Conn.); Herwig, Lloyd O. New York, American Inst. of Aeronautics and Astronautics, 1965, Preprint AIAA Paper No. 65-564, 20p. $1.00. (CONF-650610-5). ORINS

From American Inst. of Aeronautics and Astronautics, Propulsion Joint Specialists Conference, Colorado Springs.

Hot high-pressure hydrogen propellant in large reflector-moderated, gaseous-fueled cavity reactors is shown to increase critical masses by a factor of two or more relative to the critical masses required when hot hydrogen is not present in the cavity. Thermal neutron-flux distributions within the cavity are shifted by energy up-scattering collisions between the hot hydrogen and the thermal neutrons flowing into the cavity from the reflector-moderator. For low-energy neutrons, the probability for these energy up-scattering collisions is enhanced by the thermal motion of hydrogen at high temperatures. Critical masses are affected largely in proportion to changes in the flux-weighted, fuel-region cross sections, since the neutron-flux distribution in the reflector-moderator is not strongly dependent upon cavity conditions. Calculations were made for hydrogen pressures between zero and 1500 atm, for hydrogen temperatures of 20,000 and 60,000 K, for Pu-239 and U-233 fuels, and for two different reflector-moderator configurations. It was found that critical-mass increases were less drastic for U-233 than for Pu-239 due to the presence of fission resonances for U-233 in the 1.0 to 10.0 ev range. (auth)

327.

25245 (HW-84369(p.45-51)) CRITICALITY OF PLUTONIUM NITRATE SOLUTIONS IN SPHERICAL GEOMETRY. R. C. Lloyd, D. R. Skeen, and S. R. Bierman (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

Critical experiments were continued with plutonium nitrate solutions (4.6% ^{240}Pu) in spherical geometry. The experimental vessel in current use is a 15.2-in.-dia stainless steel sphere (30.2 liter volume) having a wall thickness of 0.048 in. Experiments were conducted with the vessel reflected by water and concrete; experiments were also made with the vessel unreflected. The concrete reflector was separated from the vessel by a 3⅝-in. air gap. During the course of the experiments, the effect on criticality of the vessel wall was evaluated. Gold foil traverses were made of a number of critical solutions to determine their respective critical bucklings. Also, pulsed neutron source measurements were made on a number of the assemblies. Neutron lifetime values are reported for the full bare critical spheres. (auth)

328.

45537 PHYSICAL CHARACTERISTIC OF CRITICAL ASSEMBLY WITH BeO MODERATOR. Lomakin, S. S. At. Energ. (USSR), 19: 69-71(July 1965).

Studies intended for the determination of the neutron-physics characteristics of a critical assembly with BeO moderator are discussed. The geometry of the assembly, using flat ^{235}U fuel elements, is described. Results of measurements are given for the dependence of the length of the active zone with three layers of fuel elements on the geometry, the variation of the geometrical parameter χ^2 on the number of elements in a layer at constant fuel concentration, and the distribution of thermal neutron density in a layer with three vertical fuel elements. The coefficient of fuel self-screening was calculated by various methods, including the Monte-Carlo method and the P_3 approximation, and compared with measurements. (M.J.T.)

329.

35706 (LA-3219) CRITICAL ASSEMBLIES OF FISSIONABLE MATERIALS. Mills, Carroll B. (Los Alamos Scientific Lab., Univ. of California, N. Mex.). Oct. 1959. Contract W-7405-eng-36. 21p. Dep.; $2.00(cy), 1(mn) CFSTI.

A number of critical assemblies were studied for purposes of reactor safety and criticality evaluation. These were H_2O-moderated ^{233}U, ^{235}U, and ^{239}Pu critical experiments, and D_2O, Be, BeO, and C moderated, enriched ^{235}U critical experiments that provided simple parametrics and extremes in type. The atomic densities and dimensions directly useful for computational purposes are listed for fast-to-thermal-flux spectrum assemblies. (auth)

330.

31504 (LA-3221) REACTOR MINIMUM CRITICAL DIMENSIONS. Carroll B. Mills (Los Alamos Scientific Lab., Univ. of California, N. Mex.). Oct. 1959. Contract W-7405-eng-36. 52p. Dep.; $3.00(cy), 2(mn) CFSTI.

The parametric study of minimum critical reactor dimension as a function of moderator, fissionable isotope, and size has been made, based on a consistent variety of critical experiments. Minimum critical size and mass have been computed for a range of concentration of ^{233}U, ^{235}U, and ^{239}Pu for H_2O-moderated bare and reflected slab, cylinder, and sphere geometries, as well as corresponding results for ^{235}U and heavier atom moderators D_2O, Be, BeO, and C. Some results are presented of the same sort for D_2O, Be, and C reflector-moderated reactors. (auth)

331.

40155 (LA-3229, pp 362-76) CRITICALITY. Mills, C. B. (Los Alamos Scientific Lab., Univ. of California, N. Mex.).

The effects of simple and complex geometries on critical concentration are computed for ^{235}U. The effects of different geometries on criticality are then pointed out for the conceptually simplest situation, that of a single large rocket power plant. Since criticality is closely connected with rocket reactor design, an investigation of design ideas is presented. (M.O.W.)

332.

42051 (UCRL-12013) PULSED NEUTRON MEASUREMENTS ON SOLID-MODERATED, ENRICHED-URANIUM SUBCRITICAL ASSEMBLIES. Morton III, J. R.; Goldberg, E. (Lawrence Radiation Lab., Univ. of California, Livermore). July 1964. Contract W-7405-eng-48. 16p. Dep. mn; CFSTI, $1.00 cy, $0.50 mn.

Measurements were made of the prompt neutron decay constant, alpha, for several subcritical assemblies. These measurements included the determination of both the de-

layed critical alpha and the variation of alpha with buckling. These near-homogeneous assemblies were BeO- or graphite-moderated and were fueled with enriched uranium (93.2% ^{235}U). The following moderator-fuel molar ratios were studied: $BeO/^{235}U$ = 123, 247, 370, 493, 3830, and $C/^{235}U$ = 1145. A method is described by which the data can be used for a sensitive test of quality of the nuclear cross sections such as are found in multigroup diffusion calculations. (auth)

333.

17310 THE ADVANCED EPITHERMAL THORIUM REACTOR (AETR) CRITICAL EXPERIMENTS. L. A. Mountford and H. A. Morewitz (Atomics International, Canoga Park, Calif.). Nucl. Sci. Eng., 21: 421-8(Apr. 1965).

Multiregion critical cores, built on a split-table assembly machine, were used to investigate reactor compositions typical of epithermal and fast power reactors. The design and operating features of the facility, constructed to study the Th–^{233}U fuel cycle, are described. Criticality calculations were in good agreement with experimental values. (auth)

334.

18736 (K-1629) MINIMUM CRITICAL CYLINDER DIAMETERS OF HYDROGEN MODERATED U(4.9) SYSTEMS. C. E. Newlon (Oak Ridge Gaseous Diffusion Plant, Tenn.). Mar. 15, 1965. Contract W-7405-eng-26. 13p. Dep.; $1.00(cy), 1(mn) CFSTI.

From a review of recent experimental data for a homogeneous, hydrogen moderated and reflected $U(4.9)O_2F_2$ system, and correlation with 16-group transport theory calculation, it appears that the minimum critical, infinite length cylinder diameter is 12.1 inches for the hydrogen moderated and reflected UF_6 system. The maximum uranium density of the UF_6 is taken as 3.2 g. U/cc. Calculations indicate that the use of a 0.20 inch wall thickness stainless steel or nickel container may increase the assigned minimum critical cylinder diameter about 0.9 inch. (auth)

335.

16438 (ORNL-3714(Vol.I)(p.31-42)) CRITICAL EXPERIMENTS. (Oak Ridge National Lab., Tenn.).

Research summaries are presented on critical experiments using enriched U cylinders, critical dimensions of enriched aqueous UO_2F_2, geometrically complicated enriched-U critical assemblies, critical experiments with enriched U, prompt-neutron lifetime in critical U cylinders HFIR critical experiments, reactivity effects of perturbations in critical experiments, and neutron multiplication in fissile material. References to publications in which the research is described are included. (J.R.D.)

336.

29025 THE CRITICALITY OF PLUTONIUM-238. Eric H. Ottewitte and Vahe Keshishian (Atomics International, Canoga Park, Calif.). Trans. Am. Nucl. Soc., 8: 201(May 1965).

337.

17244 CRITICALITY. H. C. Paxton and G. R. Keepin (Los Alamos Scientific Lab., N. Mex.). p.244-84 of "The Technology of Nuclear Reactor Safety. Vol. 1." M.I.T. Press, Cambridge, 1964.

The relations of critical-assembly experience, both experimental and analytical, to reactor safety are studied. An attempt is made to systematize the information available. The use of the multiplication factor as an index of nuclear safety is surveyed. Subcritical reactivity measurements and critical and zero-power operation are studied. Safety considerations for fuels outside reactors are also studied. 100 references. (T.F.H.)

338.

37511 (HW-84608(p.5-29)) A STUDY OF THREE AQUEOUS HOMOGENEOUS PLUTONIUM NITRATE CRITICAL SPHERES. Reardon, W. A.; Thomsen, D. H. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

Two water-reflected 11.5-in.-dia spheres and a 15.2-in.-dia sphere are used in criticality studies on plutonium nitrate solutions. Comparisons are given of the several methods and models for calculating k_{eff}. The RBU code, with its Monte Carlo routines and its diffusion theory code, is compared with the HRG-TEMPEST-SPECTRUM-HFN scheme of cross section preparation and computation in a multigroup, many-region, one-dimensional diffusion theory calculation. (T.F.H.)

339.

42038 (BNWL-95, pp 27-35) CRITICAL MASS PHYSICS. CRITICAL EXPERIMENTS WITH PuO_2-POLYSTYRENE COMPACTS ^{240}Pu = 8.0 PERCENT, H/Pu = 15. Richey, C. R.; White, J. D. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

A series of critical experiments were completed with homogeneous PuO_2-polystyrene compacts (hydrogen-to-plutonium atomic ratio = 15) having 8.0% ^{240}Pu. These data are applicable to the intervening systems of wet powders, precipitates, slurries, polymers, and other highly concentrated plutonium mixtures encountered in chemical plants processing reactor fuels. (M.O.W.)

340.

43521 CRITICALITY OF HOMOGENEOUS PLUTONIUM OXIDE-PLASTIC COMPACTS AT H : Pu = 15. Richey, C. R.; White, J. C.; Clayton, E. D.; Lloyd, R. C. Nucl. Sci. Eng., 23: 150-8(Oct. 1965). (BNWL-SA-184).

Critical experiments were conducted with homogeneous mixtures of PuO_2-polystyrene (H : Pu atomic ratio of 15) containing both 2.2 and 8.0% ^{240}Pu. Criticality was determined for a series of Plexiglas reflected rectangular prisms ranging from near cubes, to long columns, and to thin slabs; bare arrays of near-cubic geometry were also studied. Critical thicknesses were 16.09 ± 0.41 and 5.99 ± 0.10 cm, respectively, for the bare and reflected infinite slabs of PuO_2-polystyrene containing 2.2% ^{240}Pu. Corresponding values for the 8.0% ^{240}Pu mixtures were 18.48 ± 0.41 and 7.38 ± 0.09 cm. The infinite slab thicknesses for an equivalent ^{239}Pu-water mixture (H : Pu = 15, ρ = 1.62 g Pu/cm^3) were 11.66 ± 0.30 and 4.38 ± 0.08 cm, respectively, for the bare and water-reflected slabs. Corresponding critical radii for infinitely long cylinders were 10.52 ± 0.16 and 6.54 ± 0.14 cm; radii for critical spheres were 13.81 ± 0.16 and 10.40 ± 0.17 cm. (auth)

341.

29524 CRITICAL DIMENSIONS OF WATER-REFLECTED SYSTEMS CONTAINING U-235–H_2O–Zr. G. P. Rutledge, F. A. Dobbe, and C. H. Price (Westinghouse Electric Co., West Mifflin, Pa.). Trans. Am. Nucl. Soc., 8: 211-12(May 1965).

342.

3531 CRITICAL MASSES AND BUCKLINGS OF PuO_2–UO_2–H_2O SYSTEMS. L. C. Schmid, R. C. Liikala, W. P. Stinson, and J. R. Worden (General Electric Co., Richland, Wash.). Trans. Am. Nucl. Soc., 7: 216(Nov. 1964).

343.

35855 CRITICAL ASSEMBLIES. Thalgott, F. W. (Argonne National Lab., Idaho Falls, Idaho); Baker, A. R.; Carpenter, S. G. p.225-43 of ANS-100.

Descriptions are given of ZPR-III, ZPR-VI, Zero Power Plutonium Reactor, Epithermal Critical Experimental Facility, ZEBRA (UK), VERA (UK), BFS (USSR), SNEAK (Germany), and Masurca (France) critical assemblies. Studies in ECEF of the Advanced Epithermal Thorium Reactor are described. The characteristics and operating parameters of the critical assemblies are tabulated. (T.F.H.)

344.

10487 (LA-3166-MS) COMPUTATIONAL SURVEY OF HOMOGENEOUS WATER-MODERATED SYSTEMS. Peter M. Wallis (Los Alamos Scientific Lab., N. Mex.). Jan. 1964. Contract W-7405-eng-36. 41p. Dep.; $2.00(OTS).

The results of a calculational survey cover the critical sizes of homogeneous water-moderated systems fueled with ^{239}Pu, ^{233}U, and ^{235}U (at various enrichments). Also included are ^{235}U enrichment values of water-moderated, low-enrichment uranium mixtures for which $k_\infty = 1$. (auth)

345.

3545 OPERATIONAL MEASUREMENTS IN A TWO REGION (FAST-THERMAL) CRITICAL ASSEMBLY. F. G. Warzek, T. R. Jaworowski, and P. Meyer (General Electric Co., Pleasanton, Calif.). Trans. Am. Nucl. Soc., 7: 252-3(Nov. 1964).

346.

38127 (HW-84608(p.77-9)) CRITICAL EXPERIMENTS WITH PuO_2–POLYSTYRENE COMPACTS. White, J. D.; Richey, C. R. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

Critical experiments are carried out with PuO_2–polystyrene compacts at a H/Pu atomic ratio of 15 using a split-table device. The core material is composed of alternate 2-in. fuel cubes and 2-in. Plexiglas cubes. The core itself has a H/Pu atomic ratio of 35. Criticality characteristics are tabulated for the bare core and for various reflector and interface configurations and materials. (T.F.H.)

347.

14798 THEORETICAL ANALYSIS OF LIGHT-WATER MODERATED REACTOR EXPERIMENTS. Kazuko Yamamoto and Stoichi Terasawa (Hitachi Seisakusho Central Lab., Japan). Hitachi Hyoron, 45: 911-15(June 1963). (In Japanese)

To evaluate reactor parameters, Deutsch's analytic method (Nucleonics, 15: 47(1957)) was expanded with regard to the fast neutron fission effect, effective resonance integral, Fermi age, economy of reflector, etc., and a survey code was prepared. A good agreement was shown between experimental data and the theoretical computation for the critical mass and lattice parameters. The temperature constant was found to be very small, and it was concluded that this analytic method was too simple to deal adequately with it. (BBB)

1966

348.

36421 (AE-233) A SUMMARY REPORT ON ASSEMBLY 3 of FRO. Andersson, T. L.; Brunfelter, B.; Cecchi, P. F.; Hellstrand, E.; Kockum, J.; Londen, S-O.; Tiren, L. I. (Aktiebolaget Atomenergi, Stockholm (Sweden)). June 1966. 35p. Dep. mn.

The third core of the zero energy fast reactor FR0 consisted of 20% enriched uranium diluted to 29 vol % with graphite and had a volume of 50 liters. Like previous cores it was surrounded by a thick copper reflector. Measurements of critical mass, control rod reactivities, fine structure flux variations, and conversion ratios are summarized. In particular, effects associated with the heterogeneous arrangement of the uranium and graphite plates are examined. (auth)

349.

6579 (BNWL-149, pp 48-56) CRITICAL MASS PHYSICS. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

Three separate papers are presented along with their results. Foil activation studies in two plutonium critical systems, Monte Carlo calculated values for the critical mass of ^{238}Pu as a function of hydrogen to fuel atomic ratios and ^{239}Pu enrichment, and resonance absorber admixed in the moderator of heterogeneous arrays are the presented papers. (M.O.W.)

350.

22383 (AEC-tr-6693) CALCULATIONS OF CRITICAL EXPERIMENTS ON ASSEMBLIES IN ZPR-3 WITH THE AID OF THE SYSTEM OF CONSTANTS FEI. Bazazyants, N. O.; Zaritskii, S. N.; Troyanov, M. F. Translated for Argonne National Lab., Ill., from Byull. Inform. Tsentra Yadern. Dannym, 2: 247-80(1965). 39p. Dep. mn. CFSTI $2.00 cy, $0.50 mn. JCL $3.60 fs, $1.37 mf.

Critical experiments were made in 11 assemblies in ZPR-3, simulating metallic, oxide, carbide, and ceramic cores of fast reactors of various volumes with uranium fuel. The following characteristics of spherical assemblies were measured and calculated; critical mass (effective

coefficient of multiplication k_{eff}), fission cross section indices $\sigma_f^{25}/\sigma_f^{49}$ and reactivity introduced by various samples, R_i, in the center of the assembly and lifetime of prompt neutrons l. (M.O.W.)

351.

47079 THE ZERO-ENERGY EXPERIMENT FOR THE MULTI-PURPOSE REACTOR (MZFR). Behrens, Ernst; Gruen, Arnold E.; Ritz, Hilmar (Siemens-Schuckert Research Center, Erlangen, Ger.). Siemens Rev., 33: 179-83(Mar. 1966).

The main objective of the measurements described, made during the zero-energy experiment, was the determination of the properties of the completely loaded but cold reactor without control rods. All measurements were performed in critical or subcritical states which allowed reasonably accurate extrapolation to obtain values for the state to be studied. Two ways were used to obtain critical states; the reactor core was reduced to a critical volume, and a strong neutron absorber, such as boric acid, was added to the moderator. The possibility of calibrating control rods by boron substitution has, however, a decisive limit, namely where the rod configuration under consideration has a reactivity larger than the total excess reactivity of the reactor. One parameter determined by the zero-energy experiment was the temperature coefficient which indicates how much the reactivity changes when the reactor temperature increases by 1°C. The value for the state under consideration (full water level, nonpoisoned) was calculated. The experimental and theoretical results for two critical states and the theoretical curves for the state under consideration showed close agreement, pointing to a sufficiently reliable theoretical temperature coefficient for the state under consideration. Of the various other measurements performed, one was the neutron density distribution in a radial and axial direction. Other work included measurements of the reactivity of the coolant, of the central fuel element, and of many built-in devices. (BBB)

352.

18057 MONTE CARLO ANALYSIS OF SOME ENRICHED ^{235}U FAST CRITICAL ASSEMBLIES. Benzi, V.; Cupini, E.; De Matteis, A. (CNEN, Bologna). J. Nucl. Energy: Pt. A and B, 20: 17-24(Jan. 1966).

A Monte Carlo strategy adopted for an analysis of some highly enriched ^{235}U fast critical assemblies is briefly described. Moreover a twenty-group cross section set for ^{235}U and ^{238}U giving a satisfactory agreement between the results of calculation and experiment is reported. (auth)

353.

3369 (CEA-R-2814) ALECTO—RESULTATS DES EXPERIENCES CRITIQUES HOMOGENES REALISEES SUR LE ^{239}Pu, ^{235}U ET ^{233}U. (ALECTO—Results Obtained with Homogeneous Critical Experiments on ^{239}Pu, ^{235}U and ^{233}U). Bruna, Jean-Georges; Brunet, Jean-Paul; Caizergues, Robert; Clouet D'Orval, Christian; Kremser, Jacques; Tellier, Henry; Verriere, Philippe (Commissariat a l'Energie Atomique, Saclay (France). Centre d'Etudes Nucleaires). 1965. 130p. Dep. mn.

The results of the homogeneous critical experiments ALECTO, made on plutonium-239, uranium-235, and -233 are given. After a brief description of the equipment, the critical masses for cylinders of diameters varying from 25 to 42 cms are given and compared with other values (foreign results, criticality guide). Experiments relating to cross sections and constants to be used on these materials are presented. Lastly, kinetic experiments allow a comparison of pulsed neutron and fluctuation methods. (auth)

354.

14077 STUDY OF HOMOGENEOUS CORE ASSEMBLIES USING PULSED NEUTRON SOURCES. Bruna, J. G.; Brunet, J. P.; Buchet, J.; Houelle, M.; Tellier, H. (Centre d'Etudes Nucleaires, Saclay, France). pp 157-78 of STI/PUB/104(Vol. 2). (In French).

The pulsed neutron technique was applied to the study of light-water-moderated homogeneous core assemblies. Using fissile materials in solution form it is easy to achieve large variations in geometrical buckling and moderation ratio. In the initial series of experiments, the fuel consisted of ^{235}U or ^{233}U in the form of uranyl nitrate. The solution concentrations used varied from 44 to 326 g of uranium per liter. Adoption of a coherent series of cross sections made it possible to deduce, from the variations in the prompt neutron decay constant as a function of geometrical buckling, data on the non-leakage probability and the slowing-down area. Effective cross sections calculated on the basis of assimilating light water to a secondary differential thermalizer were adopted. Interesting comparisons are made possible by the use of two fissile materials with markedly different η and resonance capture values. In a second series of experiments, devoted mainly to safety measures, the maximum permissible concentrations in various containers at processing plants were deduced by measuring the prompt neutron decay constants in weak plutonium nitrate solutions. (auth)

355.

36441 RESULTS OF HOMOGENEOUS CRITICAL EXPERIMENTS CARRIED OUT WITH ^{239}Pu, ^{235}U, AND ^{233}U. Bruna, J. G.; Brunet, J. P.; Caizergues, R.; Clouet d'Orval, C.; Verriere, P. (Commissariat a l'Energie Atomique, Paris). pp 235-48 of STI-PUB-114. (In French).

The properties of ^{239}Pu, ^{233}U, and ^{235}U were compared on the basis of critical experiments carried out on ALECTO assemblies with essentially the same geometry. On the basis of the results obtained, an attempt was made to determine empirically the slowing-down area as a function of concentration; the multiplication factor was calculated by using, for the cross sections, a secondary differential thermalization model developed for light water. Attempts were also made to determine prompt-neutron lifetimes using a pulsed-source method and to obtain information on absorption and fission cross sections in a hydrogenated medium. After carrying out various geometrical conversions which make it possible to go from critical dimensions in cylindrical geometry to critical dimensions in spherical geometry, the critical masses obtained in ALECTO were compared with other experimental results and with the French criticality guide curves. (D.C.W.)

356.

35689 INTERPRETATION OF EXPERIMENTAL RESULTS OBTAINED AT THE VALDUC CRITICALITY STATION. Caizergues, R.; Fiegel, A.; Ithurralde, M. F. (Commissariat a l'Energie Atomique, Paris). pp 249-66 of STI-PUB-114. (In French).

The results of some criticality studies on aqueous solutions of plutonium nitrate in various geometries are summarized. The effects of the variation of the acidity and the ^{240}Pu content were studied. The experimental results are compared with the results of calculations. (D.C.W.)

357.

1410 (AHSB(S)HANDBOOK-1(1st. Rev.)) HANDBOOK OF CRITICALITY DATA. VOLUME I. Chalmers, J. H.; Walker, G.; Pugh, J. (comps.). (Authority Health and Safety Branch, United Kingdom Atomic Energy Authority, Risley (England)). 1965. 142p. Dep.; BIS $8.40; HMSO £2. 2s.

Data are given for the following systems; uranium metal-water, uranium metal-nitric acid-water, uranium metal-graphite, uranium oxide-water homogeneous, uranium oxide, plutonium metal-water, plutonium metal-nitric acid-water, plutonium oxide-water, plutonium oxalate-water, plutonium nitrates, plutonium oxide-uranium oxide-water, and other miscellaneous enriched uranium dioxide-water systems. (M.O.W.)

358.

3368 (BNWL-SA-214) CRITICALITY RESEARCH ON PLUTONIUM. Clayton, E. D.; Richey, C. R.; Lloyd, R. C.; Bierman, S. R.; Carter, L. L. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Aug. 12, 1965. Contract AT(45-1)-1830. 26p. (CONF-651103-10). Dep. mn; CFSTI $2.00 cy, $0.50 mn.

From IAEA Symposium on Criticality Control of Fissile Materials, Stockholm.

Included are significant results obtained from criticality experiments with plutonium solutions and compounds and the analyses of the obtained data. Previously estimated curves for critical dimensions of homogeneous water moderated ^{239}Pu in the undermoderated range were revised. Computed curves are also given which show the effect of ^{240}Pu on criticality, together with experimental data points at 2.2 and 8% ^{240}Pu. Measured decay constants from experiments on critical and subcritical systems are compared with computed values for H/Pu atomic ratios of 15–1060. An analysis of critical data on Pu solutions was made showing the effect of nitrate on criticality for reflected spheres with nitrate and Pu concentrations extending to several hundred grams per liter. The results from a subcritical neutron interaction experiment performed with cans of dry PuO_2 powder in a low density reflected cubic array are also presented. Aspects of the current criticality research on plutonium and of planned experiments are discussed. (M.O.W.)

359.

35690 CRITICAL PARAMETERS OF AQUEOUS SOLUTIONS OF $UO_2(NO_3)_2$ AND THE INTERACTION OF SUBCRITICAL HOMOGENEOUS ASSEMBLIES. Dubrovskii, B. G.; Kamaev, A. V.; Vladykov, G. M.; Bogartyrev, V. K.; Gurin, V. N.; Kuznetsov, F. M.; Sviridenko, V. Ya. (Inst. of Power Engineering at the USSR State Atomic Energy Commission, Obninsk). pp 267-78 of STI-PUB-114. (In Russian).

The results of experiments and calculations on the critical parameters of aqueous solutions of the salt $UO_2(NO_3)_2$ at enrichments of 5%, 10%, 36%, and 90% are presented. Measurements were made over a wide range of uranium concentrations in the solution in cylindrical vessels, flat slabs, parallelepipeds, and spheres, with water reflector and without reflector. Extrapolation lengths and geometrical parameters χ^2 were determined. The relation between the critical parameters and the nitrogen content of the solution is discussed. The minimum critical dimensions of a flat infinite layer and an infinite cylinder were determined. Measurements were made with the addition of different reflectors — water, graphite, steel and beryllium. The experimental data are compared with calculations from the equations obtained using an empirical slowing-down kernel. Multigroup calculations were carried out for the critical parameters of aqueous solutions in P_1 and P_3 approximations of the spherical harmonics method. Good agreement was obtained between the experimental and theoretical results. The results of experiments for determining the interaction between two, three and five subcritical assemblies, in the form of cylinders and parallelepipeds, in air and water are described. Critical parameters are also given for a system of many homogeneous sub-critical assemblies with and without reflector. Critical parameters for interacting systems of complex form such as intersecting cylinders and flat slabs at different angles are obtained. The spatial distribution of neutron fluxes is studied for interacting sub-critical assemblies, with and without an intervening absorber, for different k_{eff} ratios of the interacting assemblies. (G.O.Y.)

360.

6587 (NASA-TN-D-3097) CRITICAL MASS STUDIES WITH NASA ZERO POWER REACTOR II. I. CLEAN HOMOGENEOUS CONFIGURATIONS. Fix, Thomas A.; Hueller, Robert A; Ford, C. Hubbard; Alger, Donald L. (National Aeronautics and Space Administration, Cleveland, Ohio. Lewis Research Center). Nov. 1965. 20p. CFSTI $1.00 cy, $0.50.

The NASA Zero Power Reactor II (ZPR-II) was used to determine experimentally the critical masses for more than a tenfold range of highly enriched (93.2 percent ^{235}U) aqueous uranyl fluoride fuel concentrations in clean cylindrical geometries. The ZPR-II reactor tank permits the assembly of cylindrical cores 76.2 centimeters (30 in.) in diameter and with lengths (heights) up to 90 centimeters either bare or radially reflected by about 15 centimeters (6 in.) of water. The specific range of fuel concentrations was from hydrogen to uranium-235 atom ratios of about 150 (167 g ^{235}U/liter of fuel solution) to 1650 (15 g ^{235}U/liter). In addition to the critical masses, data are presented on the temperature coefficient of reactivity and the incremental reactivity worth at criticality for a similar range of concentrations. Some of the physical properties for the fuel concentrations are presented. A brief description of the ZPR-II and the experimental procedures used are also included. (auth)

Critical Experiments 361-368

361.

34948 CRITICAL PARAMETERS OF URANIUM (1.95) METAL CYLINDRICAL ANNULI. Johnson, E. B. (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 9: 185-6(June 1966).

362.

2995 THE CRITICALITY OF ^{244}Cm. Keshishian, Vahe; Ottewitte, Eric H.; Dunford, Charles L. (Atomics International, Canoga Park, Calif.). Trans. Amer. Nucl. Soc., 8: 549-50(Nov. 1965).

363.

35165 THE SORA CRITICAL EXPERIMENTS. Kistner, Gustav; Mihalozo, John T. (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 9: 184-5(June 1966).

364.

18053 ON THE COMPARISON OF THE THEORETICAL AND EXPERIMENTAL PARAMETERS FOR HOMOGENEOUS URANIUM–WATER CRITICAL ASSEMBLY. Kochenov, A. S.; Lyashchenko, N. Ya. At. Energ. (USSR), 19: 463-4(Nov. 1965). (In Russian).

The P_1 approximation of the neutron transport equation and the single-velocity thermal-neutron diffusion equation were used for the calculations; the energy range was divided into 12 groups, including the thermal range for the slowing-down equation. The critical assembly used for the experimental determinations consisted of 70 × 35 × 250 mm holders, with 250 × 70 × 2.7 mm fuel sheets pressed from polyethylene and U_3O_8 with a ^{235}U content of 90%. Foils of Al, Cu, and stainless steel were used as covering. It was found that the assembly was quasi-homogeneous when the hydrogen and ^{235}U concentration ratio was $\rho H/\rho^{235}U \cong 50$. For a value of this ratio of about 50, the water gap between the foils amounted to about 5 mm, or equal to the mean free path of the thermal neutrons in water; therefore at >50, the effect of the heterogeneity must be taken into account, and the method may be used only in the first case. For the calculation it was assumed that the assembly was spherical, with a 50-mm thick water reflector. Comparison of the results revealed agreement between theory and experiment. Thus, this method of calculation may be used for determining the critical dimensions of homogeneous, epithermal reactors using a hydrogenous moderator. (TTT)

365.

28721 MINIMUM CRITICAL MASS FOR LIMITED URANIUM CONCENTRATIONS. Kochurov, B. P. At. Energ. (USSR), 20: 243-7(Mar. 1966). (In Russian).

The minimum critical mass for limited uranium concentrations was studied. It was found that the optimum system is a three-zone system. Results were obtained for both plane and spherical geometries. (D.C.W.)

366.

16721 (AWRE-NR-1/66) MEASUREMENT OF THE CRITICAL MASS OF 37 1/2 PERCENT ENRICHED URANIUM IN REFLECTORS OF WOOD, CONCRETE, POLYETHYLENE AND WATER. Lane, R. C.; Perkins, O. J. E. (Atomic Weapons Research Establishment, Aldermaston (England)). Feb. 1966. 20p. Dep. BIS $1.40. HMSO 7s.0d.

The experimental arrangements used in ATLAS, a vertical assembly machine, for measurement of the critical mass of enriched (37½%) uranium in reflectors of wood, concrete, polyethylene, and water are described. The data obtained are presented and discussed. (auth)

367.

35687 MEASUREMENTS OF THE CRITICAL PARAMETERS OF UNDER-MODERATED URANIUM–HYDROGEN MIXTURES AT INTERMEDIATE ENRICHMENTS. Lane, R. C. (Atomic Weapons Research Establishment, Aldermaston, Eng.). pp 177-91 of STI-PUB-114.

The critical size of mixtures of 30% enriched uranium oxide and wax with $H/^{235}U$ in the range 8 to 80 and 37½% enriched uranium metal, as used in Zebra, were estimated by extrapolation from high multiplication assemblies. The cores were constructed in rectangular geometry from small blocks to facilitate stacking changes. Reflectors included polyethylene, Perspex, concrete and wood. The effect of other reflectors was estimated by replacement of one face of an otherwise polyethylene-reflected cube by the material of interest. Evidence is given that critical masses derived by this method are correct to within 5%, even when the spectrum of neutrons returned to the core from the replacement face is markedly different from that returned from the remainder of the reflector. Density exponents were estimated from measurements with low-density blocks, and one carbon dilution exponent is reported. The critical size of half-reflected cores is of interest to the criticality assessor. The parameters obtained from a near cube and a long pipe suggest that their dimensions are the arithmetic mean of those of bare and completely reflected cores of the same shape. The neutron interaction between partially reflected cores can be greater than that for the uniformly reflected case. To check the magnitude of this effect, two identical cores were reflected on all but the adjoining faces by polyethylene, and their critical separation was determined. The measurement was then repeated with the cores fully reflected, or with the polyethylene filling only the space between the reflector, leaving free space between the cores. Selected assemblies were rebuilt with a layer of cadmium between the core and the reflector to estimate its effect on critical size. (auth)

368.

29471 CRITICALITY STUDIES WITH PLUTONIUM SOLUTIONS. Lloyd, R. C.; Richey, C. R.; Clayton, E. D.; Skeen, D. R. Nucl. Sci. Eng., 25: 165-73(1966). (BNWL-SA-232).

A series of criticality experiments were performed with plutonium (4.6% ^{240}Pu) nitrate solution in stainless steel spheres of 11.5-, 14-, and 15.2-in. diam. Reflectors of water, concrete, paraffin, and stainless steel were used; experiments were also performed on the 15.2-in. sphere unreflected. The spheres were made critical with plutonium concentrations varying from 24 to 435 g Pu/liter and molarity varying from 0.2 to 7.7. The minimum critical

volumes for $Pu(NO_3)_4$ in water containing 4.6% ^{240}Pu were determined to be about 22 and 11 liters, respectively, for bare and reflected spheres at a concentration of 175 g Pu/liter. The effect of a 0.030-in. cadmium shell or a 4-in. air gap between the reflector and the vessel reduced the reflector worth to that of a nominal reflector (1-in. of water or less) for the concentrations of plutonium measured. Comparisons were made between experimental and theoretical results using multigroup diffusion theory. (auth)

3395 CRITICALITY STUDIES WITH PLUTONIUM SOLUTIONS. Lloyd, R. C.; Richey, C. R.; Clayton, E. D.; Skeen, D. R. (Battelle-Northwest, Richland, Wash.). Trans. Amer. Nucl. Soc., 8: 444-5(Nov. 1965).

369.

36499 CRITICAL MASS OF SEFOR MOCKUP IN ZPR-III. Long, J. K. (Argonne National Lab., Idaho Falls, Idaho); McVean, R. L.; Reynolds, A. B.; Stewart, S. L.; Weitzberg, A.; Leridon, A. Nucl. Sci. Eng., 25: 442-4(1966).

The design of the SEFOR mockup in the ZPR facility is shown and described. The ZPR-3 assembly composition for the SEFOR mockup is also given. It was found that critical mass calculations using the current SEFOR 18-group cross section set and one-dimensional (cylindrical) diffusion theory techniques agree with experiment to within 1%. (J.F.P.)

370.

36418 (AAEC-E-146) BUCKLING AND INTEGRAL SPECTRUM MEASUREMENTS IN ^{235}U FUELLED SUB-CRITICAL ASSEMBLIES MODERATED BY BeO/FERTILE MATERIAL MIXTURES. McCulloch, D. B.; Duerden, P.; Brittliff, E. (Australian Atomic Energy Commission Research Establishment, Lucas Heights). Dec. 1965. 80p. Dep. mn.

Exponential experiments were carried out to give the materials buckling of a number of near-homogeneous ^{235}U/aluminum alloy fueled systems having fertile oxides intimately mixed with the BeO moderator. Relative fission rates of ^{235}U, ^{233}U, and ^{239}Pu were also measured in the equilibrium spectrum region of each assembly. Five assemblies having 5 wt % natural uranium oxide in BeO were investigated for a range of BeO/^{235}U atomic ratios from 1500:1 to 5700:1. A similar range covering four assemblies was examined for 5 wt % thorium oxide in BeO. A comparison of the experimental results with diffusion theory calculations is included. (auth)

371.

42671 (AWRE-R-5/66) URANIUM-FUELLED VERA REACTOR EXPERIMENTS. McTaggart, M. H.; Goodfellow, H.; Paterson, W. J.; Weale, J. W. (Atomic Weapons Research Establishment, Aldermaston (England)). July 1966. 89p. Dep. mn. BIS $2.40. HMSO 12s. 0d.

These experiments form part of a series designed to obtain information against which the nuclear data for ^{235}U and ^{238}U can be checked over the energy range 10 to 500 kev. Seven fast reactor assemblies are described. All the cores contained ^{235}U and graphite, and some also contained ^{238}U and H_2. Detailed measurements were made on four of the assemblies but for the other three assemblies the critical masses only were measured. (auth)

372.

16030 KRITICHESKIE PARAMETRY GOMOGENNIKH RAZMNOZHAYUSHCHIKH SISTEM. (Critical Parameters of Homogeneous Multiplying Systems). Marchuk, G. I.; Kocherin, V. I.; Nevinitsa, A. I.; Uanadze, O. P. Moscow, Atomizdat, 1965. 144p.

Information on the critical parameters of reactors with various spectra is presented, obtained from neutron-physics and computer calculations of homogeneous systems. The basic theoretical methods for reactor calculations are described, including methods for calculating the spatial-angular moments of the neutron distribution function, the neutron diffusion length, kinetic effects, reactor critical masses, and multigroup constants. Homogeneous reactors with graphite, beryllium, and boron moderators are discussed. Results of calculations are correlated with experimental data. Tables of the critical masses and other physical parameters of homogeneous multiplying systems are presented. (M.J.T.)

373.

26421 (ANL-7133) CALCULATIONS OF ZPR-3 FAST ASSEMBLIES USING A 26-GROUP ARGONNE CROSS-SECTION SET. Meneghetti, D.; White, J. R. (Argonne National Lab., Ill.). Jan. 1966. Contract W-31-109-eng-38. 16p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

Calculations of a series of ZPR-3 fast critical assemblies using an Argonne 26-group cross-section set are described. Critical masses, detector-response ratios, and prompt neutron lifetimes are compared with reported experimental values. Comparison of intragroup cross section averaging by Pl and by consistent Bl are presented. Importances of reflectors in determination of prompt neutron lifetimes are also calculated. (auth)

374.

3394 CALCULATIONAL CHECKS OF PLUTONIUM CROSS SECTIONS BY COMPARISON WITH CRITICAL-ASSEMBLY RESULTS. Meneley, D. A.; White, J. R. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 8: 444(Nov. 1965).

375.

3392 CRITICAL URANIUM-METAL ANNULI REFLECTED WITH POLYETHYLENE AND GRAPHITE. Mihalczo, John T. (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 8: 442-3(Nov. 1965).

376.

6503 (ORNL-3858(Vol.1), pp 13-19) CRITICAL EXPERIMENTS. (Oak Ridge National Lab., Tenn.).

Summaries of 14 various critical experiments are presented. Experiments were carried out with uranyl fluorides, uranium-metal, uranium-molybdenum, and uranyl nitrate solutions. In addition, criticality studies of various reactor lattices are reported. (M.O.W.)

377.

35018 (DP-MS-66-38) CRITICAL EXPERIMENTS WITH HIGHLY BURNED-UP PLUTONIUM IN D_2O. Olson, R. L.; Graves, W. E. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Lab.). June 1966. Contract AT(07-2)-1. 9p. (CONF-660606-4). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From American Nuclear Society Meeting, Denver.

The research program for producing ^{244}Cm in the Savannah River Production reactors is discussed. The feed material for the program is primarily ^{239}Pu. After the concentrated fuel was fabricated into fuel assemblies, it was used for critical experiments. A brief discussion of these experiments is presented. (M.O.W.)

34946 CRITICAL EXPERIMENTS WITH HIGHLY BURNED-UP PLUTONIUM IN HEAVY WATER. Olson, R. L.; Graves, W. E. (E. I. du Pont de Nemours, Aiken, S. C.). Trans. Amer. Nucl. Soc., 9: 181-2(June 1966).

378.

13005 (LA-3366) CRITICALITY CONTROL IN OPERATIONS WITH FISSILE MATERIAL. Paxton, H. C. (Los Alamos Scientific Lab., Univ. of California, N. Mex.). Dec. 1964. Contract W-7405-eng-36. 48p. Dep. mn. CFSTI $3.00 cy, $0.50 mn.

A discussion of criticality control is presented to encourage a working knowledge on the part of those who design and perform operations with fissile material. Responsibilities of the AEC and of operating organizations and an outline of nuclear-safety experience lead to reasonable principles of nuclear safety. Next, empirical criticality information is presented to help develop a feel for conditions to be avoided during operations. The final portion covers criticality-control methods that are consistent with the stated principles and available criticality data. (auth)

379.

30858 CRITICAL AND EXPONENTIAL EXPERIMENTS WITH PLUTONIUM. Plumlee, Karl E. Power Reactor Technol., 8: 211-14(Fall 1965).

A brief survey of experiments on plutonium-bearing assemblies is presented. (D.C.W.)

380.

35164 CRITICAL EXPERIMENTS FOR THE ARGONNE ADVANCED RESEARCH REACTOR. Plumlee, K. E.; Daughtry, J. W.; Johnson, T. W.; Robinson, W. R.; Stanford, G. S. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 9: 182-3(June 1966).

381.

34947 COMPARISONS OF THEORY AND MEASUREMENT FOR AARR CRITICAL EXPERIMENTS. Shaftman, D. H.; Savio, R. P. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 9: 183-4(June 1966).

382.

28769 (WCAP-3385-54, App.A) SINGLE-REGION EXPERIMENTS. Sternberg, H. I. (Westinghouse Electric Corp., Pittsburgh, Pa. Atomic Power Div.). 105p.

The data and results of the single region UO_2 and PuO_2-UO_2 critical experiments performed under the Saxton Plutonium Program are given. These results include experiments made to determine critical configurations, bucklings, power distributions, flux shapes, reactivity worths of various perturbations and temperature coefficients. (auth)

383.

6554 (ANL-7092) CATALOG OF NUCLEAR REACTOR CONCEPTS. PART I. HOMOGENEOUS AND QUASI-HOMOGENEOUS REACTORS. SECTION III. REACTORS FUELED WITH MOLTEN-SALT SOLUTIONS. Teeter, Charles E.; Lecky, James A.; Martens, John H. (Argonne National Lab., Ill.). Sept. 1965. Contract W-31-109-eng-38. 59p. Dep. mn; CFSTI $3.00 cy, $0.75 mn.

Data are tabulated for homogeneous and quasi-homogeneous molten-salt-fueled reactors. The development and state-of-the-art of this type of reactor is discussed. Data are given for 48 one-region and 7 two-region reactors. 76 references. (T.F.H.)

384.

3378 (ORNL-P-1680) EXPERIMENTAL AND CALCULATED SYSTEM CRITICALITY. Thomas, J. T. (Oak Ridge National Lab., Tenn.). [1965]. Contract W-7405-eng-26. 38p. (CONF-651103-11). Dep. mn; CFSTI $2.00 cy, $0.50 mn.

From IAEA Symposium on Criticality Control of Fissile Materials, Stockholm.

Criticality parameter measurements on various combinations of fissile, moderating, and scattering materials are reported. The systems studied have simple geometrical configurations. The performance of various computational methods for predicting the criticality of these systems is evaluated. (T.F.H.)

385.

12359 (ORNL-TM-1195) CRITICALITY OF A SINGLE UNIT OF AQUEOUS URANYL FLUORIDE SOLUTION ENRICHED TO 5 percent IN ^{235}U. Webster, J. Wallace; Johnson, E. B. (Oak Ridge National Lab., Tenn.). July 23, 1965. Contract W-7405-eng-26. 10p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

As a part of the continuing criticality program with uranium enriched to 5% in ^{235}U, a comparison between theory and experiment was made with regard to the criticality of a volume of aqueous $U(4.98)O_2F_2$ solution (H : ^{235}U = 496) contained in a thin-walled stainless steel cylinder. The solution had a height-to-diameter ratio of about 2.5, thus making the end leakage reasonably small compared to the radial leakage. Computations indicated that the end leakage from the finite system corresponded to only 1.5% in reactivity. One-dimensional calculations, with an approximate correction for end leakage, are therefore valid. The Carlson S_n (DTF) code with the Hansen-Roach 16-group

cross sections was used for the calculation. The computed multiplication factor of 1.002 for the critical system lends further confidence to the method and to the cross section set. (auth)

1967

386

S597 (ANL-7044) CRITICAL STUDIES FOR THE FRENCH FAST REACTOR "RAPSODIE" (ZPR-3 ASSEMBLY 44). Hess, A. L.; Keeney, W. P.; Caumette, P.; Boyer, J. P. (Argonne National Lab., Idaho Falls, Idaho). Mar. 1966. Contract W-31-109-38. 73p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Criticality studies were carried out on ZPR-3 with a mockup of the French fast reactor, RAPSODIE. Critical data pertinent to the fuel specification for RAPSODIE, the evaluation of the RAPSODIE design control systems, and the collection of other data useful in verifying calculational techniques were acquired. The RAPSODIE core simulated in the mockup was approximately a 40-liter cylinder. The experiments included a physics-core and an engineering-core study. For the physics-core study, a simple core-blanket arrangement was constructed, and spectral indices and local reactivity coefficients were measured. In the fuel for the physics-core mockup, the ratio of U to Pu was about 5:1. In the fuel for the engineering core, the U-to-Pu ratio was reduced to 3.2:1. Mockups of the RAPSODIE axial regions, steel-Na end gaps, and different axial blankets were constructed with the engineering core, along with mockups of the RAPSODIE control and safety rods. Experiments in the full engineering mockup included rod-worth studies, material-reactivity-coefficient measurements, and traverses of neutron detectors and fuel-reactivity samples. This was the first ZPR-3 critical assembly to contain a significant loading of Pu. Heat generation from alpha decay of the Pu affected the accuracy of reactivity measurements, and the neutron source from spontaneous fission required different techniques for measuring reactor periods. (auth)

2. LATTICES

1956

1.

8947 AECD-4229
Brookhaven National Lab., Upton, N. Y.
k_∞ IN URANIUM–WATER LATTICES. T. Auerbach. July 28, 1955. Decl. Feb. 7, 1956. 6p. Contract AT-30-2-gen-16. $1.80(ph OTS); $1.80(mf OTS).

Theoretical values of k^∞ for slightly enriched U – H_2O matrices are determined and compared with available experimental values. (D.E.B.)

2.

4912 AERE-N/R-134
Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.
THE MEASUREMENT OF THE LAPLACIAN IN A LATTICE OF URANIUM METAL RODS AND GRAPHITE. D. V. Booker, W. S. Eastwood, F. W. Fenning, T. M. Fry, and E. R. Wiblin. Oct. 1947. Decl. Nov. 8, 1955. 26p.

The critical size of a pile built up of a particular arrangement of fissile material and moderator can be related to the Laplacian of the arrangement. In these experiments the Laplacian has been measured for three arrangements in order to check the effect on critical size of a thin annulus of water around the U rods, and a 2 cm. wide annulus of air around the U rods. The arrangement studied consisted of U rods of 1.36-in. diameter spaced at 9-in. intervals in a block of graphite 10 ft., 6 in. by 10 ft., 6 in. by 9 ft high, the U rods being enclosed in concentric Al tubes so that water could be placed in the annular space between the tubes. The method of measurement was to determine thermal neutron densities throughout the structure when a point source of neutrons was placed at the base. Indium foils were used to detect the neutrons and the experimental density distributions were fitted to the expected theoretical form. The constants necessary to give a least square fit were used to reduce $\mathcal{K}^2$. The results obtained were: no water present, with air-gap $\mathcal{K}^2 = 0.83 \pm 0.03$ m^{-2}, no water present, without air-gap $\mathcal{K}^2 = 0.90 \pm 0.02$ m^{-2}, water present, without air-gap $\mathcal{K}^2 = 0.58 \pm 0.02$ m^{-2}. (auth)

3.

6372 BNL-1578
Brookhaven National Lab., Upton, N. Y.
ANALYSIS OF THE CLEAN BUCKLINGS OF 1.3 PER CENT ENRICHED URANIUM-WATER LATTICES. J. Chernick. Apr. 6, 1953. Decl. Dec. 13, 1955. 11p. $0.15(OTS).

4.

1922 AECD-3677
Hanford Atomic Products Operation, Richland, Wash.
EXPONENTIAL PILE MEASUREMENTS IN GRAPHITE–URANIUM LATTICES. E. D. Clayton. June 1, 1954. Decl. Sept. 29, 1955. 108p. Contract W-31-109-Eng-52.

The results from a series of exponential experiments involving 30 graphite–uranium lattices are presented. The diameter of the slug was varied from 0.925" to 1.66", and measurements were also taken with one hollow slug (1.66 in. OD, 0.81 in ID). The lattice spacing was varied from $6^3/_{16}$ in. to 15 in. Both wet and dry lattice measurements were taken; that is, the lattices were measured with water and without water in the cooling annulus to determine the effect of the cooling water on the buckling or reactivity. The buckling values are given for various slug sizes and lattice spacings. The lattice diffusion lengths, utilization in the moderator, utilization in uranium, and multiplication constant are also listed for those lattices in which these quantities were determined. Calculated conversion ratios are given for all except the hollow slug size. (auth)

5.

5974
CONDUCTING AN EXPONENTIAL EXPERIMENT WITH A NATURAL-U GRAPHITE LATTICE. H. Dopchie, F. Leonard, M. Neve de Mevergnies, and G. Tavernier (Centre d'Etudes pour les Applications de l'Energie Nucleaire, Brussels, Belgium). Nucleonics 14, No. 3, 57-60(1956) Mar.

Exponential pile experiments designed to check theoretical calculations an U-graphite lattice buckling are described. (D.E.B.)

6.
3391 BNL-1785
Brookhaven National Lab., Upton, N. Y.
REACTIVITY COEFFICIENT MEASUREMENT OF BUCKLING. Kenneth W. Downes and Herbert J. Kouts. Mar. 18, 1954. Decl. Nov. 9, 1955. 10p. Contract AT-30-2-Gen-16.

Exponential measurements on H_2O—slightly enriched U lattices are reported. A volume ratio of 3 : 1 and U enrichment of 1.15% is used. (D.E.B.)

7.
3392 BNL-2016
Brookhaven National Lab., Upton, N. Y.
BUCKLING OF A NATURAL URANIUM LIGHT WATER MODERATED LATTICE. K. Downes. Aug. 23, 1954. Decl. Nov. 9, 1955. 8p. Contract AT-30-2-Gen-16.

Pile parameters for a natural U—H_2O lattice having a 1.5 H_2O-to-U ratio were measured. The best M^2 valve for this pile was found to be 33 cm^2, giving a k_∞ of 0.989. (D.E.B.)

8.
8662 NAA-SR-1535
Atomics International Div., North American Aviation, Inc. Canoga Park, Calif.
BUCKLING OF GRAPHITE MODERATED LATTICES CONTAINING SEVEN ROD FUEL CLUSTERS. F. L. Fillmore. Aug. 1, 1956. 18p. Contract AT(04-3)-49. $0.20(OTS).

Buckling and intracell flux measurements have been made on three graphite moderated exponential assemblies each having seven rod enriched U fuel clusters. Theoretical values of the buckling are calculated and compared to the experimental values. In evaluating the resonance escape probability, two values are used for the lethargy spread of resonance neutrons. The larger value of 5.6, which appears in most of the literature, compares less favorably with the experimental values than does the smaller value 2.6. The theoretical buckling compares favorably with experimental values for the two larger lattice spacings and is about 10 per cent low for the smallest lattice spacing. (auth)

9.
3038 BNL-1627
Brookhaven National Lab., Upton, N. Y.
BUCKLING OF LIGHT WATER LATTICES (.600" DIAMETER RODS, 1.3% AND 1.15% 25). Herbert J. Kouts, Kenneth W. Downes, Glen A. Price, Rudolph Sher, and Valentine J. Walsh. Nov. 5, 1953. Decl. Oct. 27, 1955. 18p. Contract AT-30-2-Gen-16.

Three methods are described for measuring the buckling of light-water lattices. In method 1, a sub-critical, cylindrical reactor lattice is placed in the neutron field of the Brookhaven Reactor thermal column. The vertical relaxation length (L) of the neutron-excited thermal neutron flux is then measured as a function of the loaded radius (R), and best values of B^2 and λ (reflector savings) are found which fit the calculated functional dependence of L and R to the measured data. In the second procedure, that of the standard exponential experiment, B^2 and λ are determined from plots of the measured radial and axial variation of the thermal neutron flux, determined by foil exposure. Method 3 is essentially that of the critical assembly. U is loaded in the presence of a Po—Be source until a k_{eff} of 0.99 is attained. The source is then removed, and the lattice is partly unloaded by steps. Foil counting methods and experimental procedures for obtaining relaxation lengths and radial traverses are explained. Tabulated buckling measurements are given for 1.15 and 1.3% enrichment factor~ at 4:1, 3:1, 2:1, 1.5:1 volume ratios. Graphs for B^2 and λ vs. volume ratio and spontaneous fission fluxes vs. R_{eff}^{-2} are also included. (K.S.)

10.
3229 BNL-1812
Brookhaven National Lab., Upton, N. Y.
LIGHT WATER LATTICE STUDIES—PAPER PRESENTED AT THE REACTOR INFORMATION MEETING AT ANL, OCTOBER 7-9, 1953. H. Kouts. Nov. 5, 1953. Decl. Oct. 27, 1955. 8p. Contract AT-30-2-Gen-16.

A study was made of pile core parameters for light water moderated, slightly enriched uranium rod assemblies. This information is provided by measurements in a series of exponential assemblies which differ in uranium enrichment, moderator-to-fuel volume ratio, and rod diameter. The enrichment range explored varies from 1.3% to 1%, the rod diameters vary from 0.600 to 0.250 in., and the volume ratios lie in the range from 4:1 to 1:1 (and in some cases are even smaller). The quantities measured were f, ϵ, p, B^2, M^2, and reflector savings (since the assemblies are reflected). Similar measurements reported at the last Reactor Information Meeting were done with 0.750" diameter rods with 1% nominal enrichment. (auth)

11.
3398 BNL-2184
Brookhaven National Lab., Upton, N. Y.
BUCKLING OF LIGHT-WATER MODERATED LATTICES OF .387" DIAMETER, 1.027% ENRICHED URANIUM RODS. H[erbert J. C.] Kouts, G[len A.] Price, K[enneth W.] Downes, R[udolph] Sher, and V[alentine J.] Walsh. Feb. 7, 1955. Decl. Oct. 10, 1955. 7p. $0.15(OTS).

12.
4936
REACTOR PARAMETERS OF A LIGHT WATER—NORMAL URANIUM LATTICE. H. Kouts, G. Price, K. Downes, R. Sher, and V. Walsh (Brookhaven National Lab.,

Upton, N. Y.). J. Nuclear Energy 2, 141(1955) Dec.

Measurements were made of the buckling, fast neutron effect, and thermal utilization of a sub-critical lattice consisting of 1.1-in. normal U rods, clad in Al, and immersed in light water. (B.J.H.)

13.

4362 CP-1381

[Chicago. Univ.] Metallurgical Lab.

η TEMPERATURE EFFECT. E. W. Bragdon, D. Hughes, and John Marshall. Feb. 25, 1944. Decl. Dec. 13, 1955. 7p. (A-2022). $1.80(ph OTS); $1.80(mf OTS).

The temperature coefficient of η has been measured with the thermal column of the Argonne pile using U foils of different isotopic ratios. The temperature change was effected by filtering neutrons through Ag. The measured fractional change/°C is $-5.2 \times 10^{-5} \pm 1.1 \times 10^{-5}$. (auth)

1957

14.

2676 CP-2048

Clinton Labs:, Oak Ridge, Tenn.

EXPONENTIAL EXPERIMENTS WITH WATER-METAL ROD LATTICES. Preliminary Report [on] Problem Assignment 303-X25P. T. Arnette, S. Bernstein, G. Branch, F. Byerly, C. Clifford, B. Feld, Haydn Jones, R. McCord, H. W. Newson, W. Myer, F. V. Pruitt, R. B. Stewart, J. R. Rush, R. Scalletar, L. Slotin, and L. B. Watson. Oct. 3, 1944. Decl. Dec. 14, 1955. 27p. Contract [W-7405-eng-39]. $0.25(OTS).

Measurements on twelve water metal lattices ranging in water to metal volume ratio from 3.27 to 0.273 are reported here. The values for k lay between 0.6 and 1.00 With the materials used and no gap, a k greater than 1.00 does not seem to be obtainable. By introducing a gap and decreasing the rod size, it may be possible to build a lattice having a k greater than one. (auth)

15.

13172 NAA-SR-140

North American Aviation, Inc., Downey, Calif.

A MEASUREMENT OF THE NEUTRON TEMPERATURE IN A NATURAL URANIUM D_2O LATTICE. A. T. Biehl and E. R. Cohen. Sept. 25, 1951. Decl. Mar. 28, 1957. 23p. Contract AT-11-1-GEN-8. $0.25(OTS).

An experimental determination of the mean neutron temperature in the fuel rods of a natural U–D_2O lattice using rods 1-in. in diameter on two different cell spacings is reported. The temperature of the neutrons was found to be 35° ± 25°C and 33° ± 16°C above ambient room temperature for the two lattices. The experimental technique used was to compare the flux distributions in a unit cell in (1) a normal hot multiplying lattice and (2) a single U slug in a cold, Pb–Cd alloy, non-multiplying lattice. This allowed a comparison to be made between diffusion length and absorption cross section of the U in the two cases, and accordingly, a comparison between the mean temperatures of the neutrons in the two lattices. (auth)

16.

9983 BNL-418

Brookhaven National Lab., Upton, N. Y.

NUCLEAR ENGINEERING DEPARTMENT PROGRESS REPORT [FOR] NOVEMBER 15, 1955–FEBRUARY 15, 1956. 29p. $0.25(OTS).

Values of buckling and reflector savings are given for boron-poisoned lattices of 0.387-in. diameter, 1.15% enriched U rods. Fast effect values for the same diameter rods in H_2O are given. The distribution of La between the fused-salt eutectic $MgCl_2$–KCl–NaCl and a U–Bi–La alloy was measured. Further work with EMF concentration cells on the Na–Bi system has yielded activity coefficients for Na in the concentration range of 0 to 0.20 mole/fraction. These coefficients increase slightly with concentration and are of the order of 10^{-5}. On the basis of infrared spectra obtained on pure liquid samples of BrF_3, BrF_5, and UF_6, it appears that the Perkin-Elmer infrared analyzer can be used to provide continuous analysis of the circulating dissolver stream of the pilot plant for the Volatile Fluoride Process. The solubility of Ti in Bi is given by the equation $\log_{10}$ (ppm Ti) = 2250/T + 6.08. It is shown that the product of the solubilities of Cr and Fe is constant at a fixed temperature. (For preceding period see BNL-380.) (M.H.R.)

17.

13859 Y-B23-22

Oak Ridge National Lab., Y-12 Area, Tenn.

RING TAMPING. Dixon Callihan and Frank Cronin. Nov. 5, 1952. Decl. Mar. 21, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

In a potential nuclear chain reacting system with two or more volumes of fissionable material surrounded by, but not separated by, a hydrogeneous neutron reflecting medium reliance is placed on neutron absorption by the H separating the components. The fissionable material used was U_3O_8 enriched to 5% in U^{235} mixed with glycerol tristearate for moderation. The H content of glycerol tristearate is 96% that of H_2O by volume and 111% by weight. The change in reactivity incurred by removal of the reflector material from between the components was determined. (F.S.)

18.

1857 HW-40930

Hanford Atomic Products Operation, Richland, Wash.

EXPONENTIAL PILE MEASUREMENTS IN WATER MODERATED LATTICES WITH ENRICHED URANIUM RODS. E. D. Clayton. Appendix II: BUCKLING CALCULATIONS FOR ONE PER CENT ENRICHED URANIUM–WATER ROD LATTICES. H. Neumann. Jan. 16, 1956. Decl. Nov. 28, 1956. 45p. Contract W-31-109-Eng-52. $0.40(OTS).

The results of a series of buckling measurements involving the use of enriched U rods (1.007% by weight U^{235})

in light water are presented. The buckling was determined for two rod sizes. The water-to-U-volume ratios were found. The information obtained with the preceding rod sizes and H_2O/U-volume ratios was sufficient (when coupled with existing data for other rod sizes) to estimate the maximum obtainable buckling for 1.007% U in light water. A curve is given relating maximum buckling with H_2O/U-volume ratio and rod diameter from which estimates may be made regarding criticality hazards with the 1.007% enriched U. The effect of replacing a water reflector with one of uranyl nitrate solution of two different concentrations. Some information was also obtained on the extrapolation length to be used in unreflected cases. (auth)

19.

4013 HW-40345(Del.)

Hanford Atomic Products Operation, Richland, Wash.

PHYSICS RESEARCH QUARTERLY REPORT [FOR] JULY, AUGUST, SEPTEMBER 1955. Dec. 7, 1955. Decl. with deletions Jan. 21, 1957. 43p. Contract W-31-109-Eng-52. $7.80(ph OTS); $3.30(mf OTS).

A new method of calculating resonance escape probabilities, results for the effective mass of a proton bound in the water molecule, and a comparison of similar techniques for calculating lattice constants developed independently by Hanford physicists and by a school of Russian physicists (as reported at the Geneva Conference) are reported. An estimate of maximum errors in f and η for thermal systems is described. Experiments and analysis thereof on the effects of neutron streaming in air channels through a moderator are reported. Buckling calculations and experimental results are given for graphite lattices employing 1.17-inch diameter natural U slugs and U^{235}–Al alloy slugs. An experimental measurement of the critical mass of an annular cylindrical array is described. Some measurements on, and the status of, construction of the Lattice Testing Reactor are reported. Development of a BF_3 counter suitable for operation at elevated temperatures is described. A method of determining the screening parameter in the Thomas-Fermi model of the atom is given for use in calculating atomic displacements produced by radiations in crystals. Calculated results are reported for chain-reacting concentration limits of U–Pu–H_2O systems. (auth)

20.

1275 HW-43441

Hanford Atomic Products Operation, Richland, Wash.

PHYSICS RESEARCH QUARTERLY REPORT [FOR] JANUARY, FEBRUARY, MARCH 1956. May 31, 1956. 69p. Contract W-31-109-Eng-52. $0.45(OTS).

The measurement of the multiplication factor of a lattice in the Lattice Testing Reactor is described and results given. Measurements are reported of the "favorableness" factors for this reactor. An experiment is described to measure that enrichment of uranium which will just permit unit multiplication in a water-U system. Buckling measurements are reported for graphite-enriched U lattices and for natural U lattices employing cluster-type rods. An exponential experiment to measure temperature coefficients of reactivity is described. Calculational results are given on the variation of lattice buckling with enrichment, on the thermal neutron flux in a cell and on the effect of graphite density on buckling. In theoretical reactor physics, a variational principle for multiple scattering is formulated, a new method for estimating fast leakage from bare reactors is outlined, and the spherical harmonic components of the flux in a void are computed. Experimental results showing that ν for Pu^{239} does not vary in the thermal region are quoted. A nuclear safety problem involving the interaction of an array of vessels is described. Finally, two instruments, a new type of pulse height analyzer and an automatic scanner and sample changer are described. (For preceding period see HW-42182.) (auth)

21.

2116 HW-44525

Hanford Atomic Products Operation, Richland, Wash.

PHYSICS RESEARCH QUARTERLY REPORT [FOR] APRIL, MAY, JUNE 1956. July 25, 1956. 75p. Contract W-31-109-Eng-52. $0.45(OTS).

The Lattice Testing Reactor has been used to measure k_∞ for a number of lattices. The temperature coefficient of this reactor has been measured. End effects on the measurement of k_∞ have been investigated. A series of measurements of the bucklings of enriched (1% U^{235}) U-graphite lattices are reported. Calculations are reported on the equilibrium parameters of Pu in a steady state U–Pu cycle reactor, on the reactivity effects of various coolants in a U-graphite lattice, on multiplication factors of homogeneous enriched U-water systems, and on neutron interaction between separated fissile systems. Measurements are reported on a precise determination of the U^{235} fission cross section with respect to Au^{197}, on the slow neutron fission cross section of Pu^{240}, and on the variation of η (Pu^{233}) with neutron energy. An analysis of the Pu^{239} fission cross section is given. A series of measurements on the critical masses of enriched U slugs in water is reported and analyzed to determine safe masses for handling. The energy losses of fission fragments in the U crystal are calculated and results are given. (For preceding period see HW-43441.) (auth)

22.

5493 HW-47012

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

NUCLEAR PHYSICS RESEARCH QUARTERLY REPORT [FOR] JULY, AUGUST, SEPTEMBER 1956. Nov. 5, 1956. 56p. Contract W-31-109-Eng-52. $0.40(OTS).

The Thermal Test Reactor was put in operation and the initial calibrations are described. Measurements of bucklings of enriched uranium–graphite lattices are reported. Results are also given for light water–enriched uranium lattices. A high temperature exponential experiment is described. Calculated curves for the bucklings of some special graphite lattices are given. A method for com-

puting the resonance escape probability is described, and some results are given. A table of disadvantage factors calculated with the P_3 method for a variety of fuel element sizes is included. A comparison of the U^{235} fission cross section and the Au^{197} capture cross section is reported. An experimental value is given for the slow neutron fission cross section of Pu^{240}. Development of a new circulating line chronotron for measurements in the millimicrosecond region is described. (For preceding period see HW-44525.) (auth)

23.

2663 CF-55-5-189.

[Oak Ridge National Lab., Tenn.]

PRELIMINARY CRITICAL TESTS USING AQUARIUM FUEL ELEMENTS IN THE BSR GRID. K. M. Henry, E. B. Johnson, and F. C. Maienschein. May 31, 1955. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80 (mf OTS).

Preliminary critical tests for the Aquarium reactor were performed in the BSR grid, in attempts to determine minimum critical mass and excess reactivity. The critical mass of Aquarium elements was lower than estimated but higher than BSR elements. (T.R.H.)

24.

10972 NAA-SR-209

North American Aviation, Inc., Downey, Calif.

BUCKLING MEASUREMENTS OF THERMAL NEUTRONS IN NATURAL URANIUM – D_2O SQUARE LATTICES. S. W. Kash. Dec. 1, 1952. Decl. Feb. 27, 1957. 16p. Contract AT-11-1-GEN-8. $0.25(OTS).

The buckling of the thermal neutron distribution for 22 natural U–D_2O square lattices has been experimentally determined. A brief description of the apparatus used is given. The computational methods employed and the corrections applied to the data are presented in detail. (auth)

25.

10279 BNL-3145

Brookhaven National Lab., Upton, N. Y.

CRITICAL ASSEMBLIES OF LIFHT WATER MODERATED, SLIGHTLY ENRICHED URANIUM ROD LATTICES AT BROOKHAVEN. Hazards Report. Herbert Kouts. Feb. 28, 1956. 25p. $0.25(OTS).

Descriptions are given of series of critical experiments with lattices of slightly enriched U rods moderated by light water. The experiments were carried out to supplement and extend the current series of exponential experiments on similar assemblies. A revised list of safety rules is included. (R.V.J.)

26.

9474 HW-49700

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

COMMENTS ON THE MEASUREMENT OF B^2_{in} AND K_∞. David D. Lanning. Apr. 17, 1957. 3p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

It is concluded that for natural U-graphite lattices with $K_\infty \simeq 1.04$, if the Xe effect is ~3%, then K_{eff} must be greater than 1.03 and it would be better to know K_∞ for the design of such a reactor. However, for enriched U reactors or D_2O lattices where $K_\infty \simeq 1.1$, and the Xe effect is ~3%, it would be better to know B^2_m for the design of the reactor. (T.R.H.)

27.

7843 NAA-SR-259

North American Aviation, Inc., Downey, Calif.

REACTOR PHYSICS QUARTERLY PROGRESS REPORT [FOR] FEBRUARY, MARCH, AND APRIL 1953. R. A. Laubenstein, ed. July 8, 1953. Decl. Feb. 11, 1957. 31p. Contract AT-11-1-GEN-8. $1.80(ph OTS); $2.70(mf OTS).

Exponential Experiment. A series of measurements of the thermal neutron bucklings of depleted U–D_2O lattices has been completed with the investigation of three additional lattices. Three lattices of slightly enriched U (0.9% U^{235}) in D_2O have also been studied. Besides the buckling measurements, intra-cell neutron density distributions and effective neutron temperature variations have also been measured. A nomograph for making resolving-time corrections has simplified the reduction of raw counting data to saturated foil activities. Data obtained from foils exposed to a neutron current were found to exhibit a linear correlation between the magnitude of the current and the ratio of activities counted on the two sides of the foil. A measurement of the thermal neutron diffusion length in D_2O can be used to determine the macroscopic absorption cross section of the water, and hence the light-water impurity. A discussion of the accuracy which can be obtained with this method of impurity determination is given. Water Boiler Neutron Source. Use of the water boiler neutron source for the exponential experiments and for miscellaneous irradiations has continued to increase. Preparations have been made for doing accurate danger coefficient measurements with the water boiler. These include the installation of an accurate indicator of control rod position and a thermocouple to indicate reactor temperature. Standard samples of B have been prepared. General Reactor Program. An investigation has been made into the possibility of securing intrinsic safety for heterogeneous power reactors. A possible method is to include in the reactor a relatively small homogeneous region containing a moderator, such as graphite, impregnated with U^{235} and a substance whose thermal neutron cross section increases with increasing neutron temperature. Er is a suitable material for this purpose. Formulas and curves relating to the heat production by capture γ radiation in slabs have been worked out as an aid to reactor design work. (auth)

28.

2126 WAPD-TM-24

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

AN ALUMINUM–URANIUM CRITICAL ASSEMBLY IN CYLINDRICAL GEOMETRY. S. H. Levine and P. G. Johnson. Nov. 1956. 30p. Contract AT-11-1-GEN-14.

$6.30(ph OTS); $3.00(mf OTS).

An Al–U, heterogeneous, water-moderated critical assembly having a metal-to-water ratio (MW) of 1.39 was constructed to measure criticality and reflector savings and the age of the system. The age is inferred from the nonleakage probability. The multiplication curve is somewhat irregular and tends to change it's direction of curvature as it approaches criticality. The age calculated from the measured buckling is in fair agreement with previous experiments. (F.S.)

29.

2139

EXPONTIAL PILE EXPERIMENTS WITH NATURAL URANIUM AND HEAVY WATER. R. Persson, E. Bloms jö, M. Bustraan, and R. Meier (AB Atomenergi, Stockholm, Sweden). J. Nuclear Energy 3, 188-206(1956) Oct.

The material buckling of various hexagonal lattices of natural uranium rods (diameter 2.00, 2.53, and 3.05 cm) in heavy water has been determined in an exponential assembly. The diameter of the tank was only 1.0 metre. The detectors used were long BF_3 proportional counters, which give a high counting rate and by a special arrangement eliminate higher radial harmonics. For comparison between different lattices the measurements yield the material buckling to an accuracy better than $\pm 0.05 \cdot 10^{-4}$ cm^{-2}, but the absolute value is less accurate, owing to the uncertainty of the extrapolated radius (±0.2 cm). The multiplication factor is calculated using published values of the disadvantage factor, and the constant η and the resonance integral are evaluated. The possibility of correlating intensity and diffusion length is pointed out. (auth)

30.

9089 AERE-Inf/Bib-112

Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.

PHYSICS OF HETEROGENEOUS, LIGHT WATER MODERATED REACTORS. A BIBLIOGRAPHY OF PUBLISHED AND UNCLASSIFIED REPORT LITERATURE. A. Potter. Jan. 1957. 24p.

1958

31.

9451 WAPD-176

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

KINETIC AND BUCKLING MEASUREMENTS ON LATTICES OF SLIGHTLY ENRICHED URANIUM OR UO_2 RODS IN LIGHT WATER. J. R. Brown, D. R. Harris, F. S. Frantz, J. J. Volpe, J. C. Andrews, and B. H. Noordhoff. Jan. 1958. 67p. Contract AT-11-1-GEN-14. $2.00(OTS).

Results of measurements made in the TRX critical facility on light water-moderated lattices of 1.3 and 1.15 wt. % enriched uranium metal fuel rods and of 1.3 wt. % enriched UO_2 fuel rods are described. Diameters of 0.60 in. and 0.39 in. for each fuel have been used as well as densities of 7.5 and 10.5 gm/cm^3 for the UO_2 fuel. Several water-to-uranium ratios have been investigated for each fuel rod type. Measurements have been made of radial and axial buckling, reflector savings, and temperature coefficient of reactivity. A migration area is inferred from the change of reactivity with change in either radial buckling or axial buckling. Anisotropy of migration area is determined from measurements of buckling on lattices of various height-to-diameter ratios. (auth)

32.

16709 WAPD-BT-8(p.73-9)

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

CRITICAL EXPERIMENTS ON WATER-MODERATED LATTICES OF SLIGHTLY ENRICHED URANIUM DIOXIDE (UO_2) FUEL RODS. J. R. Brown. p.73-9 [of] BETTIS TECHNICAL REVIEW. REACTOR PHYSICS AND MATHEMATICS. 7p.

Critical assembly measurements were made to investigate some of the parameters of water-moderated lattices of slightly enriched UO_2 fuel rods. Measurements of buckling, reflector savings, temperature coefficient of reactivity, buckling coefficient of reactivity and anisotropy were made on eight lattices of varying H_2O ratio, UO_2 density, and fuel diameter. The experimental methods used and the results obtained are presented and the errors discussed. (auth)

33.

15020 A/CONF.15/P/595

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

EXPONENTIAL EXPERIMENTS WITH ORGANIC MODERATED URANIUM LATTICES. W. W. Brown. 12p. $0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

A series of exponential experiments was carried out on lattices of cylindrical uranium metal rods in a moderator of diphenyl. Three different concentrations of U^{235} in uranium were available for study, 0.50, 0.72, and 0.91 at. %. The fuel rods were arranged in square lattice arrays at moderator to fuel volume ratios 1.5, 2, 3, and 4. A cadmium clad $2\frac{1}{4}$ ft diameter aluminum tank was used to hold the lattices. External heaters maintained the contents at 185°F, just above the melting point of diphenyl. The lattices were driven by neutrons from the thermal column of a water boiler reactor. Foil activation techniques were employed to make the buckling and intracell flux distribution measurements. Measurements in pure and borated diphenyl were made in order to obtain the thermal neutron diffusion length and transport mean free path in diphenyl. Buckling and intracell measurements were made for some lattices in the borated diphenyl in order to determine the neutron age in these lattices. (auth)

34.
17423 TID-2506(Del.)(p.15-38)
North American Aviation, Inc., Downey, Calif.
MEASUREMENTS AND ANALYSIS OF URANIUM–D_2O LATTICES. Frank B. Estabrook and Sidney W. Kash. p.15-38 [of] NUCLEAR SCIENCE AND TECHNOLOGY. (EXTRACTS FROM NUCLEAR SCIENCE AND TECHNOLOGY. VOL. 1, ISSUES 1 TO 3, FEBRUARY–JUNE 1955). 24p.

During the past few years an extensive series of measurements on low-enrichment uranium-rod lattices in D_2O have been carried out at North American Aviation, Inc. This article briefly summarizes the measured values of the material bucklings, the fuel and moderator disadvantage factors, and the activity ratios for all the lattices and then gives the details and results of a careful two-group analysis into the consistency and significance of the measurements. The effective thermal-neutron temperature of each lattice is obtained from the ratio of absorption cross section to low-energy slowing-down power. The slowing-down power of D_2O in the thermal region is shown to be 0.173 cm^{-1}, and some agreement with experimental values of the average cosine in a scattering collision is adduced. Thermal utilizations are obtained from the experimental intracell measurements. Comparison is made with diffusion calculations of rod disadvantage factors and with excess absorption calculations for the moderator. Calculated in conventional fashion are η, ϵ, and L^2; in L^2 it may be noted that experimental disadvantage factors are used. The ratio of thermal to epithermal foil activities is shown to be a parameter characteristic of the asymptotic lattice spectrum and is successfully correlated with the predictions of two-group theory. The possible use of this experimental ratio for determining lattice diffusion lengths is critically examined. The neutron ages in the various lattices are computed by taking the lower limit of the age integral at an effective thermal energy some 16 times greater than would be proper in a nonmultiplying medium. The resonance escape probabilities that can be computed from the above quantities are then compared with resonance escape probabilities computed from the usual Fermi-Wigner-Weinberg theory. A series of small, but definite, discrepancies, functions of rod uranium enrichment, are found. These are believed to be real and independent of experimental scatter. Three supplemental experiments are recorded in the appendixes. These indicate the presence of a small amount of anisotropic diffusion, show the thermal utilization to be independent of large flux gradients that exist near the boundary of an exponential tank, and support the method used for computing effective neutron ages. (auth)

35.
2070 CF-56-7-148
Oak Ridge National Lab., Tenn.
PRELIMINARY REPORT OF CRITICAL EXPERIMENTS IN SLAB GEOMETRY. J. K. Fox and L. W. Gilley. July 30, 1956. Decl. Apr. 1, 1957. 15p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The critical parameters of aqueous UO_2F_2 solutions enriched to 90% U^{235} in various arrays of interacting slab geometries have been determined for a single concentration. Critical solution heights for various configurations of three nominally 3-in. thick Al slab vessels and one 6-in. Al slab vessel were determined. (T.R.H.)

36.
8738 UK/C/6/109
Atomic Energy of Canada Ltd. Chalk River Project, Chalk River, Ont.
LATTICE EXPERIMENTS WITH 19-ELEMENT UO_2 RODS IN ZEEP. [Paper presented at] Sixth UK/Canada Technical Conference, Chalk River, Ontario, October 21-22, 1957. R. E. Green. Sept. 16, 1957. 14p.

ZEEP is being used to study the lattice properties of 19 element UO_2 rods and two lattices are assembled in it. They are: (a) a 24.13 cm. hexagonal lattice of 55 UO_2 rods, and (b) a 21.59 cm. hexagonal lattice of 55 UO_2 rods with 30 ZEEP rods placed around the UO_2 core with the same spacing as the core. In the first experiment, the pile did not go critical and no flux distributions were obtained. However, an estimate of the Laplacian for this lattice was made using the results of the approach-to-critical experiment. In (b) full scale flux distributions, central rod experiments and fine structure measurements were made and practically all the results quoted will be those obtained with this 21.59 cm. lattice. (auth)

37.
2163 WAPD-PWR-Ph-112
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
SLAB EXPERIMENTS. II. URANIUM METAL REFLECTOR. W. H. Hartley. May 1956. 59p. $9.30(ph OTS); $3.60(mf OTS).

Part III issued as WAPD-PWR-Ph-126.

A series of experiments on various configurations of a slab made of a highly enriched seed flanked by a natural U reflector has yielded information on core reactivity, flux distributions, and temperature coefficients. The reactivity was obtained experimentally by integration of rod worth and water worth curves. The curves were constructed from points taken with several configurations of blanket material. This procedure is shown to be valid if the flux is separable. Calculated values of core reactivity are high compared to experimental values derived by use of the standard inhour equation. Shutdown reactivity was determined semi-empirically by measurements of negative buckling in a region of the core occupied by control rods. (auth)

38.
2164 WAPD-PWR-Ph-126
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
SLAB EXPERIMENTS. III. URANIUM OXIDE REFLECTOR. W. H. Hartley. July 1956. 41p. $6.30(ph OTS); $3.00(mf OTS).

Part II issued as WAPD-PWR-Ph-112.

Experiments similar to those performed on an enriched

slab seed with U metal reflector were made with a UO_2 reflector. Differential control rod worth and differential moderator worth curves were obtained and were compared with similar curves obtained from the metal reflected slab. From these curves, the excess reactivity of a slab was determined from a measurement of the slab critical height. Measurements of flux shapes in three directions were made in order to determine critical bucklings in the vertical and one horizontal direction and to investigate water channel flux peaking. (auth)

39.

16711 WAPD-BT-8(p.84-9)

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
CRITICALITY CALCULATIONS FOR THE TRX. R. B. Horst. p.84-9 [of] BETTIS TECHNICAL REVIEW. REACTOR PHYSICS AND MATHEMATICS. 6p.

The high U^{238} content of slightly enriched, water-moderated reactors makes the theoretical prediction of criticality difficult. Seven slightly enriched clean critical assemblies were constructed and their nuclear characteristics were measured and compared with those calculated for a theoretical, high U^{238} content core. While the design of a power reactor would introduce more complex problems, the agreement between the reactivity characteristics of the model and the seven critical assemblies obtained in these experiments is good and the model described herein should be useful in establishing reactivity characteristics of future cores of this type. (auth)

40.

6813 BNL-3145(Suppl.)

[Brookhaven National Lab., Upton, N. Y.]
CRITICAL ASSEMBLIES OF LIGHT WATER MODERATED, SLIGHTLY ENRICHED URANIUM ROD LATTICES AT BROOKHAVEN. Supplement to Hazards Summary Report. Herbert Kouts. Sept. 11, 1956. 25p.

41.

15076 A/CONF.15/P/1841

Brookhaven National Lab., Upton, N. Y.; Westinghouse Electric Corp. Bettis Atomic Power Div., Pittsburgh; and Combustion Engineering Corp., New York.
PHYSICS OF SLIGHTLY ENRICHED, NORMAL WATER LATTICES (THEORY AND EXPERIMENT). H. Kouts, R. Sher, J. R. Brown, D. Klein, S. Stein, R. L. Heliens, and H. Arnold. 70p. $0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

Measurements made with water moderated lattices of slightly enriched uranium rods since the last Geneva conference are presented and discussed. Additional measurements on assemblies having uranium oxide as the fuel are also presented. Improvements in technique are described and analyzed. The evolution of theoretical methods of interpreting and predicting the neutron behavior of reactor cores of this kind is recounted. The best present ways of analyzing the neutron economy and criticality are described and discussed, and comparisons with the available data are given. The methods range from multigroup treatments based on homogenization to a few group treatments which may be carried out by desk calculator methods. The derivation of the few group parameters from multigroup results is described. The theoretical treatments are supplemented by comparisons with the older four-factor methods. Finally, some measurements obtained with plutonium enriched rods are described, and tentative results are given. (auth)

42.

15019 A/CONF.15/P/594

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.
EXPONENTIAL EXPERIMENTS ON GRAPHITE LATTICES WHICH CONTAIN MULTI-ROD FUEL ELEMENTS. R. A. Laubenstein. 16p. $0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

A series of exponential experiments were carried out on graphite-moderated lattices containing multi-rod fuel clusters, in order to obtain information for design calculations on sodium-graphite power reactors. Measurements of the lattice buckling and intracell flux distribution were included. Experimental measurements of the critical mass with and without sodium in the core of the Sodium Reactor Experiment were intended as a check on the application of exponential experiment results to a critical assembly. (M.H.R.)

43.

1643 LA-749

Los Alamos Scientific Lab., N. Mex.
POLYTHENE-25 CRITICAL ASSEMBLY AND NEUTRON DISTRIBUTION STUDIES. Hugh C. Paxton and G. A. Linenbarger. Sept. 30, 1949. Decl. Oct. 17, 1957. 51p. Contract [W-7405-eng-36]. $9.30(ph OTS); $3.60(mf OTS).

Delayed critical behavior was investigated for an array of ½ in. polythene cubes and U^{235} cubes of average composition $UF_{3.8}C_{1.8}$, in an 8 in. thick natural U tamper. The pseudosphere of critical size (2130 cm^3) contained 12.0 kg of 94.5% U. Rossi time-scale measurements gave $\alpha = -.36 \times 10^4$ at delayed critical. Activity distributions for S, Au, U^{235}, and U^{238} samples were obtained as a function of radius of the assembly. Appreciable local variations in neutron spectrum resulted from the inhomogeneity of the active array. (auth)

44.

16730 AEC-tr-3362

EXPONENTIAL PILE MEASUREMENTS ON R3a FUEL ELEMENTS. Rolf Persson. Translated by W. K. Ergen (Oak Ridge National Lab.) from report AEF-65, Aktiebolaget Atomenergi, Stockholm, Sept. 19, 1956. 14p.

Clusters of seven uranium rods were investigated in an exponential assembly. The material buckling was

measured by an extrapolation procedure and the thermal diffusion length was established from the neutron intensity. (auth)

45.
15025 A/CONF.15/P/600
Argonne National Lab., Lemont, Ill.
PROPERTIES OF EXPONENTIAL AND CRITICAL SYSTEMS OF THORIA–URANIA AND HEAVY WATER, AND THEIR APPLICATION TO REACTOR DESIGN. W. C. Redman and J. A. Thie. 18p. $0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

Physics statics information was obtained through a series of heavy water exponential and critical experiments employing quarter-inch diameter fuel rods with thorium-U^{235} atom ratios of 25 and 50. Bucklings were determined for exponential triangular lattices of 1½ to 7½ in. pitch. Void and temperature coefficient of buckling also were measured. Systems of concentric zones having different bucklings were investigated. Critical experiments with a ¾ in. triangular lattice spacing and height to diameter ratios from 0.8 to 2.8 gave buckling and reflector savings information. Migration area was determined both by differential water worth and by variation of enrichment. Reactivity worth of voids and control rods was investigated. Thermal utilization and resonance escape were also measured. Physics considerations in boiling reactor design are assisted by these statics experiments and also the observed operating characteristics of BORAX-IV. In seeking compatibility among reactivity, void coefficient, conversion ratio, safety and stability, yet heeding boundary conditions imposed by mechanical, hydraulic and heat transfer considerations, judicious compromises are necessary. Illustrative of these design principles, a small 40 Mw ThO_2-UO_2-D_2O prototype boiling reactor having 400 kw/liter of coolant is presented. Although quite small, its attractiveness is further enhanced when extrapolated in size and power to those of practical interest. (auth)

1959

46.
6589 A/CONF.15/P/575
EXPONENTIAL EXPERIMENTS WITH URANIUM OXIDE CLUSTERS IN HEAVY WATER. E. Andersen and O. Aspelund (Joint Establishment for Nuclear Energy Research, Kjeller, Norway). 7p.

Measurements of the buckling of natural uranium oxide clusters in heavy water are presented. The clusters contain 6 and 7 oxide rods canned in aluminum. (A.C.)

47.
6590 A/CONF.15/P/1192
CRITICAL AND SUBCRITICAL EXPERIMENTS ON U-BeO LATTICES. P. Benoist, Ch. Gourdon, J. Martelly, M. Sagot, and G. Wanner (Commissariat à l'Énergie Atomique, Paris); and V. Deniz, B. V. Joshi, and K. Sahai (Atomic Energy Establishment, Trombay, India). 60p.

The material buckling of natural uranium–beryllium oxide lattices was measured. The square cell or element had a pitch of 15 cm and uranium rods of 2.60, 2.92, 3.56, and 4.40 cm. A critical experiment was conducted with hollow uranium rods enriched to 1.35%. (W.D.M.)

48.
7023 NAA-SR-3096
Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.
EXPONENTIAL EXPERIMENTS WITH GRAPHITE LATTICES CONTAINING MULTIROD SLIGHTLY ENRICHED URANIUM FUEL CLUSTERS. W. W. Brown, F. L. Fillmore, and B. L. Scott. Jan. 15, 1959. 46p. Contract AT-11-1-GEN-8. $1.25(OTS).

Studies of 13 exponential experiments with graphite moderated lattices containing multirod fuel clusters are presented. The bucklings and detailed intracell flux distributions were measured for each lattice. Average flux values for each material of the unit cell are given. The theoretical analysis yields a value of the effective resonance integral and of the resonance neutron inverse diffusion length in the moderator, which can be used in 2-group sodium graphite reactor calculations. There is evidence that neutron spectral hardening corrections are important, but a crude treatment of this effect did not improve the fit to the experimental measurements. The calculations are presented in detail, and various lattice parameters are tabulated. (auth)

49.
22211 NAA-SR-Memo-3980
Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.
CRITICAL MASS OF AN OMR CORE USING PLATE TYPE FUEL ELEMENTS. H. C. Field. June 8, 1959. 4p. $1.80(ph), $1.80(mf) OTS.

The clean critical mass of an OMR core was determined experimentally using low enrichment U plate-type fuel elements in a six-inch lattice spacing and Santowax R at 340°F as the moderator. Criticality was attained with 39 fuel elements (3915 kg) present in the core, three shim rods withdrawn, and the fourth shim rod about ⅞ withdrawn. (W.D.M.)

50.

2857 CF-58-8-3

Oak Ridge National Lab., Tenn.

CRITICAL EXPERIMENTS WITH 2.09% U^{235}-ENRICHED URANIUM METAL PLATES IN WATER. J. K. Fox, J. T. Mihalczo, and L. W. Gilley. Aug. 3, 1958. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Experiments were performed with 2.09% U^{235}-enriched uranium metal plates in a light-water-moderated and -reflected assembly. Each plate was 30 in. long, 3 1/8 in. wide, and 1/4 in. thick and contained 7.09 kg of uranium. In the first assemblies the plates were arranged in rows with edges adjacent, and the spacing between rows was varied from 5/8 to 1 1/8 in. The optimum spacing was about 7/8 in. With the spacing between rows maintained at 7/8 in., the spacing between the edges of the plates was varied from 0 to 5/8 in. The optimum spacing between edges was 5/16 in. The minimum critical mass in these assemblies was 6.74 kg of U^{235}. (auth)

51.

2853 CF-58-8-5

Oak Ridge National Lab., Tenn.

CRITICAL PARAMETERS FOR POISONED ANNULAR CYLINDERS CONTAINING AQUEOUS SOLUTIONS OF U^{235}. J. K. Fox and L. W. Gilley. Aug. 1, 1958. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Experiments were performed to determine the critical parameters of aqueous solutions of 93.2% U^{235}-enriched uranyl fluoride contained in cylindrical annuli formed by various combinations of aluminum cylinders varying in diameter from 2 to 30 in. In all of the experiments the inside cylinder was lined with a 20-mil-thick cadmium sheet and filled with water to a height of 48 in., and in some experiments a water reflector was used on the sides and bottom of the outside cylinder. The data indicate that for the solution having an $H:U^{235}$ atomic ratio of 50.4 the critical infinitely high reflected annulus would have a minimum thickness between 2.5 and 3 in., while the unreflected annulus would have a thickness between 3.75 and 4.5 in. The corresponding thicknesses for the solution having an $H:U^{235}$ atomic ratio of 309 would be between 3.5 and 4 in. for the reflected annulus and between 4.5 and 5.5 in. for the unreflected annulus. (auth)

52.

2859 CF-58-9-40

Oak Ridge National Lab., Tenn.

CRITICAL EXPERIMENTS WITH ARRAYS OF ORR AND BSR FUEL ELEMENTS. J. K. Fox and L. W. Gilley. Oct. 2, 1958. 6p. Contract [W-7405-eng-26]. $1.80 (ph OTS); $1.80(mf OTS).

Critical experiments were performed with ORR and BSR fuel elements to determine safe arrays in which the elements could be handled and stored. The data indicate that the optimum spacing for criticality of 168-g ORR elements in water is 0.2 in. between locating bosses and that an infinite array of vertically placed elements one element high would probably be subcritical with a 1 1/2-in. spacing between locating bosses. For uniform arrays of adjacent elements in water, variation in the fuel loading per element between 140 and 200 g made very little difference in the critical mass. When 132 elements with an average loading of 160 g per element were closely packed in water in an 11 by 12 element array in which the rows were separated with 20-mil-thick cadmium sheets, no appreciable source neutron multiplication was observed; nor was there any appreciable multiplication when both the cadmium sheets and the water moderator were removed and the array was surrounded with 12-in.-thick paraffin reflector. A two-row slab-shaped array with 24 200-g center elements and 14 168-g elements on each end, all spaced 0.2 in. between locating bosses, was subcritical, and it appears that two infinitely long rows of 168-g elements would be subcritical. (auth)

53.

7113 A/CONF.15/P/336

NATURAL URANIUM–HEAVY WATER LATTICES. Y. Girard, J. C. Koechlin, J. Moreau, and R. Naudet (Commissariat à l'Énergie Atomique, Paris). 97p.

A group of Laplacian determinations were made, under critical running conditions in the heavy water-natural uranium pile "Aquilon" constructed for this purpose, using either complete lattices or samples of lattices employing a two-zone method. The experimental equipment and methods of measurement are briefly described. Measurements were made for square lattice cells, going in 20 mm intervals from 130 to 250 mm. Results are presented for various bars. Other effects were measured. (W.D.M.)

54.

8252

CRITICAL EXPERIMENTS ON THE AQUILON REACTOR. Y. Girard, J. C. Koechlin, and J. M. Moreau. Bull. inform. sci. et tech. No. 14, 14-22(1958) Feb. (In French)

The Aquilon is a natural uranium research reactor cooled with heavy water. The shielding, auxiliary installations, and the control installations are described. A study was made of the critical conditions of the reactor, and an experimental determination of the buckling was also performed. The values obtained are given. (J.S.R.)

55.

4798 HW-56919

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

NUCLEAR PHYSICS RESEARCH QUARTERLY REPORT [FOR] APRIL, MAY, AND JUNE 1958. July 21, 1958. 76p. Contract W-31-109-Eng-52. $2.25(OTS).

The constants are calculated for the analytical solu-

tion for the thermal neutron flux in a nonabsorbing heavy gas with a temperature discontinuity, and an alternative method of deriving the group constants for Selengut's two-thermal-group approximations is given. Calculations are shown for thermal flux in a cell with temperature discontinuities by a method employing the formalism of few-group theory but retaining the qualitative features of neutron distribution. Results of buckling measurements for fuel elements in random vs. uniform array are summarized. Measurements of the number of fuel rods required for criticality for several lattices with 1.6% enriched U and light water moderator and reflector were completed using the "Approach-to-Criticality" method. In connection with processing of fuels enriched to 3%, K_∞ values were measured in PCTR for UO_3 hydrogen-moderated systems. A re-evaluation of η which results in more realistic critical parameters for natural U–light water lattices is reported. A summary is given of the empirical method used to estimate critical masses of oralloy vs. core density when surrounded by a full density natural U reflector. The techniques for measurement of K_∞ in PCTR are extended to the case where there is a temperature change in the fuel. Measurement of the metal temperature coefficient of K_∞ for 19-rod clusters of UO_2 in the PCTR is described. Expressions are given for the harmonic and end corrections for a three-region exponential pile. Measurement of material bucklings of seven-rod clusters of 0.925-in.-diam. natural U fuel elements in a graphite moderator was completed. Fission cross sections for Am^{241} and Np^{237} at 0.1 to 5 ev were measured using the crystal spectrometer. The weights of the U^{235} fission foils used at Hanford and Harwell for absolute fission cross section measurements are compared. Results of measurements of the age to In resonance of monoenergetic neutrons in kerosene and water are given. (For preceding period see HW-55879.) (T.R.H.)

56.

9472 HW-57861

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

NUCLEAR PHYSICS RESEARCH QUARTERLY REPORT [FOR] JULY, AUGUST, SEPTEMBER 1958. Oct. 20, 1958. 34p. Contract W-31-109-Eng-52. $1.00(OTS).

Theoretical Physics. Values are given for the numerical constant C_n in the equation for thermal neutron flux in a nonabsorbing heavy gas medium with a temperature discontinuity. Experimental Nuclear Physics. Errors in previous measurements of the slow neutron fission cross section of Am^{241} made new measurements on a new sample necessary. The new results are given. Critical Mass Physics. Results are given from the approach-to-critical experiments with 3.01 wt. % U^{235} U fuel elements, 0.925-in. outside diam., arranged in light-water moderated and reflected hexagonal lattices of various spacings: Results of buckling calculations for 3%-enriched UO_2 rod–water lattices and 5%-enriched U rod–water lattices are given. The development of a semi-empirical formulation for simplifying calculations of the fast effect for U metal and UO_2 fuel elements in light water is presented. Fuel element enrichment, rod size, and lattice spacing are the variables considered. (For preceding period see HW-56919.) (T.R.H.)

57.

19691 NP-7762

Iowa State Coll., Ames.

OPERATING CHARACTERISTICS OF A URANIUM GRAPHITE SUBCRITICAL ASSEMBLY WITH COOLANT SIMULATION (thesis). John Henry Hoganson. 1957. 88p.

Experimental and theoretical investigations with uranium–graphite subcritical assemblies were carried out to determine the various nuclear constants of a lattice. The lattice constants were the material buckling, multiplication constant, lattice diffusion length, and thermal utilization of the unit cell including the process tube assembly. In this investigation six lattice configurations were considered. (W.D.M.)

58.

8237 HW-58678

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

CORRELATION OF EXPONENTIAL AND PCTR MEASUREMENTS ON CLUSTER FUEL ELEMENTS WITH THEORY. G. D. Joanou. Dec. 31, 1958. Decl. Feb. 11, 1959. 21p. Contract [W-31-109-Eng-52]. $4.80(ph), $2.70(mf) OTS.

Recent improvements in methods of lattice parameter calculation are reported. By refinement of conventional calculation methods, successful applications to cases of cluster fuel elements were found. The model used is described, and the resonance escape probability integral, fast effect, fission neutron release to neutron fuel capture ratio, and thermal utilization are calculated. In addition, the diffusion length and Fermi age were calculated as well as neutron streaming and density corrections. The experimental work is described, and a comparison of theoretical and experimental results is made. (J.R.D.)

59.

22209 HW-61547

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

CRITICALITY OF FUELS OF LOW ENRICHMENT IN CYLINDERS. N. Ketzlach. Aug. 14, 1959. 12p. $3.30(ph), $2.40(mf) OTS.

Curves are presented of critical masses as a function of cylinder diameter and fuel rod size for 1.6, 3.0, and 5.0% U^{235} enriched metal-water systems. (W.D.M.)

60.
125 BNL-486
Brookhaven National Lab., Upton, N. Y.
EXPERIMENTAL STUDIES OF SLIGHTLY ENRICHED URANIUM, WATER MODERATED LATTICES. PART I. 0.600-IN.-DIAMETER RODS. Herbert Kouts and Rudolph Sher. Sept. 1957. 41p. $1.25(OTS).

Studies on 15 uranium–water lattices of 0.600-in.-diameter uranium rods are presented. Three fuel enrichments were used; these had nominal U^{235} contents of 1.0%, 1.15%, and 1.3%. Actual enrichments and densities for the fuel is given. Five ratios of water-to-uranium volumes were studied with each fuel enrichment. The pertinent geometrical data on these lattices are given. (W.L.H.)

61.
7104 A/CONF.15/P/245
LES PROPRIETES PHYSIQUES DE QUELQUES ELEMENTS COMPOSES, DANS DES REACTEURS DU TYPE U–D_2O. (Physical Characteristics of some Composite Natural Uranium Fuel Elements in Heavy Water.) P. Lehmann, R. W. Meier, and J. P. Schneeberger (Sulzer Bros. Ltd., Winterthur, Switzerland). 21p.

Lattices of natural uranium and heavy water were investigated for two types of fuel elements—the rod cluster and tube cluster types. The geometric dimensions of uranium are determined by thermal considerations assuming a surface heat flux of 100 w/cm² and fixing the number of subelements per cluster. The specific power (w/g of fuel) is a parameter varied within reasonable limits. New methods for the calculation of the lattice parameters ϵ, p, and f were employed, including recently published American, Canadian, and French methods. Some further refinements for the resonance capture of interior surfaces of the composite elements were developed which are applicable to systems with a coolant material different from the moderator. Finally, the material buckling is obtained for the cold-clean as well as the hot-poisoned case. The methods of calculation are then applied for other subdivided elements very similar in geometric shape for which the material buckling is known from exponential or critical experiments. (auth)

62.
18252 WAPD-TM-130
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
CRITICAL EXPERIMENTS WITH A SEED AND BLANKET SLAB GEOMETRY ASSEMBLY. S. H. Levine, E. J. Hanraty, P. G. Johnson, L. A. Mikoleit, S. L. Shufler, and W. F. G. Vogelsang. July 1959. 95p. Contract AT-11-1-GEN-14. $2.25(OTS).

A series of critical experiments was performed on slab geometry critical assemblies to study the basic behavior of seed-blanket type cores. These experiments included water-worth measurements, inference of excess and shutdown reactivities, temperature coefficients, three-dimensional activations, control rod worths, and the effect of burnable poison (boron). In most cases, theoretical calculations are compared to the experimental results, and the agreement between experiment and theory, in general, is quite good. (auth)

63.
16919 HW-57853
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
BUCKLING MEASUREMENTS OF FUEL ELEMENTS IN RANDOM ARRAYS. R. C. Lloyd. Oct. 21, 1958. 8p. Contract W-31-109-Eng-52. $1.80(ph), $1.80(mf) OTS.

The buckling was measured for three different fuel element types in random arrays. The average value of the buckling was less than the buckling of uniform arrays in all three cases, but in view of the 95% confidence limit, it is not considered advisable to increase the safe critical limits of random arrays above those of uniform arrays. (D.E.B.)

64.
17365
NEUTRON MULTIPLICATION MEASUREMENTS OF HETEROGENEOUS 3.1 PER CENT ENRICHED URANIUM–WATER SYSTEMS. R. C. Lloyd, R. B. Smith, and E. D. Clayton (General Electric Co., Richland, Wash.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 62-3(1959) June.

The design of efficient systems for processing spent power reactor fuel elements requires information on the critical masses involved. To obtain some of the needed information, approach-to-critical experiments were carried out with light-water-moderated and-**reflected hexagonal-lattice assemblies using enriched solid U fuel elements. Experiments were made for fuel elements of 0.925, 0.600, and 0.300-in. O.D. with the H_2O/U ratios for lattice spacings to span maximum buckling and minimum critical mass. (W.D.M.)**

65.
17049 BNL-536
Brookhaven National Lab., Upton, N. Y.
NUCLEAR ENGINEERING DEPARTMENT PROGRESS REPORT [FOR] JULY 1–SEPTEMBER 30, 1958. F. Maslan, ed. 76p. $2.25(OTS).

The study of effective resonance integrals in a heterogeneous system was continued. Measurement of the buckling and reflector savings of the 0.250-in.-diam. rod, water-moderated lattices was completed for both 1.027 and 1.5% enriched U metal.

66.
19051 CF-59-7-87
Oak Ridge National Lab.
MULTIPLICATION MEASUREMENTS WITH HIGHLY ENRICHED URANIUM METAL SLABS. J. T. Mihalczo

and J. J. Lynn. July 27, 1959. 35p. Contract [W-7405-eng-26]. $6.30(ph), $3.00(mf) OTS.

A series of neutron multiplication measurements with arrays of 1 by 8 by 10 in. slabs of 93.4% U^{235}-enriched uranium metal was made to provide data from which safety criteria for the storage of these fissile units can be established. Each slab contained 22.9 kg of U^{235}. A maximum of 125 units was assembled. The arrays studied were cubic lattices of the units and were usually parallelepipedal in shape. Arrays were both unmoderated and Plexiglas-moderated and were surrounded in most cases by a 1-in.-thick Plexiglas reflector. The lattice densities (ratio of fissile unit volume to lattice cell volume) were between 0.023 and 0.06. Unmoderated lattices with a density of 0.06 would require 145 ± 5 units for criticality, while those with a density of 0.023 would require 350 ± 30 units. In lattices in which the fissile units are separated by 1 in. of Plexiglas, approximately 27 units would be required for a critical array with a lattice density of 0.06 and about 75 units for a density of 0.023. Distributing Foamglas (containing 2% boron) throughout a moderated array increased the critical number of fissile units by a factor of 5, while Styrofoam had a small effect. (auth)

67.

10629 CRRP-648

Atomic Energy of Canada Ltd. Chalk River Project, Chalk River, Ont.

LATTICE EXPERIMENTS FOR CLUSTERS OF NINETEEN 1.31 CM DIAMETER URANIUM METAL RODS IN HEAVY WATER. A. Okazaki, D. F. Allen, and D. W. Hone. Jan. 1959. 40p. $1.00(AECL).

The natural uranium metal fuel was in the form of cylindrical slugs 1.31 ± 0.01 cm diam. by 15 cm long fitted loosely into aluminum tubes 210 cm long. The clusters were suspended by a gimbal arrangement on beams which were placed across the top of the pile. The buckling of the lattice under investigation was obtained from measurements of the macroscopic flux distribution through the reactor. The thermal flux distribution through the central cell was measured to obtain the thermal utilization and thermal diffusion area. The slowing down area for the lattice was calculated. In addition to the fine structure measurements made with manganese wires, the distribution of neutron capture by U^{238} through the central cluster was observed. (W.D.M.)

68.

7071 A/CONF.15/P/151

CALCULATION OF LATTICE PARAMETERS FOR URANIUM ROD CLUSTERS IN HEAVY WATER AND CORRELATION WITH EXPERIMENTS. B. Pershagen, G. Andersson, and I. Carlvik (AB Atomenergi, Stockholm). 24p.

Methods have been developed for predicting the buckling of cold clean lattices of uranium metal or uranium oxide (UO_2) rod clusters in heavy water. The methods involve refinements and modifications of the standard two-group recipe and have been correlated with exponential and critical experiments. In principle, the lattice parameters ϵ, p, f, L^2, and τ are calculated, and η is determined from measurements of the buckling. (auth)

69.

6587 A/CONF.15/P/160

EXPONENTIAL EXPERIMENTS ON HEAVY WATER–NATURAL URANIUM METAL AND OXIDE LATTICES. R. Persson, E. Blomsjö, J. Döderlein, E. Andersen, and O. Aspelund (AB Atomenergi, Stockholm). 23p.

A description is given of some unique features of the experimental facility. The theory of mixed lattices is discussed. Measurements are reported on the buckling for lattices consisting of clusters of uranium metal and oxide rods in D_2O at room temperature, the temperature coefficient between 10 and 80°C for D_2O and for different lattices, and control rod effects. (W.D.M.)

70.

9432 BNL-483(p.61-72)

Argonne National Lab., Lemont, Ill.

AGE AND RESONANCE ESCAPE PROBABILITY FOR THE THUD CRITICAL. W. C. Redman. p.61-72 [of] THORIUM–U^{233} SYMPOSIUM, SPONSORED BY THE UNITED STATES ATOMIC ENERGY COMMISSION AT BROOKHAVEN NATIONAL LABORATORY, JANUARY 9-10, 1958. 12p.

The term THUD designates a system of ThO_2–UO_2 fuel elements and heavy water currently under investigation in both exponential and critical experiments. The specific object of the THUD program at its outset was to develop the information required for the design of a heavy water–oxide fuel loading for the EBWR. The geometrical buckling for a ¾-inch triangular lattice of 25:1 Th to U^{235} fuel rods was determined from a sequence of seven clean criticals. Two independent methods have been employed to establish experimentally the neutron age for the THUD system. Values of the resonance escape probability for the ¾ inch triangular lattice have been obtained by fuel and analysis of the fuel substitution experiments. Several experimental determinations have been made of the effective resonance integral for Th and ThO_2. (W.L.H.)

71.

1802 AERE-T/M-166

United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.

AN INVESTIGATION INTO THE CAUSES OF A DISCREPANCY IN THE THEORETICALLY PREDICTED K_∞ FOR ENRICHED URANIUM HEAVY WATER LATTICES. K. T. Spinney. June 1958. 24p.

The discrepancy between theoretical and experimental values of the reactivity of enriched uranium–heavy water lattices was investigated. A description of the

experimental apparatus, known as DIMPLE, is given, along with a discussion of the procedure. It was concluded that all of the discrepancy could not be accounted for; however, some reasons are given, such as incomplete allowance for the inhomogeneity of the core resulting in an underestimate of the leakage. (J.R.D.)

1960

72.

7985 BAW-1179

Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.

THORIUM URANIUM PHYSICS EXPERIMENTS (TUPE). Monthly Report [for] October 1959. R. C. Anderson, M. L. Batch, R. H. Lewis, N. L. Snidow, and W. M. Vannoy. 27p. Contract AT(11-1)-766. OTS.

The Thorium–Uranium Physics Experiment (TUPE) is a study of uniform-lattice cores moderated with H_2O. The objectives of the TUPE project are: (1) to determine the critical size and mass for cores with M/W in the range 0.3 to 1.0 and for N_{O_2}/N_{25} = 25/1 and 15/1; (2) to determine nonleakage probability to allow determination of K_∞; (3) to separate k_∞ into fast and slow components; (4) to determine as many two-group parameters as possible in a critical experiment, including p_{O_2}, $\bar{\phi}$ water/$\bar{\phi}$ fuel, and k1/k2; and (5) to compare experimental results with predictions from various calculational models. A description of the cores, the status of the project, the results of calculations of the 25/1 lattices, and the results of measurements on Core 25 D are given. (W.D.M.)

73.

17270 YAEC-152

Westinghouse Electric Corp. Atomic Power Dept., Pittsburgh.

CRITICAL MASSES AND LATTICE PARAMETERS OF H_2O–UO_2 CRITICAL EXPERIMENTS. A COMPARISON OF THEORY AND EXPERIMENT. W. H. Arnold, Jr. Nov. 1959. 47p. For Yankee Atomic Electric Co. Contract AT(30-3)-222, Subcontract No. 1. OTS.

A semi-empirical method of performing lattice calculations in UO_2–H_2O cores is described which was used in the design of Yankee core 1. Calculated results using this method give close agreement with experimental results from the Bettis TRX facilities, from the Yankee and BR-3 critical experiments, and from a critical experiment performed for the NSS Savannah. Although the semi-empirical method gives adequate agreement, another system of calculation is presented which uses the MUFT code and has a somewhat firmer theoretical justification. It should prove valuable in the design of future cores. (auth)

74.

812 DP-407

Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.

PROCESS DEVELOPMENT PILE MEASUREMENTS OF LATTICE PARAMETERS OF NATURAL URANIUM IN HEAVY WATER. Norman P. Baumann. July 1959. 32p. Contract AT(07-2)-1. OTS

Details and complete results are given of a series of lattice studies performed in the Process Development Pile (PDP). The lattices studied covered a large range of configurations of natural uranium fuel in D_2O. Fuel assemblies consisted of single bare metal rods, clustered bare metal rods, clustered bare metal plates, metal tubes in aluminum housings, and clustered rods of aluminum-clad UO_2. Additional measurements were made with assemblies of metal plates inside gas-filled tubes. Triangular or equivalent lattice spacings were varied from 7.00 to 10.69 inches by the use of 7, 6, 5, 4, or 3 assemblies in the PDP test region. A detailed tabulation contains the measured bucklings and the moderator purity corrections for all lattices measured. The tabulation also includes calculated parameters, such as ϵ, p, f, L^2, and τ. Corrections required because of the irregular spacing of the test assemblies are discussed and listed for each lattice. The two-group, two-region method of analyzing critical water heights is described in detail. A comparison is made of the PDP measurements with similar measurements that were made at Chalk River, A.B. Atomenergi, Saclay, and North American Aviation. (auth)

75.

11288 BAW-1158

Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.

CRITICAL EXPERIMENTS WITH GENERAL ELECTRIC COMPANY FUEL FOR THE N. S. SAVANNAH. R. M. Ball, A. L. MacKinney, J. H. Mortenson, and H. F. Johnston. July 27, 1959. 79p. Contract AT(30-1)-2340. OTS.

Experimental data are tabulated as obtained from comparative measurements using fuel rods provided by the General Electric Company together with The Babcock and Wilcox MARTY pins. Both sets of fuel elements were clad with stainless steel and contained uranium dioxide enriched to 4% U-235. The results provide critical configurations and data for determining reactivity effects of spacer bars (flow blocks), reactivity effects of thick-walled fuel rods and corner poison rods, flux distributions in the G.E. fuel, differential water worths, reactivity effects of rod followers, and temperature effects. (auth)

76.

18613 BNL-297

Brookhaven National Lab., Upton, N. Y.

QUARTERLY PROGRESS REPORT [FOR] FEBRUARY 16–MAY 15, 1954. Decl. Apr. 8, 1957. 48p. OTS.

Buckling measurements with 0.600-in. fuel rods in clean water were completed for U enrichments of 1.3, 1.15, and

1.0%. Water-to-U volume ratios of 1, 1.5, 2, 3, and 4 to 1 were used. Values of $B^2 = +0.26 \pm 19 \times 10^{-4}/cm^2$ and $k = 1.0009 \pm 0.0006$ were obtained from a lattice of natural U rods in H_2O at a volume ratio of 1.5:1. A proposed shim control system for the LMFR is described which will maintain continuous control of the fuel concentration. In a test of the "torch" process for removing Pa and U from ThF_4 blankets by the fluorination of contaminants in a H_2-F_2 flame, 77% of the original feed lost 72% of its original activity. A method using fused salts was less successful. It was found that 92 to 95% of mixed fission product activity in molten $NaNO_3$ was scavenged out by a suspension of metallic oxides such as Mn and Fe. Isothermal studies of Th_3Bi_5 in liquid Bi and a liquid Pb–Bi eutectic showed that after 2000 hr, the compound particles grow larger in Bi, but thermal cycling tests with the same dispersions indicated that the particles grow larger in Pb–Bi. Static corrosion tests of B and W alloy 701 and chromized Croloy $2\frac{1}{4}$ showed no corrosion in Bi or U–Bi. Wetting experiments revealed that pure Fe can be wetted by heating to 1600°F under high vacuum and immersing in Bi at 725°C for 5 hr. Additional data are presented on the erosion of graphite by Th–Bi, the solubility of Fe and Zr in Bi, the adsorption of Zr on Fe from liquid Bi, the radiation damage of SiO_2, and the thermal cycling of U. The transmission of neutrons through an air slot was investigated as a function of slot dimensions. Parameters of several neutron resonance levels in U^{238} were obtained. (For preceding period see BNL-285.) (C.J.G.)

77.

26408 NAA-SR-5409

Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.

EXPONENTIAL EXPERIMENTS WITH GRAPHITE-MODERATED URANIUM–METAL LATTICES. R. W. Campbell and R. K. Paschall. Sept. 15, 1960. 32p. Contract AT-11-1-GEN-8. OTS.

Measurements of material buckling and intracell flux distributions were made in eight uranium-graphite exponential assemblies constructed from combinations of three square-cell sizes (7-, 9.5-, and 12-in.) and three enrichments (0.4962, 0.7205, and 0.9124 a/o) of 1-in.-diameter uranium metal fuel rods. Data from these assemblies in conjunction with single-element experiments indicate that the maximum buckling for the enriched fuel was approximately 4.65 m^{-2} at a 5.6-in. lattice spacing. Intracell thermal-flux distributions agreed to within 2 or 3% of values obtained from the S_4 approximation of the transport equation although the theory showed a systematically larger flux depression in the more highly enriched fuel, perhaps owing to neglect of spectral hardening. Resonance-escape probabilities were measured for the three lattice spacings of natural (0.7205 at. % U^{235}) fuel. The average value obtained for the effective resonance integral for U^{238} was 10.02 ± 0.40 barns. This value is in close agreement with a value of 9.77 barns obtained from the empirical formula $I = 27.8\sqrt{S/M + 0.04}$, deduced from Hellstrand's data. (auth)

78.

18572 HW-36174

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

EXPONENTIAL PILE MEASUREMENTS WITH HOLLOW SLUGS IN GRAPHITE–URANIUM LATTICES. E. D. Clayton. Apr. 12, 1955. Decl. Mar. 31, 1960. 32p. Contract W-31-109-Eng-52. OTS.

Buckling measurements were taken with hollow natural U slugs in graphite-moderated lattices, in which the lattice spacing was varied from $6\frac{3}{16}$ to 15 in. in 5 steps. The results of wet (water in the core) and dry measurements are reported. Buckling curves are given showing the effect on the buckling of coring the slug and placing water in the core. Simple three-region diffusion theory is applied to the measured lattices in an effort to correlate measurement with theory, but with only limited success. (auth)

79.

19689 HW-65328

General Electric Co. Hanford Atomic Products Operation, Richland Wash.

NUCLEARLY SAFE MASS LIMITS, VOLUME LIMITS, INFINITE CYLINDER DIAMETERS AND SLAB THICKNESSES FOR SLIGHTLY ENRICHED URANIUM RODS IN LIGHT WATER. E. D. Clayton. May 24, 1960. 15p. Contract AT(45-1)-1350. OTS.

A series of investigations involving critical approach and exponential measurements made with 3.06% enriched U is reported. Data from these experiments were used to calculate critical parameters and safe values which are shown graphically. The data pertain to U rods of diameters which in one case result in the minimum critical mass, and in the second case give the maximum buckling. An illustrative problem in nuclear safety is included in which mass and volume are discussed for a hypothetical dissolver used to process U fuel elements of 3.1% enrichment. (J.R.D.)

80.

22619 WCAP-1413

Westinghouse Electric Corp. Atomic Power Dept., Pittsburgh.

MULTI-REGION REACTOR LATTICE STUDIES OF FUEL CYCLE DEVELOPMENT PROGRAM FOR THE PERIOD JULY 1 TO DECEMBER 31, 1960. Ira H. Coen. Apr. 11, 1960. 47p. Contract AT(30-1)-2176. OTS.

Critical experiments at high water-to-metal ratios using 2.7% enriched UO_2 stainless steel clad fuel rods and the associated core components from the Yankee critical experiment were performed to obtain critical mass data in the range of moderating ratios close to the optimum. Experimental work for the loose-lattice experiments was completed. Calculations of the critical mass at moderating ratios close to the optimum were made. (See also WCAP-1414.) (C.J.G.)

81.

13048 HW-27921

Hanford [Atomic Products Operation], General Electric Co. Richland, Wash.

PHYSICS UNIT—APPLIED RESEARCH SUBSECTION TECHNICAL ACTIVITIES REPORT FOR APRIL 1953. J. E. Faulkner, D. E. Davenport, and G. E. Duvall. May 4, 1953. Decl. Mar. 3, 1960. 15p. OTS.

The β spectra of Tl^{204} and Rh^{106} were determined. The results of neutron streaming calculations for Hanford graphite are reported. Buckling measurements on a $6^3/_{16}$ in. small slug lattice were completed. Critical lattice experiments and neutron distribution studies are also reported. (D.E.B.)

82.

2878 NAA-SR-Memo-3980

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

CRITICAL MASS OF AN OMR CORE USING PLATE TYPE FUEL ELEMENTS. H. C. Field. June 8, 1959. 4p. OTS.

The clean critical mass of a core using low enrichment U plate-type fuel elements in a six-inch lattice spacing, and using Santowax R at 340°F as the moderator was experimentally determined. Criticality was attained with thirty-nine fuel elements, containing 3915 kg of U, present in the core, three shim rods withdrawn, and the fourth shim rod about $^7/_8$ withdrawn. The interpolated clean critical mass was determined to be 3910 kg of U. (W.L.H.)

83.

9183

EFFECTS OF THE CONTAMINATION BY LIGHT WATER ON LATTICES OF NATURAL URANIUM RODS IN HEAVY WATER. Bernard S. Finn and James W. Wade (E. I. du Pont de Nemours & Co., Aiken, S. C.). Nuclear Sci. and Eng. 7, 93-6(1960) Feb.

The effects of H_2O contamination on lattices of natural uranium metal in D_2O were measured. The buckling changes associated with H_2O contamination were determined for two lattices with moderator-to-fuel volume ratios of 12.3 and 14.6 over a range of H_2O concentrations from 0.2 to 8.2 mole %. The agreement between calculated and experimental changes in buckling for these lattices was within $\pm 25 \times 10^{-6}$ cm^{-2}. Similar measurements on seven other lattices with moderator-to-fuel ratios in the range from 31 to 212 were made for a change in the H_2O concentration from 0.18 to 3.92 mole %. For these measurements the experimental change in buckling was about 15% greater than the calculated change. (auth)

84.

20783 RFP-182

Dow Chemical Co. Rocky Flats Plant, Denver.

CRITICALITY STUDIES OF ENRICHED URANIUM METAL IN $UO_2(NO_3)_2$ SOLUTIONS. A. Goodwin, Jr., C. L. Schuske, and G. H. Bidinger. July 28, 1960. 17p. Contract AT(29-1)-1106. OTS

Neutron multiplication measurements were made on 6.5-in. diam cylindrical assemblies of enriched U discs immersed in aqueous solutions of enriched $UO_2(NO_3)_2$. Diffusion calculations were made on homogeneous mixtures of the enriched U with varying H:U atomic ratios. (auth)

85.

18334 LA-2026

Los Alamos Scientific Lab., N. Mex.

CRITICAL MASSES OF ORALLOY LATTICES IMMERSED IN WATER. J. C. Hoogterp, G. E. Hansen, H. C. Paxton, and D. P. Wood. Nov. 1955. Decl. May 18, 1960. 55p. Contract W-7405-eng-36. OTS.

A solid cube of oralloy becomes critical at 24 kg when immersed in an infinite water reflector. Various critical lattices were obtained by dividing this solid shape into small units and uniformly dispersing them at various mean densities. For a given size of oralloy unit, there is a mean density at which the critical mass is a minimum. The H to U atomic ratio of the cores with minimum critical masses was determined. Measurements with nonuniformly dispersed oralloy do not indicate a critical mass below the minimum observed with a uniform lattice. Multiplication measurements with Au, Ag, and Cd rods inserted in the oralloy matrix yielded the effective cross section ratios: σ_a (Ag)/σ_a (Au) = 0.86, and σ_a (Cd)/σ_a (Au) = 1.58. These values are independent of position and lattice spacing for ranges examined. (auth)

86.

16424

International Atomic Energy Agency, Vienna.

HEAVY WATER LATTICES. 1960. 142p. $1.50.

Reactor physicists from Chalk River, Saclay, Kjeller, Harwell, Savannah River, A.B. Atomenergi, and Argonne attended the panel meeting on the Physics of Heavy Water Lattices. The research programs of the various laboratories were discussed and the existing data were compared and evaluated. A series of 14 technical papers is presented in the appendix; they cover practical aspects of heavy water reactor lattices such as operating experience with the NRU, definitions of lattice parameters, lattice buckling, resonance escape probability, and reactor spectra. (D.L.C.)

87.

2877 NAA-SR-Memo-3892

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

CALCULATIONS ON THE TRX CRITICAL ASSEMBLY. L. Maki and W. Allen. [1959?]. 5p. OTS.

The following items were calculated for the different lattice spacing and were compared with the experimental results: resonance escape probability of U^{238}, relative thermal flux in fuel and moderator, and thermal utilization and effective multiplication. (W.L.H.)

88.

20781 HW-63576(p.70-6)

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

SHAPE PERTURBATIONS IN CRITICAL EXPERIMENTS. W. A. Reardon and R. C. Lloyd. p.70-6 of NUCLEAR PHYSICS RESEARCH QUARTERLY REPORT [FOR] OCTOBER, NOVEMBER, DECEMBER 1959.

When making heterogeneous critical mass measurements, several perturbations of unknown magnitude are usually present, such as the effect of an irregular outer boundary. Critical approach type measurements were made with circular, elliptic, and rectangular cylinders to try to evaluate some of the effects. The uranium rods used were 23.5 inches long, 0.175 inches in diameter, and 3.063% U^{235}. These were encased in 0.025-inch wall Lucite tubes and were arranged in a 0.5-inch triangular lattice; the resulting H_2O/U (total) volume ratio was 8.0. The assemblies were both fully reflected and moderated with H_2O. (auth)

89.

4726 RFP-169

Dow Chemical Co. Rocky Flats Plant, Denver.

ISOLATION THICKNESSES OF WATER FOR UO_2F_2 SOLUTION SYSTEMS. C. L. Schuske and A. N. Nickel. Dec. 21, 1959. 11p. Contract AT(29-1)-1106. OTS.

Previously published data were used for an empirical study of critical arrays of slab and cylindrically shaped vessels interacting through water. These vessels contained an aqueous solution of UO_2F_2 and the U^{235} isotopic enrichment was greater than 90%. Approximate isolation thicknesses of water for each of the above two cases are given. (auth)

90.

9910 RFP-174

Dow Chemical Co. Rocky Flats Plant, Denver.

INTERACTION OF TWO METAL SLABS OF PLUTONIUM IN PLEXIGLAS. C. L. Schuske, A. Goodwin, Jr., G. H. Bidinger, and D. F. Smith. Dec. 28, 1959. 14p. Contract AT(29-1)-1106. OTS.

Neutron multiplication measurements were performed on two identical finite Pu-metal slab assemblies separated and reflected by Plexiglas. (auth)

91.

7837 NAA-SR-4238

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

EXPONENTIAL EXPERIMENT WITH A THORIUM–URANIUM FUEL IN GRAPHITE. C. H. Skeen, G. L. Blackshaw, and W. W. Brown. Mar. 1, 1960. 25p. Contract AT-11-1-GEN-8. OTS.

An exponential experiment with multirod fuel elements in graphite moderator is described. The fuel is a thorium–uranium alloy. The buckling, thermal utilization, and resonance escape probability for one lattice configuration have been measured. In the analysis, emphasis is placed on determining the effective resonance integral of the fuel element. From the measured resonance escape probability, 0.856 ± 0.008, an integral (including 1/v capture) of 9.7 ± 0.8 barns is obtained. From the measured buckling (10.89 ± 0.05 m^{-2}) and thermal utilization (0.9445 ± 0.0009) a value of 11 ± 2 barns is obtained for the integral. Although in agreement with one another, these values are not in general agreement with either calculated values or the results inferred from other measurements. (auth)

92.

21077 BAW-1191

Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.

THORIUM URANIUM PHYSICS EXPERIMENTS FINAL REPORT. N. L. Snidow, R. C. Anderson, M. L. Batch, G. A. G. deCoulon, R. H. Lewis, and W. M. Vannoy. May 1960. 213p. Contract AT(11-1)-766. OTS.

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 Al and have metal-to-water volume ratios, M/W, ranging from 0.3 to 1.0. The basic aim of the project was to measure lattice parameters of cores with ratios ranging from 0.3 to 1.0, and to compare the results with different theoretical models. Results of the measurements on various cores are discussed in detail. (See also BAW-1193.) (W.D.M.)

93.

7087 NAA-SR-Memo-3872

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

FLUX DISTRIBUTION MEASUREMENTS IN OMR CORES. V. A. Swanson. May 14, 1959. 14p. OTS.

The neutron flux distributions in a flexible mockup of an OMR core portion were measured. Two lattice configurations were studied. The diffusion length of thermal neutrons in the moderator and the critical sizes of the lattices were also measured. Results, together with some comparisons with theoretical calculations, are tabulated. (J.R.D.)

94.

7005 (NAA-SR-Memo-4240) SRE CRITICALITY WITH SEVEN SOLID ROD CLUSTER OF URANIUM CARBIDE FUEL. A. Aronson (Atomics International, Div. of North American Aviation, Inc., Canoga Park, Calif.). Aug. 6, 1959. 13p. OTS.

Studies were made to determine the enrichment required to provide excess reactivity of 4.5% above hot and poisoned for various numbers of clusters of UC fuel in the SRE. The fuel element is in the form of seven-rod clusters 6-ft. long. A plot is given of effective multiplication constants (K_{eff}) vs. enrichment for cores consisting of 31, 37, and 43 clusters. The effects of additional clusters on reactivity with constant enrichment are shown. Cell thermal flux and core radial and axial thermal flux plots are given for 31 clusters with 5.0 at.% enriched fuel and are also plotted for 43 clusters with 4.0 at.% enriched fuel. (W.L.H.)

95.

25575 (BAW-1220) SPECTRAL SHIFT CONTROL REACTOR BASIC PHYSICS PROGRAM, QUARTERLY TECHNICAL REPORT NUMBER 3, JANUARY–MARCH 1961. (Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.). Contract AT(30-1)-2602. 82p.

A summary of the work accomplished during the third contract quarter (January 1 to March 31, 1961) on the Spectral Shift Control Reactor (SSCR) Basic Physics Program is presented. The major objective of the program is to determine basic physics parameters of tight-packed lattices of slightly enriched fuel in moderators consisting of D_2O-H_2O mixtures. The concentration of the D_2O-H_2O mixtures are varied so as to apply to the spectral shift concept. The required license amendment was issued, and the first critical experiment containing heavy water in the moderator (76.7 mole % D_2O) was performed with 4%-enriched UO_2 fuel rods. Reported are measurements of the critical mass, critical buckling, thermal disadvantage factor, cadmium ratio of U^{235} and U^{238}, epithermal neutron spectrum, and the effect of moderator channels. The operating license for the room temperature exponential experiments was issued, and preliminary buckling measurements with $(U^{235} + Th)O_2$ fuel in a moderator containing 80 mole % D_2O were made. The results of hot exponential experiments at 70 to 400°F with 4%-enriched UO_2 fuel in light water moderator are also reported. Neutron age measurements in a lattice of ThO_2 rods in light water and 90 mole % D_2O, parallel and perpendicular to the rod axis, were completed and a preliminary analysis of the results is presented. Theoretical studies continued. The BPG computer code, which will be used to analyze experiments with D_2O in the moderator, was refined by improving the methods of computing resonance absorption and the transport cross sections for fast neutrons. Alternate methods of computing Dancoff shielding in the lattice and resonance absorption in fertile material were considered, and BPG and P1MG calculations were compared to assess the importance of the choice of neutron spectrum in the reflector. Additional calculations in support of the experimental work also continued. (auth)

96.

21400 BUCKLING MEASUREMENTS OF D_2O MODERATED CLUSTERS OF NATURAL URANIUM OXIDE RODS. N. P. Baumann and E. J. Hennelly (E. I. duPont de Nemours and Co., Aiken, S. C.). Trans. Am. Nuclear Soc., 4: No. 1, 48-9(June 1961).

97.

13575 (HW-67179) MEASUREMENTS OF NEGATIVE k_{ex} FOR A GRAPHITE URANIUM LATTICE IN THE PCTR. E. Z. Block and D. E. Wood (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Dec. 1, 1960. Contract AT(45-1)-1350. 24p.

The values of k_∞ and f were measured for a cluster fuel element in a 7-in. graphite lattice. The value of k_{ex} was negative for natural uranium fuel; so two methods of measurement were compared: (1) the normal method of poisoning the lattice with thin strips of copper with an extrapolation of the result back to a k_∞ of one and (2) producing the condition of unit k_∞ by adding to the cell U^{235} as thin strips of U^{235}-aluminum alloy. A measurement was made for a slightly enriched fuel in the same geometry, with k_{ex} positive, to provide a check on the other methods used. The measured values of k_∞ agreed within the quoted error. The values of k_∞ and f are, respectively, 0.944 and 0.907 as measured by adding U^{235} to the cell containing natural uranium fuel. (auth)

98.

24540 EXPONENTIAL EXPERIMENTS ON NATURAL URANIUM-IMPREGNATED GRAPHITE LATTICES. R. Bonalumi, C. Bruschi, G. B. Zorzoli (Laboratori CISE, Segrate, Italy). Energia nucleare (Milan), 8: 321-5(May 1961). (In English)

Exponential experiments on natural uranium-impregnated graphite lattices are described. Buckling was measured on three different lattice pitches, corresponding to spacings of 10, 14.1, and 20 cm; the experimental results are discussed. (auth)

99.

28337 (HW-68405) A SEMI-EMPIRICAL METHOD OF ESTIMATING MATERIAL BUCKLINGS FOR SLIGHTLY ENRICHED URANIUM–WATER LATTICES. C. L. Brown (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 1961. Contract AT(45-1)-1350. 39p.

The calculated bucklings are compared with experimental bucklings measured on uranium rods in water at enrichments of 1, 2, and 3 per cent U^{235}, for rod diameters up to three inches. Twenty two such measurements are included. Bucklings are calculated using the one-group criticality equation. Lattice parameters are calculated. The maximum material bucklings obtained are biased high (conservative from the standpoint of nuclear safety). The bias, however, is not unreasonably large: for 1.0-inch diameter rods at 1.0 per cent U^{235}, the bias is 130 ± 320 microbucks; for 0.5-inch diameter rods at 3.0 per cent U^{235}, the bias is 190 ± 330 microbucks; and for 0.3-inch diameter rods at 5.0 per cent U^{235}, the bias is 320 ± 420 microbucks. Maximum bucklings and minimum critical masses are presented for water-reflected lattices of 1.03, 2.0, 3.063, and 5.0 per cent U^{235} enriched uranium rods in light water. Extrapolation lengths are estimated from experimental measurements. (auth)

100.

21399 EXPONENTIAL EXPERIMENTS IN URANIUM–DIPHENYL LATTICES. R. W. Campbell, T. L. Guzzle, and R. K. Paschall (Atomics International, Canoga Park, Calif.). Trans. Am. Nuclear Soc., 4: No. 1, 47-8(June 1961).

1961

101.

29882 (NAA-SR-6446) EXPONENTIAL EXPERIMENTS WITH HEAVY-WATER, GRAPHITE-AND-DIPHENYL MODERATED, URANIUM METAL LATTICES. R. W. Campbell and C. H. Skeen (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). Sept. 15, 1961. Contract AT-11-1-GEN-8. 37p

A series of experiments were carried out during the past several years to study the basic reactor physics of several clean lattices and their constituents. Three moderators, heavy water, graphite, and diphenyl were used, in turn, in these lattices. Three metal fuels (but not mixtures thereof) were used. The fuel enrichments (at.%) were 0.4962 (depleted), 0.7205 (natural) and 0.9124 (enriched). The fuel elements made from 4-inch long slugs, were in the form of a cylinder which had a diameter of one inch. These elements were 5 ft long. These experiments are summarized and simplified two group theoretical comparison of the results is presented. It is demonstrated that this model is not accurate for the description of these lattices. (auth)

102.

30213 (NAA-SR-6409) EXPONENTIAL MEASUREMENTS IN DIPHENYL-MODERATED, URANIUM-METAL LATTICES. R. W. Campbell, T. L. Guzzle, and R. K. Paschall (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). Sept. 15, 1961. Contract AT(11-1)-GEN-8. 23p.

Material buckling and intracell flux distribution measurements are made for a series of diphenyl-moderated, uranium-metal exponential assemblies. The lattices consist of square cells with a 1-in.-diameter fuel element located at the center and spaced to give moderator-to-fuel (M-F) ratios of approximately 1.5, 2, 3, 3.5, and 4. Fuel enrichments of 0.4962, 0.7205, and 0.9124 at.% are used. The lattice is maintained at a temperature of 180°F. Measurements are conducted for natural fuel with a 3.15 M-F ratio, and the buckling is found to be $-4.1 \pm 0.5\ m^{-2}$. Intracell flux distributions are measured for eight lattices and compared with calculations using cross-sections averaged over a Wigner-Wilkins spectrum. Agreement is good for small spacings but grows worse as the cell size increases, because the cell structure departs from the homogenized cell used to calculate the neutron spectrum. Thermal utilizations are calculated from both the measured and calculated flux distributions and are found to disagree by less than 2%. (auth)

103.

21784 ANALYSIS OF CRITICAL EXPERIMENTS WITH SLIGHTLY ENRICHED UO_2 CLUSTERS IN HEAVY WATER. J. D. Cleary (Westinghouse Electric Corp., Pittsburgh), B. Jennings, F. L. Langford, and W. H. Arnold, Jr. Trans. Am. Nuclear Soc., 4: No. 1, 102-3 (June 1961).

104.

10392 (WCAP-1423) MULTI-REGION REACTOR LATTICE STUDIES. Quarterly Progress Report [for the period] July 1 to September 30, 1960. Ira H. Coen (Westinghouse Electric Corp. Atomic Power Dept., Pittsburgh). Oct. 28, 1960. 53p. Contract AT(30-1)-2176.

Critical Experiments. After reports of fuel-element and control-rod fabrication progress, the critical experiments performed are outlined. These experiments included measurements on a 3.7% core, criticality measurements on a three-region core, peripheral fuel rod worth for a three-region core, fuel-rod and foil scans in a three-region core, multiple foil measuring techniques, analysis of errors in U^{238} Cd ratio measurements, and comparison of 3.7 and 2.7% cores. Nuclear Analysis of Multi-region Reactor Cores. Work continued on calculations of lattice parameters, comparison of results with experiment, and evaluation of methods of analysis. (For preceding period see WCAP-1419.) (T.R.H.)

105.

17753 (WCAP-1428) MULTI-REGION REACTOR LATTICE STUDIES. Quarterly Progress Report, October 1 to December 31, 1960. Ira H. Coen (Westinghouse Electric Corp. Atomic Power Dept., Pittsburgh). Jan. 30, 1961. Contract AT(30-1)-2176. 81p.

Results are reported for an extension of critical experiments and analyses which was made in the program for evaluation of water-moderated multiregion cores to study the discrepancy between calculated and experimental values of ρ^{28} (resonance capture/thermal capture ratio in U^{238}) and δ^{25} (resonance fission/thermal fission ratio in U^{235}). (auth)

106.

21728 (WCAP-1432) MULTI-REGION REACTOR LATTICE STUDIES. Quarterly Progress Report, January 1 – March 31, 1961. Ira H. Coen (Westinghouse Electric Corp. Atomic Power Dept., Pittsburgh). Apr. 1961. Contract AT(30-1)-2176. 78p.

Progress in analytical studies and critical experiments for the multi-region reactor lattice studies of the fuel cycle development program is described. The experiments utilized stainless steel clad uranium dioxide fuel rods of three different enrichments and two moderating ratios. Performance of the scheduled experiments with the 4.5:1 W/U lattice was continued without interruption. The experimental results are reported under the headings of criticality measurements, flux distributions, and microscopic parameter measurements. Criticality measurements were made on various two- and three-region cores. Results for loading, critical water height, banked rod position, and peripheral fuel rod woth are tabulated. The cross section schematic diagrams of the cores utilized are shown. The cadmium ratios for gold and U^{235} in the moderators were determined. Scans are shown of fuel rods, gold foils, and U^{238} foils for water slots and for various slab materials inserted in the slot. The parameters of the lattices studied experimentally were calculated and the results were compared with those obtained experimentally. (M.C.G.

107.

30223 (WCAP-1434) MICROSCOPIC LATTICE PARAMETERS IN SINGLE- AND MULTI-REGION CORES: A COMPARISON OF THEORY AND EXPERIMENT. P. W. Davison, J. D. Cleary, B. Jennings, H. A. Risti, and G. H. Minton (Westinghouse Electric Corp. Atomic Power Dept., Pittsburgh). June 1961. Contract AT(30-1)-2176. 187p.

Light water critical experiments are performed in order to measure microscopic parameters and conversion ratios in single region and multi-region cores containing slightly enriched, stainless steel-clad, UO_2 fuel rods. The experimental results are analyzed with multi-group codes, which are supplemented for the resonance energy region by a Monte Carlo code. Experimental and analytical results are presented, along with a description of investigations of discrepancies between theory and experiment. (auth)

108.

21782 REACTIVITY AND SPATIAL DISTRIBUTIONS IN MULTI-REGION LATTICES: COMPARISON BETWEEN ANALYSIS AND EXPERIMENT. W. J. Eich and W. P. Kovacik (Westinghouse Electric Corp., Pittsburgh). Trans. Am. Nuclear Soc., 4: No. 1, 100-1(June 1961).

109.

6999 (HW-64866(p.65-87)) MATERIAL BUCKLINGS OF GRAPHITE URANIUM LATTICES. G. W. R. Endres and D. E. Wood (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Material buckling measurements were completed for a series of lattices with a tube-in-tube natural uranium fuel element for most lattice spacings with tube-and-rod and I and E fuel elements and for one lattice, using a second tube-and-rod fuel element. Bucklings were measured in exponential piles for both air and water coolant. Material buckling values and side-to-side extrapolation lengths for the tube-in-tube, tube-and-rod, and I and E fuel elements are shown. Dimensions, cladding, densities, volume ratios, and volumes per centimeter of length for the fuel elements are given. (W.D.M.)

110.

13918 (NAA-SR-Memo-810) MEASUREMENTS AND ANALYSIS OF URANIUM-D_2O LATTICES. F. B. Estabrook and S. W. Kash (North American Aviation, Inc., Downey, Calif.). Nov. 16, 1953. Decl. Nov. 6, 1958. 28p.

A series of exponential measurements performed on uranium-heavy water lattices is summarized. Three enrichments of fuel materials were used: 0.49, 0.71 (natural), and 0.90 wt. % U^{235}. The uranium fuel slugs ranged from 0.75 to 2 in. in diameter and were encased in tubes of 0.40-in. aluminum. The lattice cell spacings varied from 3 to 12 in. The analysis of the data is divided into two parts. The first part deals directly with the neutron flux measurements and reports for each lattice, the material buckling (B^2), a pair of disadvantage factors (F and F_m), and a flux ratio (R). The second part of the analysis is concerned with the long-range problem of improving reactor calculations. Attempts were made to correlate the measurements with themselves and with other nuclear measurements so as to obtain a consistent formalism for computing uranium–heavy water reactor parameters. (W.L.H.)

111.

21403 k_∞ MEASUREMENTS IN THE PAWLING LATTICE TEST RIG. G. Foster, W. L. Brooks, M. Fleishmann, and R. D. Schamberger (Nuclear Development Corp. of America, White Plains, N. Y.). Trans. Am. Nuclear Soc., 4: No. 1, 51(June 1961).

112.

6594 (ORNL-3016(p.72-3)) CRITICAL DIMENSIONS OF RANDOM ARRAYS OF ThO_2–UO_2 PELLETS. J. K. Fox and L. W. Gilley (Oak Ridge National Lab., Tenn.).

Several criticality experiments were performed with assemblies of unclad ThO_2 - UO_2 pellets, randomly stacked in aluminum or steel cylinders. Water was added to the cylinders to serve as neutron moderator, and all assemblies were completely water-reflected. The results of these measurements and of preliminary experiments in which the pellets were contained in plastic bottles indicated that the critical dimensions of assemblies of such pellets are extremely sensitive to the degree of neutron moderation. (auth)

113.

21408 CRITICAL ARRAYS OF NEUTRON INTERACTING UNITS. L. W. Gilley and J. T. Thomas (Oak Ridge National Lab.,.Tenn.). Trans. Am. Nuclear Soc., 4: No. 1, 54-5(June 1961).

114.

21401 PRECISION CRITICAL SUBSTITUTION MEASUREMENTS OF BUCKLINGS IN D_2O MODERATED LATTICES. W. E. Graves (E. I. duPont de Nemours and Co., Aiken, S. C.). Trans. Am. Nuclear Soc., 4: No. 1, 49(June 1961).

115.

2067 WCAP-1412

Westinghouse Electric Corp. Atomic Power Dept., Pittsburgh.

MULTI-REGION REACTOR LATTICE STUDIES RESULTS OF CRITICAL EXPERIMENTS IN LOOSE LATTICES OF UO_2 RODS IN H_2O. V. E. Grob, P. W. Davison, D. F. Hanlen, R. D. Leamer, F. L. Kelly, and J. D. Cleary. Mar. 30, 1960. 53p. Contract AT(30-1)-2176. OTS.

The loose lattice experimental work performed under the Multi-Region Lattice Program conducted at the Westinghouse Reactor Evaluation Center is described. The fuel used in these experiments was the 2.73% en-

riched Yankee CRX fuel, described on page 14 of YAEC 94[1]. The purpose of the experiments was to obtain critical mass information in the range of moderating ratios close to the optimum and to verify the physics computational methods now used for fuel storage calculations. The experimental program consisted of critical size, buckling, reflector savings, and microscopic parameter measurements. (auth)

116.

21786 HETEROGENEOUS URANIUM METAL–GRAPHITE CRITICAL EXPERIMENTS. C. A. Guderjahn (Atomics International, Canoga Park, Calif.). Trans. Am. Nuclear Soc., 4: No. 1, 104(June 1961).

117.

13459 (HW-67219) NUCLEAR PHYSICS RESEARCH QUARTERLY REPORT, JULY, AUGUST, SEPTEMBER 1960. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 20, 1960. Contract AT(45-1)-1350. 114p.

Results are reported of the neutron-induced fission in low-energy resonances of Pu^{240} and of resonances in the target nuclides Pu^{242}, U^{236}, U^{238}, and Pa^{231}. A three-axis crystal spectrometer was constructed for the study of different inelastic scattering cross sections of neutrons in the energy range from near thermal to a few tenths of an ev. Results are reported of the scattering of 0.147-ev neutrons from water. Hanford's 2-Mev positive ion Van de Graaff accelerator was equipped with motor-driven supports, which permit mechanical positioning of the accelerator from a remote station in the control room. The electrode system installed to determine the location of the ion beam within the vacuum system is described. The system is designed to furnish information necessary for the positioning of the accelerator and further alignment of the beam. A study is reported of the position and energy dependence of the neutron flux in a heterogeneous system. The effect of a 1/v absorbing cylinder on the thermal flux in a surrounding nonabsorbing moderator is treated in an improved approximation. The basic improvement embodied in the solution is the use of a more accurate description of the energy dependence of the blackness of the absorber. A study was made on the interpretation of PCTR measurements of K_∞ for the case in which the addition of poison perturbs the neutron flux distribution in any or all of the various components of the test cell. Measurements were also made of K_∞ for lattices in which only one fuel element or one central cell is available. Lattice-cell flux traverses from experiments were examined in order to develop a method for calculating reaction rates in uranium-graphite heterogeneous systems. Variations in the neutron energy spectrum were examined to improve standard methods, especially for systems at elevated physical temperatures. The material buckling and extrapolation lengths were measured in three exponential piles. One pile was 6-ft wide and 8-ft long with a lattice spacing of $8^3/_8$ in. The second was 8-ft wide and 8-ft long with a lattice spacing of $8^3/_8$ in. The third pile was 10-ft wide and 8-ft long with a lattice spacing of $14^9/_{16}$ in. A tube-in-tube natural-uranium fuel element with water coolant was used in all three piles. Proposed studies for the critical facility of the Plutonium Recycle Program are reported. A series of exponential pile measurements were made to determine the criticality of 2.6% enriched uranium dioxide rods in light-water lattices. These measurements were made primarily to obtain data for establishing nuclear safety criteria relative to proposals for processing power reactor fuels. (For preceding period see HW-64866.) (W.L.H.)

118.

5572 WATER-MODERATED CORES WITH BORON STEEL SEPTA AT ELEVATED TEMPERATURES. G. D. Hickman, J. A. Bistline, and L. A. MacNaughton (Knolls Atomic Power Lab., Schenectady, N. Y.). Nuclear Sci. and Eng. 8, 381-92(1960) Nov.

A series of 15 experiments was carried out on an 8 × 30 × 32 in. core in the Pressurized Critical Assembly at KAPL. In 12 of these experiments, 0.030-in. boron stainless steel septa bisected the 8-in. dimension. These septa contained various weight per cent B^{10}. In the remaining three experiments, there were no boron-stainless steel septa in the core. The eigenvalues and neutron density distributions were compared with values calculated from Deutsch cross sections and "Thin Region Theory." The calculated eigenvalues were within 1% of the experimental values, with a spread of approximately 1%. For all the cores, the calculated eigenvalues were lower than the experimental values. Analyses of the neutron density distributions showed the calculated results in fairly good agreement with the experimental results. In all cases, this agreement was as good for the cores which contained the boron septa as for the ones which did not. It therefore appears that the boron was well represented by "Thin Region Theory," and that the main discrepancies between calculated and experimental values are due to the inadequacies of adapting the Deutsch scheme to these cores. (auth)

119.

21785 NATURAL URANIUM-D_2O BUCKLINGS OVER AN EXTENDED RANGE OF PITCH AND FUEL ASSEMBLY SIZE. T. J. Hurley, Jr., H. R. Fike, and G. F. O'Neill (E. I. duPont de Nemours and Co., Aiken, S. C.). Trans. Am. Nuclear Soc., 4: No. 1, 103-4(June 1961).

120.

28336 (CRRP-894) SOME CLOSE-PACKED LATTICES IN LIGHT WATER AND HEAVY WATER. PART I. BUCKLING MEASUREMENTS. R. G. Jarvis, G. J. Phillips, and W. H. Walker (Atomic Energy of Canada Ltd. Chalk River Project, Chalk River, Ont.). Apr. 1961. 40p. (AECL-1254)

Measurements of buckling were made in an exponential system for a set of twelve lattices. The fuel was in rods 1.28 in. in diameter, in aluminum cans, and was arranged in square lattices at spacings of 1.60, 2.11, and 3.20 in. At each spacing the buckling was measured for natural and

depleted uranium in light water and in heavy water. The depleted uranium contained 0.26% of U-235. (auth)

121.

7019 (ORNL-3016(p.76-8)) CRITICAL EXPERIMENTS WITH PRNC RESEARCH REACTOR FUEL ELEMENTS. E. B. Johnson and K. M. Henry, Jr. (Oak Ridge National Lab., Tenn.).

A series of critical experiments with the Puerto Rico Nuclear Center research reactor fuel elements was completed in the Pool Critical Assembly of the Bulk Shielding Facility. The experiments validated the computations on which the design fuel-plate loading was based, and they determined the amount of excess reactivity to be expected from three clean, cold loadings proposed by the reactor's designer as operational configurations. (auth)

122.

25377 (NAA-SR-Memo-6414) NUCLEAR SAFETY IN HANDLING 10 PER CENT U^{235}-ENRICHED UO_2 FUEL. Norman Ketzlach (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). May 16, 1961. 4p.

Criteria were established for the safe fabrication and storage of 9.87% U^{235} enriched UO_2 fuel rods. Due to the limited amount of fuel to be handled in this project, safe criteria were based on fuel rods at optimum rod size and lattice spacing and water flooding. When the fuel is handled in pellet form, 530 g contained U^{235} is the maximum safe batch size independent of container size. The maximum safe volume, independent of mass, is 6.5 liters. When handled as 0.30-inch diameter finished fuel rods, it is safe to handle 26 rods (920 g contained U^{235}). A two-dimensional infinite array of fuel elements (24-rod bundles) is safe with a minimum edge-to-edge separation of 24 inches. This same spacing between safe batches of pellets is also safe. Fifty columns (2.75 inches by 2.75 inches cross section) of fuel elements are safe with a minimum edge-to-edge spacing of 12 inches. (auth)

123.

25378 (NAA-SR-Memo-6415) NUCLEAR SAFETY CRITERIA IN USE OF RMDF TRANSFER CASK. Norman Ketzlach (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). May 16, 1961. 6p.

Criteria were established for the storage and transportation of a maximum of one kg contained U^{235} scrap in each 55-gal drum. These drums are stored four drums per pallet in a square lattice array. An array of such pallets, two pallets wide by two pallets high by six pallets long containing a total of 96 drums, is safe independent of the degree of water flooding. Two kg contained U^{235} is allowable per drum in a 180-drum, single-plane array. These arrays may be repeated with an edge-to-edge separation of at least 6 ft. (auth)

124.

6591 (HW-64866(p.143-7)) CRITICAL APPROACH AND EXPONENTIAL MEASUREMENTS WITH 3.1 PER CENT ENRICHED URANIUM RODS IN LIGHT WATER. R. C. Lloyd, E. D. Clayton, and R. B. Smith (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A series of exponential and critical approach measurements were previously made with 3.063% enriched uranium rods in water. The data are summarized, and plots are given of maximum buckling and minimum critical mass in spherical geometry vs. rod diameter and the extrapolation length vs. H_2O/U volume ratio. (W.D.M.)

125.

7004 (HW-64866(p.148-9)) EXPONENTIAL EXPERIMENTS WITH POISONED MODERATOR. R. C. Lloyd (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A series of measurements was completed for determining the amount of neutron absorber required to make a heterogeneous system of 1.007% enriched uranium rods safe. The fuel rods (aluminum clad) were 0.925-in. in diameter and 44-in. long. Eighty-five of the fuel rods were loaded into a stainless-steel tank, and increments of boric acid (H_3BO_3) were added to the moderator until the buckling became negative. (W.D.M.)

126.

6592 (HW-64866(p.150-1)) EXPONENTIAL MEASUREMENTS OF NATURAL URANIUM–WATER SYSTEMS. R. C. Lloyd (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A series of exponential measurements was started with 0.925-in. diameter natural uranium fuel elements in water. The purpose was to obtain further data on the critical mass of natural uranium in light water and to obtain buckling values for the interpretation of planned experiments involving a 3% enriched uranium lattice reflected by a natural uranium tamper. (W.D.M.)

127.

2064 HW-65552

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

SUMMARY LISTING OF SUBCRITICAL MEASUREMENTS OF HETEROGENEOUS WATER–URANIUM LATTICES MADE AT HANFORD. R. C. Lloyd. June 8, 1960. 53p. Contract W-31-109-eng-52. OTS.

A listing of all measurements on water–uranium heterogeneous lattices made at Hanford is presented. All lattices were water-moderated hexagonal arrays loaded with uranium enrichments up to 3.1%. (J.R.D.)

128.

6997 (HW-66215(p.30-3)) CRITICAL APPROACH AND EXPONENTIAL MEASUREMENTS WITH 2.00 PER CENT ENRICHED URANIUM RODS IN LIGHT WATER. R. C. Lloyd, E. D. Clayton, and B. L. Jones (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Critical approach measurements were completed with 2.00% enriched, 0.925 dia. rods 16 and 32 in. long; exponential measurements were made with the 32-in. rod length. The rods were incased in lucite tubes for insertion in the water reflected lattice assemblies. The critical number of rods in cylindrical geometry and critical mass values for spherical geometry were determined. (W.D.M.)

129.

6998 (HW-66215(p.34-6)) EXPONENTIAL MEASUREMENTS OF NATURAL URANIUM–WATER SYSTEMS. R. C. Lloyd (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A series of exponential measurements was completed with 0.925-in. dia. natural uranium fuel elements in light water. A plot of buckling vs. H_2O/U volume ratio is given. (W.D.M.)

130.

16303 (HW-SA-2109) NEUTRON MULTIPLICATION MEASUREMENTS OF HETEROGENEOUS 3.1 PER CENT ENRICHED URANIUM - WATER SYSTEMS. R. C. Lloyd, R. B. Smith, and E. D. Clayton (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). May 28, 1959. 16p.

A series of approach-to-critical type experiments were conducted to obtain information needed for the design of efficient systems for processing spent power reactor fuel elements. The series included experiments such that the lattice spacings spanned maximum buckling and minimum critical mass. Maximum buckling and minimum critical mass data are tabulated for the three fuel rod sizes tested. (B.O.G.)

131.

7003 (HW-64866(p.137-42)) NEUTRON MULTIPLICATION MEASUREMENTS WITH Pu–Al ALLOY RODS IN LIGHT WATER. V. I. Neeley, R. C. Lloyd and E. D. Clayton (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Neutron multiplication and exponential experiments were conducted with Pu–Al alloy rods having 5 wt. % plutonium enrichment. The rods were 24-in. long and 0.506-in. in diameter and were clad with 0.030-in. Zircaloy-2 with 0.020- and 0.125-in.-thick end caps. The average rod contained 11.01 g of plutonium. This gave a Zr/Pu of 31.92 and Al/Pu of 168.20. Experiments were conducted in the TTR reactor room. Experiments were conducted using critical approach, "back-off," and exponential measurement techniques. (W.D.M.)

132.

24545 NATURAL URANIUM AS A REACTOR FUEL. H. N. Schludi. Kerntechnik, 3: 208-10(May 1961). (In German)

Natural uranium fuel rods of equal strength made of uranium metal and UO_2 are investigated as to their maximum excess reactivity. The result is that with high moderation, a maximum reactivity gain of 0.020 is obtained with uranium metal as compared to UO_2. As regards the maximum excess reactivity, a metallic natural uranium fuel rod is equivalent to a UO_2 fuel rod of equal strength enriched to 0.748% uranium-235. (auth)

133.

8270 (RFP-201) NUCLEAR SAFETY EXPERIMENTS ON PLUTONIUM AND ENRICHED URANIUM HYDROGEN MODERATED ASSEMBLIES CONTAINING BORON. C. L. Schuske, G. H. Bidinger, and D. F. Smith (Dow Chemical Co. Rocky Flats Plant, Denver). July 7, 1960. Decl. Aug. 25, 1960. 14p. Contract AT(29-1)-1106. OTS.

Neutron multiplication measurements were made on cylindrical assemblies containing layers of plutonium metal disks, Plexiglas disks, and boron carbide impregnated Epolene-n disks. In addition to the above nuclear safety measurements, curves were drawn for a 42-in.-diameter stainless-steel tank containing an aqueous solution of $UO_2(NO_3)_2$ and poisoned with Pyrex Raschig rings. Attempts were made to calculate sphere, infinite cylinder and slab shapes from the experimental finite cylindrical assemblies. (auth)

134.

26666 (NP-10247(p.228-32)) BUCKLINGS OF ORGANIC COOLED HEAVY WATER MODERATED LATTICES. K. R. Srinivasan (India. Atomic Energy Establishment, Trombay). p.228-32 of PROCEEDINGS OF THE SYMPOSIUM ON LOW ENERGY NUCLEAR PHYSICS, 1961, BOMBAY, FEBRUARY 27–MARCH 2, 1961.

Measurements are made of the buckling of diphenyl-cooled, heavy-water-moderated, natural U fuel clusters. Seven- and nineteen-rod clusters are used. The buckling is measured as a function of the number of rods in the cluster, the rod diameter, the cell radius, and the minimum distance between the rods. The measurements are carried out in the ZERLINA assembly. (T.F.H.)

135.

10402 METHODS FOR THE MEASUREMENT OF THE PHYSICAL PARAMETERS OF THE CORE OF HETEROGENEOUS REACTORS BY MEANS OF EXPONENTIAL EXPERIMENTS. III. S. Tassan (Brookhaven National Lab., Upton, N. Y.). Energia nucleare (Milan), 7: 821-32(Dec. 1960). (In Italian)

The macroscopic reactor core parameters determined by exponential experiments (buckling, reflector saving, relaxation length, and migration area) are reviewed. The most significant experimental results, some still unpublished, are compared with the correspondent theoretical values. (auth)

136.

18651 MEASUREMENTS IN A SUBCRITICAL ASSEMBLY OF NATURAL URANIUM AND LIGHT WATER. M. A. Vigon and C. E. Granados (Junta de Energia Nuclear, Madrid). Anales real soc. espan. fis. y quim (Madrid), Ser. A, 56: 249-56(Sept.-Oct. 1960). (In Spanish)

Some lattice parameters of a natural U, light water subcritical assembly were measured. Results obtained with and without radial reflector are compared. Values of B^2 for the bare assembly and reflector savings for the reflected assembly are given. Fine distribution of thermal flux throughout a cell was measured and f, the thermal utilization factor, determined. (auth)

137.

21787 CRITICAL EXPERIMENTS ON ENRICHED URANIUM STAINLESS STEEL WATER MODERATED LATTICES. L. M. Welshans and K. M. Johnson (Martin Co., Baltimore). Trans. Am. Nuclear Soc., 4: No. 1, 104-6(June 1961).

138.

25589 (HW-67094) LATTICE PARAMETER MEASUREMENTS FOR A CONCENTRIC TUBE FUEL ELEMENT. D. E. Wood, K. R. Birney, and E. Z. Block (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Dec. 1, 1960. Contract AT(45-1)-1350. 64p.

Lattice parameters for a concentric tube fuel element were measured in the Physical Constants Test Reactor (PCTR). The measurements reported include k_∞, f, p, and ϵ for a $10\frac{1}{2}$-inch graphite lattice with both water and air in the coolant channels, and k_∞ and f for an $8\frac{3}{8}$-inch lattice, water cooled only. The value of η derived from the $10\frac{1}{2}$-inch lattice measurements is 1.30. Measured fluxes are compared to P_3 calculations using an adjusted neutron temperature. Some of the correction factors and sources of error in the measurements are discussed. (auth)

21402 LATTICE PARAMETER MEASUREMENTS FOR A CONCENTRIC TUBE FUEL ELEMENT. D. E. Wood, K. R. Birney, and E. Z. Block (General Electric Co., Richland, Wash.). Trans. Am. Nuclear Soc., 4: No. 1, 49-50 (June 1961).

139.

7085 (HW-66215(p.14-17)) LATTICE PARAMETERS FOR THE EXPERIMENTAL GAS COOLED REACTOR. J. R. Worden and P. F. Nichols (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

The final experiment of a series in the PCTR, designed to support the EGCR program, has been completed. Values of K_∞ and f were measured using standard PCTR techniques. The fuel consisted of 2.6 wt. % enriched UO_2. Results for a 21.875-in. cell are given. (W.D.M.)

140.

18653 THEORETICAL ANALYSIS AND EXPERIMENTAL RESULTS FOR C-H MODERATED ASSEMBLIES. G. B. Zorzoli (CISE, [Milan]). Energia nucleare (Milan), 8: 255-60(Apr. 1961). (In English)

A survey is made of the exponential experiments on natural uranium lattices having a hydrogenous moderator, with particular reference to organic moderators. The measured bucklings for diphenyl moderated lattices appear to be high; a comparison is made between previous measurements and the experimental values obtained at CISE on impregnated graphite moderated lattices. A theoretical analysis confirms the presence of a systematic error in the measurements on diphenyl moderated lattices. This error can be shown to result from the contributions of spurious epithermal neutrons. (auth)

1962

141.

11172 OPTIMIZATION OF SPACING OF NATURAL URANIUM FOR VARIOUS MODERATORS. [PART] I. A. Morales y Amado and A. Serment C. (Universidad Nacional de Mexico, Mexico City). Rev. mex. fis., 10: 309-37(1961). (In Spanish)

For a cylindrical natural uranium and light water subcritical system, an optimum cell and uranium rod radius was found among all the cells and rod radii studied: $k_\infty = 0.97$, $r_0 = 1$ cm, $r_1 = 1.8$ cm. The feasibility of interchanging natural uranium rods from the $U-H_2O$ subcritical system to a $U-D_2O$ critical zero power system was shown possible utilizing only optimum diameter rods. The effect of an air gap around the uranium rod, for the light water system, was studied and it was shown that for a given cell and rod radius, there was an air gap thickness that optimized the infinite multiplication factor. Some estimates on the masses of both the subcritical natural $U-H_2O$ and the critical $U-D_2O$ systems are given. Monoenergetic diffusion theory was used and all computations were done on the IBM-650 computer of the University of Mexico. (auth)

142.

33977 (BAW-1259) SPECTRAL SHIFT CONTROL REACTOR—BASIC PHYSICS PROGRAM. Quarterly Technical Report No. 7, April–June 1962. (Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.). Contract AT(30-1)-2602. 77p.

Work performed on the Spectral Shift Control Reactor Basic Physics Program is summarized. The major objective of this program is to study the nuclear properties of slightly enriched lattices moderated by D_2O-H_2O mixtures. Critical experiments were performed with 4%-enriched UO_2 fuel rods in a lattice having a non-moderator-to-moderator volume ratio of 1.2, and moderated by a 49.7% D_2O-H_2O mixture. Measured were critical mass, $d\rho/dh$, buckling and reflector savings, and cadmium ratios of U^{235} and U^{238}. Small systematic errors in the cadmium ratios were evaluated by correlating the measurements over a range of

moderator compositions from zero to 70.1% D_2O. The critical parameters of the first assembly with 2.46%-enriched UO_2 fuel rods are reported. The lattice had a non-moderator-to-moderator volume ratio of 1.0, and was moderated by light water. Physical and chemical properties of the 2.46%-enriched fuel are summarized. The objectives, conceptual design, and final design details of the small lattice experiments are given. These experiments will test the applicability of the PCTR technique to lattice measurements in epithermal systems. Also presented are improvements to the BPG code: a neutron source is now permitted in each group and a standard set of thermal cross sectional data is incorporated. Previous analytical methods were used to check the most recent critical experiment data. The results of a systematic analysis of other light water moderated systems, using methods developed for D_2O-H_2O moderated lattices, are summarized. Design calculations for the small lattice experiments and the reactivity-period relation for critical assemblies using 2.46%-enriched UO_2 fuel are given. (auth)

143.

13856 (BAW-1232) EXPONENTIAL EXPERIMENTS ON LATTICES MODERATED BY D_2O-H_2O MIXTURES. Spectral Shift Control Reactor—Basic Physics Program. L. G. Barrett and J. H. Mortenson (Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va.). Jan. 1962. Contract AT(30-1)-2602. 78p.

The results of exponential experiments performed under the Spectral Shift Control Reactor (SSCR) Basic Physics Program are summarized. The material buckling and cadmium ratio of U^{235} were measured at room temperature in rod lattices moderated by D_2O-H_2O mixtures in which the non-moderator-to-moderator volume ratio was approximately 1.0. The two types of fuel studied were 93%-enriched UO_2-ThO_2 pellets (N_{Th}/N_{25} = 15) clad in 0.308-inch OD aluminum tubes and 4%-enriched UO_2 swaged in 0.476-inch OD stainless-steel tubes. The results are shown in the following tabulation, where the measured cadmium ratio of thorium in the first lattice is 1.14 ± 0.02. Also given is design information on the Lynchburg Source Reactor (LSR), which was assembled and operated at 1000 watts to provide neutrons for these experiments. (auth)

144.

4678 (DP-640) PDP MEASUREMENTS ON NATURAL URANIUM FUEL TUBES IN D_2O. Norman P. Baumann (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). Sept. 1961. Contract AT(07-2)-1. 30p.

Buckling measurements were made in the Process Development Pile (PDP) in D_2O-moderated lattices of tubular natural uranium fuel assemblies, placed at a triangular pitch of 7 inches. The assembly types studied included four sets of single fuel tubes and six sets of double tubes. Bucklings were determined by an analysis of the critical water heights obtained when seven test assemblies were inserted into the central test region of the PDP. The effects of removing the central fuel assemblies and of replacing them by varying numbers of poison rods were also studied. Foil activation measurements of the lattice parameters f, L^2, p, and ϵ were made at the 7-inch pitch and, in two cases, at a pitch of 14 inches. The experimental results were compared with calculations. (auth)

145.

146.

26461 (CRRP-1072) LATTICE MEASUREMENTS WITH 7-ELEMENT UO_2 CLUSTERS IN ZED-2. PART I: BUCKLINGS OVER A RANGE OF SPACINGS WITH THREE COOLANTS. G. A. Beer and D. W. Hone (Atomic Energy of Canada Ltd., Chalk River, Ont.). Jan. 1962. 99p. (AECL-1505)

Buckling measurements were made in ZED-2 in a series of hexagonal lattices of 55 7-element UO_2 clusters with spacings from 18 cm to 36 cm. Three materials in the cluster coolant region were investigated: D_2O of moderator purity; helium, to simulate a voided condition; and an organic liquid, HB-40. The experimental results were compared with values calculated using the latest Chalk River lattice recipes. Buckling measurements made previously with these 7-element UO_2 clusters in ZEEP are in agreement with the measurements made in ZED-2. A short description of the ZED-2 reactor is included. (auth)

147.

9693 (CNEN-100) EXPERIMENTS AND CALCULATIONS ON A D_2O 20% ENRICHED REACTOR. M. Bernardini, A. Bracci, U. Farinelli, P. Loizzo, M. Motta, and C. Tamagnini (Italy. Comitato Nazionale per l'Energia Nucleare, Rome). Nov. 1961. 79p.

The results of several critical experiments performed on the Ispra I reactor are discussed and compared with calculations. Two-group theory is shown to yield accurate results in most cases. The relevance of the various effects, in experiment and calculation, is analyzed. (auth)

148.

22847 (HW-69300(Rev.)) A SEMI-EMPIRICAL METHOD OF ESTIMATING MATERIAL BUCKLINGS FOR SLIGHTLY ENRICHED URANIUM TUBES IN LIGHT WATER. C. L. Brown (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 5, 1962. Contract AT(45-1)-1350. 17p.

A semi-empirical method for calculating material bucklings for slightly enriched uranium tubes in light water is presented. Lattice parameters were calculated by the IDIOT code. Buckling was calculated by the one group critical equation, using an adjusted regeneration factor (eta) to obtain a correlation to measured values. Measured bucklings on tube lattices at 0.95, 1.0, 1.25, 1.44, 1.47, and 1.6% U^{235} were used in the correlation. The calculated and measured maximum bucklings at these enrichments agreed to within ±350 10^{-6} cm^{-2}. A curve of adjusted eta versus U^{235} enrichment is given. The reliability of the method is limited to the range of tube sizes and U^{235} enrichments used in the correlation. Beyond this range, the method is less reliable, but useful for lattice survey studies. (auth)

149.

1205 MATERIAL BUCKLING OF HEAVY WATER LATTICES. STATISTICAL FIT TO EXPERIMENTAL DATA. U. L. Businaro (FIAT, Sezione Energia Nucleare, Turin). Energia nucleare (Milan), 8: 460-6(July 1961). (In English)

Available experimental data on material buckling for heterogeneous, heavy water lattices were statistically analyzed with quadratic polynomial correlations. The data were divided in four groups: metal, single bar lattices; metal, bundles of bars; metal, bundles of plates; oxide, bundles of bars. For each group correlations were obtained and compared. In the case of bundles-of-bars lattices, three different definitions for the effective bundle surface were tested. For the case of bundles-of-plates lattices, the difference between experimentally and theoretically determined values were statistically analyzed and a statistically significant deviation between experimental and theoretical values was indicated. (auth)

150.

31002 CRITICAL EXPERIMENTS ON NATURAL URANIUM OXIDE, ORGANIC COOLED, HEAVY WATER MODERATED LATTICES. G. Casini, C. Foggi, and F. Toselli (Euratom CCR, Ispra, Italy). Energia nuclear, (Milan), 9: 455-61(Aug. 1962). (In English)

A series of critical experiments was carried out at Saclay, France, by the Reactor Physics Department of the C.C.R., Euratom. The purpose of the measurements was to determine the material buckling of nine different ORGEL lattices using natural uranium oxide fuel elements, moderated by heavy water and cooled by an organic liquid. The experimental technique, the results obtained, and their comparison with calculated values are described. The latter turn out to be systematically lower than the former. Some criteria are given to improve the agreement between the two sets of values. (auth)

151.

8094 (HW-50257) EXPONENTIAL MEASUREMENTS WITH ENRICHED URANIUM IN GRAPHITE LATTICES. E. D. Clayton (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). May 17, 1957. Contract W-31-109-Eng-52. 41p.

A series of buckling measurements was undertaken in wet and dry graphite lattices at spacings of $4^3/_{16}$ to $10^3/_8$ in. for U^{235}–aluminum fuel elements for U^{235} compositions of 38.33 and 16.45 g by means of critical size from multiplication measurements, exponential methods, and quasi-exponential methods. Wet lattice bucklings, at small spacings, are considerably higher than the dry lattice bucklings, because of the smaller age in the lattices which overcompensate for the increased thermal leakage and decreased k_∞. The extrapolation length and the graphite reflector savings for a rectangular prism of 4×4 ft cross sectional dimensions were determined for the lattices measured. (B.O.G.)

152.

8301 (BAW-1231) SPECTRAL SHIFT CONTROL REACTOR BASIC PHYSICS PROGRAM. CRITICAL EXPERIMENTS ON LATTICES MODERATED BY D_2O–H_2O MIXTURES. T. C. Engelder, N. L. Snidow, R. H. Clark, C. E. Barksdale, R. H. Lewis, and M. N. Baldwin (Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va.). Dec. 1961. Contract AT(30-1)-2602. 120p.

The results of critical experiments performed under the Spectral Shift Control Reactor Basic Physics Program are summarized. Nine major critical assemblies of rod lattices were studied in moderator mixtures of light and heavy water ranging from zero to 81.2 mole % D_2O. In some assemblies, the moderator was poisoned with boric acid. The non-moderator-to-moderator volume ratio in all lattices was approximately 1.0. The fuel in most lattices was 4%-enriched UO_2 swaged in stainless steel, although two experiments were performed with 93%-enriched UO_2–ThO_2 pellets in aluminum tubes. One assembly was zone-loaded radially and contained both types of fuel. The critical mass, D_2O concentration, boron concentration, buckling, thermal disadvantage factor, and cadmium ratio of U^{235} were measured in each assembly. In most assemblies, the cadmium ratio of U^{238} or Th^{232} was measured, and in five assemblies, the epithermal neutron spectrum was derived from the measurements taken of the resonance activity of detector foils. In special experiments at high D_2O concentrations, the perturbations by moderator gaps and control blades were studied, and the reflector savings versus reflector thickness was measured. The flux distribution in the zone-loaded assembly was also mapped. The measured lattice parameters are summarized. (auth)

153.

11164 STUDY OF A URANIUM ROD ASSEMBLY OF THE BR-5 REACTOR. V. I. Galkov, V. I. Ivanov, G. N. Smirenkin, and A. P. Smirnov-Averin. Atomnaya Energ., 12: 56-7(Jan. 1962). (In Russian)

In order to obtain information on the isotopic composition of irradiated fuel elements and on certain other characteristics of the Pu oxide-fueled BR-5 fast reactor, a portion of the U reflector was investigated after a 100-day long exposure to an integrated flux of about $5 \cdot 10^{21}$ n/cm^2. The center of the material examined was located 12.6 cm from the center of the reactor. The number of fission was calculated from the absolute Cs^{137} activity while the number of captures in U^{238} was derived from the specific α activity of samples taken from various locations. The neutron leakage in the reactor was high as a result of the many Na-filled channels. The concentration of Pu^{240} which was found to be about 0.1% was determined by comparing the number of spontaneous fissions in test samples and in reference samples using an ionization chamber. The experimentally obtained ratio of the average capture and fission and capture cross sections agrees well with the value of 1.93 calculated from 18 neutron spectrum group constants and with data obtained with natural U. The average capture cross section of Pu^{239} was found to be equal to 0.415 ± 0.035 barn while the capture and fission cross section ratio was 0.19 ± 0.02. The value of this ratio

decreased rapidly from the center of the core toward the periphery of the core because of the influx of neutrons into the core which were slowed down by elastic and inelastic collisions in the Ni screen. (TTT)

154.

19921 (FTD-TT-62-376) INVESTIGATION OF THE ASSEMBLY OF URANIUM RODS IN THE BR-5 REACTOR. V. I. Galkov, V. I. Ivanov, G. N. Smirenkin, and A. P. Smirnov-Averin. Translated from Atomnaya Energ., 12: No. 1, 56-7(1962). 4p.

This paper was previously abstracted from the original language and appears in NSA, Vol. 16, abstract no. 11164.

155.

12324 (NAA-SR-6608) CRITICAL LOADING EXPERIMENT FOR THE SRE WITH THORIUM-URANIUM FUEL. R. Garcia and R. L. Lewis (Atomics International, Div. of North American Aviation, Inc., Canoga Park, Calif.). Mar. 1, 1962. Contract AT(11-1)-GEN-8. 21p.

The minimum critical loading of the SRE with Th-U fuel and Na in the core at 340°F was found to be 30.6 ± 0.2 fuel clusters. The operational loading, defined as criticality with two of the four shim rods in the core, was found to be 40.6 ± 0.2 fuel clusters. An evaluation of various plotting techniques for critical experiment data is given. It was found that plotting the inverse multiplication as a function of radial buckling, using the physical radius plus reflector savings, gave the most linear approach to criticality. (auth)

156.

8310 (DP-457) PHYSICS DATA FROM A MOCKUP OF THE HWCTR LATTICE IN THE PDP. William E. Graves (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). Feb. 1960. Contract AT(07-2)-1. 84p.

Experiments were performed in the Process Development Pile (PDP) with a critical mockup of the Heavy Water Components Test Reactor (HWCTR) lattice. Reactivity measurements served to determine control rod worths, the temperature coefficient of the reactor at low temperatures, the dependence of k_{eff} on concentrations of U^{235} and poisons in the "driver" region of the reactor, and the changes in k_{eff} that occur in certain conceivable accidents in the HWCTR. Flux distribution measurements were made to determine over-all flux shapes, local flux variations around individual components, azimuthal flux variations due to asymmetrical combinations of control rods, and vertical flux variations obtainable through the use of control rods of partial length. Sufficiently detailed flux measurements were made to permit the calculation of neutron economy tables and to determine the neutron leakage from the reactor. Suitable adjustments in the input parameters made it possible to fit the experimental results with a two-group calculation. (auth)

157.

33580 (HW-74190) PHYSICS RESEARCH QUARTERLY REPORT, APRIL-JUNE 1962. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). July 16, 1962. Contract AT(45-1)-1350. 38p.

Performance tests were carried out on a circuit designed to distinguish between recoil electrons and recoil protons, due to incident photons and neutrons, respectively, in a hydrogenous scintillator. The activities produced in lutetium foils by irradiation were found to be a function of thickness. Data are presented for correction of lutetium activity measurements from thick foils to that for thin foils. Subcritical experiments with 1.82 wt % Pu-Al rods in light water were conducted. A series of criticality experiments were begun with plutonium-nitrate solutions in a 14-in. diameter stainless steel sphere fully reflected with water. Monte Carlo calculations were made of the limiting critical concentration of Pu^{239} and U^{235} in aqueous solutions. (M.C.G.)

158.

757 MEASURED NUCLEAR PARAMETERS FOR NATURAL URANIUM ROD LATTICES IN D_2O. T. J. Hurley and G. F. O'Neil (E. I. du Pont de Nemours & Co., Aiken, S. C.). Trans. Am. Nuclear Soc., 4: No. 2, 295-6 (Nov. 1961).

159.

12683 EXPERIMENTAL BUCKLINGS OF HEAVY WATER MODERATED LATTICES OF NATURAL URANIUM METAL ROD CLUSTERS. T. J. Hurley, Jr., H. R. Fike, and G. F. O'Neill (E. I. du Pont de Nemours & Co., Aiken, S. C.). Nuclear Sci. and Eng., 12: 341-7(Mar. 1962).

Studies performed in the Process Development Pile of the Savannah River Laboratory provide precise measurements of the material bucklings of a number of D_2O-moderated lattices of natural uranium metal rods over an extended range of fuel assembly sizes and lattice pitches. The 1-in. uranium rods are clad with 0.032 in. of aluminum. Fuel assembly sizes vary from single rods to clusters of 3, 7, and 19 rods (0.09 to 1.81 kg U/cm) and lattice spacings from 7.00 to 21.00 in., covering a range of moderator-to-fuel volume ratios from 10.23 to 161.53. A few lattices are studied at different D_2O purities, in loadings of different sizes, and in reflected loadings. (auth)

160.

4681 (NAA-SR-Memo-6587) CRITICALITY STUDY FOR POPR CRITICAL EXPERIMENT FUEL FABRICATION. N. Ketzlach (Atomics International, Div. of North American Aviation, Inc., Canoga Park, Calif.). July 17, 1961. Contract [AT(11-1)-Gen-8]. 21p.

Nuclear safety criteria were established for the fabrication, storage, and transportation of UO_2 fuels having an enrichment of 2.0 and 3.0% U^{235}. Twelve figures are included summarizing the results of this study. The following criteria were established for the 3.0% U^{235} enriched fuel and may be safely applied to the 2.5% U^{235} enriched

UO_2 fuel of interest. The maximum safe batch size of UO_2, independent of container size or degree of water flooding is 1.22 kgm contained U^{235}. Safe batch sizes as a function of container volume are also presented. The maximum safe geometries are 27 liters for the sphere volume, 9.75 inches for the infinite cylinder diameter, and 4.18 inches for the infinite slab thickness. The maximum safe numbers of rods, independent of container size or degree of water flooding, are 145, 110, and 90 for 0.30-inch, 0.34-inch, and 0.38 inch diameter rods, respectively. Safe criteria are also given for the spacing between batches of pellets or rods. Safe criteria were also established for the 2.0% U^{235} enriched UO_2. The maximum safe batch size, independent of container size or degree of water flooding is 1.99 kgm contained U^{235}. Safe batch sizes as a function of container volume are also presented. The maximum safe geometries are 52 liters for the sphere volume, 12.5 inches for the infinite cylinder diameter, and 5.95 inches for the infinite slab thickness. The maximum safe numbers of rods, independent of container size or degree of water flooding, are 290, 230, and 180 for 0.30-inch, 0.34-inch, and 0.38-inch diameter rods, respectively. Safe criteria are also given for the spacing between batches of pellets or rods. (auth)

161.

13392 (NAA-SR-Memo-6872) NUCLEAR SAFETY IN HANDLING U–Zr COMPACTS. N. Ketzlach (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). Nov. 16, 1961. Contract [AT(11-1)-Gen-8]. 6p.

Nuclear safety criteria are given for the handling of 1½-in. by 1½-in. compacts having a composition of 10 wt.% highly enriched U and 90 wt.% Zr. The maximum safe batch size for these compacts is 1.68 kg contained-U^{235}, independent of spacing between compacts or the degree of water flooding. For long compacts the maximum safe mass per unit length is 0.94 kg contained-U^{235}/ft of length. This, too, is independent of spacing between compacts or the degree of water flooding. Maximum safe batches containing no more than 1.68 kg U^{235} each are safe with a minimum edge-to-edge spacing of 20 in. (auth)

162.

8093 (DP-644) NUCLEAR PARAMETERS OF MASSIVE URANIUM RODS IN D_2O. Frank E. Kinard (Du Pont de Nemours (E. I.) and Co. Savannah River Lab., Aiken, S. C.). Nov. 1961. Contract AT(07-2)-1. 42p.

Measurements were made of the nuclear parameters of heavy-water-moderated lattices of uranium metal rods, 3 inches in diameter and enriched to 3.0 wt % U^{235}, at a triangular lattice pitch of 18 inches. The value of k_∞, obtained from the four-factor formula with experimental values of f, η, ϵ, and p, agreed to within 0.6% with the value obtained from the two-group diffusion kernel together with experimental values of L^2 and B_m^2. The value of k_∞ obtained from the four-factor formula using experimental values of the parameters agreed within 0.5% with that obtained by inserting calculated values, but the measured and calculated values of the individual parameters differed by as much as 3%. (auth)

163.

760 CRITICAL EXPERIMENTS WITH LOW ENRICHED DOUBLE ANNULAR SUPERHEATER FUEL ELEMENTS IN LIGHT WATER AND THEIR ANALYSIS. P. G. Klann, C. O. Dechand, E. Fein, and W. B. Wright (Combustion Engineering, Inc., Windsor, Conn.). Trans. Am. Nuclear Soc., 4: No. 2, 298-9(Nov. 1961).

164.

1207 SUBCRITICAL TWO-ZONE SYSTEMS. H. Lutz (Universität, Bern), R. Meier and W. Winkler. Neue Tech., 3: 343-51(June 1961). (In German)

The present work intends to clarify the question of whether material bucklings of D_2O–natural uranium lattices can be measured with precision in subcritical two-zone systems. The modified two-group formalism developed in this report was checked in two different experimental series. The results of the analysis of the experiments are very satisfactory and encourage similar studies under different conditions. In the last section, a special set of orthogonal functions for two-zone systems is presented and correlated with some experiments. (auth)

165.

22846 (BAW-1199) REACTIVITY MEASUREMENTS ON UNPERTURBED, SLIGHTLY ENRICHED URANIUM DIOXIDE LATTICES. Arland L. MacKinney and Russell M. Ball (Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va.). June 1960. Contract [AT(30-1)-2602]. 21p.

Uniform lattice critical experiments with 3 and 4% enriched uranium dioxide, clad with stainless steel, were performed in ordinary water. These experiments were done with fuel pins, 0.500 inches O.D. with an active fuel length of 66.5 inches. The fuel pins were fabricated by swaging the UO_2 powders to 87% theoretical density, in stainless steel tubing. The final wall thickness was 0.0282 inches. The pitch spacings which were used and the volume fractions of materials present in the core are listed. The minimum critical radius obtained for the fully reflected cores is shown as a function of the ratio of water to equivalent uranium metal. Measurements of the differential reactivity as a function of water height for each lattice yielded the common straight lines of (differential reactivity)$^{-1/4}$ as a function of water height. Numerical integration of the differential reactivity curve yields the excess reactivity as a function of core radius. (auth)

166.

7318 AN ENGINEERING CRITICAL EXPERIMENT USING THE IRL REACTOR. J. MacPhee, R. R. Powell, O. A. Schulze, and P. M. Williams (AMF Atomics, Inc., New York). Research Reactor J., 2: No. 1, 11-16(Oct. 1961).

An engineering critical experiment of the Pressure Tube Reactor (PTR) core using the facilities of the Industrial Reactor Laboratories pool-type research reactor is described and some of the results are presented. The ex-

periment was planned and performed so that both engineering and physics objectives would be realized simultaneously and costs minimized. Successful data are obtained on the following engineering and physics objectives: control rod performance; performance of fuel handling tools; compatibility with pool reactor instrumentation; general suitability of component design to a pool type installation; mass coefficients; cadmium ratios; critical mass; control rod worths; reactivity effects of void, boron, aluminum, and stainless steel; and detailed flux distributions. (N.W.R.)

167.

2398 (APAE-54(Suppl.1)) EXTENDED SM-2 CRITICAL EXPERIMENTS CE-2. W. J. McCool, R. A. Robinson, S. H. Weiss, T. M. Raby, E. W. Schrader, and L. D. Walthousen (Alco Products, Inc., Schenectady, N. Y.). June 30, 1961. Contract AT(30-3)-326. 153p.

A description and results of a second series of critical experiments performed on the SM-2 core mockup are presented. The SM-2 core mockup contains 36.4 kg of U^{235} and an estimated 67.9 g of B^{10}. The equivalent diameter and the active height are about 22 in. The metal-to-water volume ratio is 0.344. Data are presented on activation, reactivity, and stuck rod measurements. All measurements were conducted on the open 7 control rod array employing 38 stationary fuel elements. Activation measurements consisted of neutron flux measurements using uranium fission foils for relative power distribution studies, the effect of flux suppressors on reducing power peaks, blocked coolant channel measurements, and gamma dose distribution. Reactivity measurements were performed to determine the effect of flow divider, flux suppressors, and simulated high temperature and pressure operation, B^{10} loading in the SM-2 core, and core material coefficients. For the latter, the worth in cents per gram or cents per cm^3 was determined at a simulated temperature of 510°F for B^{10}, U^{235}, stainless steel, and void. Stuck rod measurements were made to obtain an indication of the criticality margin in the event one or more control rods should stick in the operating position. (auth)

168.

11170 MEASUREMENTS ON SUBCRITICAL D_2O MODERATED REACTOR ASSEMBLIES. H. Meister (Institut für Neutronenphysik und Reaktortechnik, Kernforschungszentrum, Karlsruhe, Ger.). Nukleonik, 3: 236-46(Nov. 1961). (In German)

In an exponential experiment the multiplication characteristic magnitudes of various heavy water moderated reactor assemblies were measured. The lattice studied consisted of a quadratic arrangement of cylindrical natural uranium rods (32 mm diameter) with 1 mm thick aluminum can (fuel rods of the FR-2 Research Reactor); the lattice distances d were varied in the range from 10.8 to 24.0 cm. By measurement of the macroscopic flux distribution within a tank of 136-cm diameter, the material formfactor ("material buckling") B_m^2 was determined with the help of a BF_3 counter. These measurement values were in good agreement with those obtained by Cohen on similar assemblies. By measurement of the microdistribution of the thermal flux within a lattice cell with small Mn probes, the ratio of the mean flux value in the moderator and fuel element and therefore the thermal utilization $\underline{f}$ and the effective diffusion length $\underline{L}$ were determined. The ratio found is about 2 to 3% higher than the corresponding value found by Cohen. From the magnitudes B_m^2, L^2, and the Fermi age, the multiplication factor k_∞ was calculated, and the value $\eta = 1.307 \pm 0.015$ was found by an extrapolation method given by Mummery. (tr-auth)

169.

4679 (HW-70944) NEUTRON MULTIPLICATION MEASUREMENT WITH Pu–Al ALLOY RODS IN LIGHT WATER. V. I. Neeley, R. C. Lloyd, and E. D. Clayton (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Aug. 29, 1961. Contract AT(45-1)-1350. 12p.

Neutron multiplication and exponential measurements were conducted with Pu–Al alloy fuel elements in light water moderated lattices; a hexagonal pattern was used for the lattices which were fully water reflected. The critical mass was determined for 24-inch high cylinders by neutron multiplication measurements. The extrapolation length and bucklings for the lattices were determined by equating the buckling expression from the exponential measurements to the buckling expression for the critical size as determined from the neutron multiplication measurements. These equations were then solved for the extrapolation length and buckling. The critical mass for spherical geometry was calculated from the measured buckling and extrapolation length. The minimum critical mass for the Al–5 wt % Pu alloy rods in light water was 1.5 kg plutonium. The maximum buckling was $11{,}300 \times 10^{-6}$ cm^{-2}. The effect on the critical mass of the Pu^{240} (~5%) was determined from calculations; the results indicate the critical mass (including all isotopes) would be reduced by about 8.2% in the absence of Pu^{240}, or to 1.38 kg Pu^{239}. A curve of the critical mass and buckling obtained from these experiments is shown. The various data are tabulated. (auth)

170.

15525 (NYO-9660) MEASUREMENTS OF THE MATERIAL BUCKLINGS OF LATTICES OF NATURAL URANIUM RODS IN D_2O. P. F. Palmedo, I. Kaplan, and T. J. Thompson (Massachusetts Inst. of Tech., Cambridge). Jan. 20, 1962. Contract AT(30-1)-2344. 179p. (MITNE-13)

The experimental and analytical bases of the determination of the material bucklings of uranium-D_2O lattices are presented. Techniques which were developed, particularly with the intent of measuring material bucklings in the MIT lattice facility, are described. The design considerations and experiments dealing with the spatial distribution and magnitude of the neutron source in the lattice facility are discussed. The source distribution was analyzed as it entered the subcritical assembly tank when the tank contained only D_2O and when the tank contained a lattice of uranium rods in D_2O. The detailed investigation of the over-all flux distributions in lattices included a study of the non-

separability of the macroscopic and microscopic radial distribution. A set of computer codes was developed to reduce and analyze fully the data from flux distribution measurements. The bucklings of three lattices of 1.010-inch diameter natural uranium rods in D_2O were measured. These measurements are shown to be in good agreement with measurements made in similar lattices at other laboratories. (auth)

171.

15766 (AE-66) CRITICAL AND EXPONENTIAL EXPERIMENTS ON 19-ROD CLUSTERS (R3 FUEL) IN HEAVY WATER. Rolf Persson, Carl-Erik Wikdahl, and Zenon Zadwóski (Aktiebolaget Atomenergi, Studsvik, Sweden). Mar. 1962. 37p.

Buckling measurements on clusters of 19 UO_2 rods in heavy water were performed in an exponential assembly and by means of substitution measurements in a critical facility. The material buckling was determined as a function of lattice pitch (range of $V_{mod.}/V_{fuel}$: 7 to 22), internal spacing, void, and temperature ($20 < T < 90°C$). The change of diffusion coefficients (about 6 to 8%) caused by voids was studied with single test fuel assemblies. The progressive substitution measurements were analyzed by means of a modified one-group perturbation theory in combination with an unconventional cell definition. The buckling differences between test and reference lattices were of the order of -1.0 to -3.5 m^{-2}. The results of the exponential and the critical experiments were compared with similar measurements on the same kind of fuel. This comparison showed that the results of the various experiments agree quite well, whereas theoretical predictions fail in the extreme ranges. (auth)

172.

31203 (GEAP-3882) AEC SUPERHEAT CRITICALS—A COMPARISON OF EXPERIMENT AND THEORY OF UNIFORM LATTICES. G. T. Petersen and F. G. Warzek (General Electric Co. Atomic Power Equipment Dept., San Jose, Calif.). Jan. 1962. Contract AT(04-3)-189. 107p.

A critical experiment program was undertaken to supply the basic physics information, over a wide range of water-to-fuel volume ratios, which is necessary to predict with confidence the nuclear behavior of superheat cores utilizing slightly enriched annular oxide fuel. The experimental measurements included were: critical size, $(\partial\rho/\partial H)$ versus H (ρ = reactivity, H = water height), void coefficient, temperature coefficient, flux distributions, thermal utilization, and conversion ratio. All the measurements were carried out utilizing uniformly spaced arrays of fuel (no controls or water gaps). The experimental results were compared to the prediction of engineering design methods as well as more detailed calculations. Based on these comparisons, a reliable engineering design model was developed for this lattice type. (auth)

173.

24354 MEASUREMENTS OF REACTOR PARAMETERS IN SUBCRITICAL AND CRITICAL ASSEMBLIES—LATTICES OF SLIGHTLY ENRICHED URANIUM IN WATER. Glen A. Price (Brookhaven National Lab., Upton, N. Y.). Trans. Am. Nuclear Soc., 5: No. 1, 25-6(June 1962).

174.

9344 (ANL-6378) CRITICAL EXPERIMENTS WITH THORIA–URANIA FUEL IN HEAVY WATER. W. C. Redman, S. G. Kaufmann, K. E. Plumlee, and Q. L. Baird (Argonne National Lab., Ill.). Dec. 1961. Contract W-31-109-Eng-38. 126p.

The nuclear characteristics of a variety of small reactors composed of thoria–urania fuel in heavy water were determined in a program of critical experimentation. The fuel element consisted of ceramic ThO_2–$U^{235}O_2$ pellets stacked to a height of 1.5 m within 0.787-cm-OD aluminum tubing. The pellets used most frequently were of 0.587-cm diameter and had a Th/U^{235} atom ratio of 25. Rods containing similar pellets with only half as much U^{235} were used to achieve small changes in the U^{235} content of the cores. Some cores were assembled with 0.660-cm-diameter pellets having an atom ratio of 15. All cores were located in a 2-m-diameter tank containing D_2O. Three distinct core structures were used, allowing measurements with uniformly distributed fuel rods, loading patterns compatible with the EBWR core geometry, and clustered lattic arrangements with D_2O, H_2O, and air surrounding the clustered fuel. Most of the cores assembled had some amount of radial D_2O reflector. A 0.3-m-thick bottom reflector composed of D_2O and aluminum was always present. For most of the assemblies, the control rods were fully withdrawn, criticality being achieved by adjustment of the water level. Observed critical dimensions and the results of the conventional reactivity, foil activation, and fuel substitution experiments used to determine core parameters such as p, ϵ, f, and τ, are reported. Information on void and temperature coefficients and control rod worth is included. No comparison of the experimental results with theoretical predictions is made nor are observations made on flux-trap systems of this composition included. (auth)

175.

2399 (DP-643) EXPONENTIAL VOID COEFFICIENT MEASUREMENTS FOR NATURAL URANIUM TUBES IN D_2O. Warren B. Rogers, Jr. (Du Pont de Nemours (E. I.) and Co. Savannah River Lab., Aiken, S. C.). Sept. 1961. Contract AT(07-2)-1. 23p.

Exponential experiments and theoretical calculations were performed to determine the reactivity effect of expelling D_2O coolant from D_2O-moderated lattices of natural uranium fuel assemblies composed of coaxial metal tubes. Both the exponential experiments and the calculations confirmed independent critical measurements in showing large increases (up to 4%) in K_∞ when water is expelled from the coolant channels. (auth)

176.

15527 (RFP-258) EMPIRICAL ANALYSIS OF CRITICAL MASS DATA INVOLVING ENRICHED URANIUM METAL IN WATER. C. L. Schuske and C. L. Bell (Dow Chemical Co. Rocky Flats Plant, Denver). Mar. 21, 1962. Contract AT(29-1)-1106. 6p.

An empirical equation was derived which relates the surface to volume ratio of a fissile unit (enriched uranium metal ~90% U^{235}) in lattices immersed in water with the array minimum critical mass. The critical mass data used in this study were developed at the Los Alamos Scientific Laboratory. (auth)

177.

768 CRITICAL EXPERIMENTS ON 4%-ENRICHED UO_2 LATTICES MODERATED BY D_2O–H_2O MIXTURES. N. L. Snidow, R. H. Clark, C. E. Barksdale, R. H. Lewis, and T. C. Engelder (Babcock & Wilcox Co., Lynchburg, Va.). Trans. Am. Nuclear Soc., 4: No. 2, 305(Nov. 1961).

178.

13859 HEAVY WATER EXPONENTIAL EXPERIMENTS USING ThO_2 AND UO_2. J. A. Thie. International Series of Monographs on Nuclear Energy. Division X. Reactor Design Physics. Volume 5. New York, Pergamon Press, 1961. 179p. $6.00.

A survey is given of facilities, experimental data, and theoretical interpretations of D_2O-moderated exponential experiments utilizing UO_2 or ThO_2–UO_2 mixtures as fuel. (T.F.H.)

179.

6903 (NYO-9658) HEAVY WATER LATTICE RESEARCH PROJECT ANNUAL REPORT [ENDING] SEPTEMBER 30, 1961. T. J. Thompson, Irving Kaplan, and A. E. Profio, eds. (Massachusetts Inst. of Tech., Cambridge). Contract AT(30-1)-2344. 109p.

An experimental and theoretical program on the physics of heavy water moderated, partially enriched uranium metal lattices is being conducted. A subcritical assembly which uses the M. I. T. research reactor for the neutron source has been built. Theoretical and experimental research on buckling, fast fission, resonance capture, and thermal capture has been carried out for the calibration lattice of one-inch diameter natural uranium rods. Programs in pulsed neutron source research and reactor control research have been initiated. (auth)

180.

758 COMPARISON OF EXPONENTIAL AND CRITICAL BUCKLINGS FOR NATURAL URANIUM RODS IN HEAVY WATER. E. C. Wingfield and E. J. Hennelly (E. I. du Pont de Nemours & Co., Aiken, S. C.). Trans. Am. Nuclear Soc., 4: No. 2, 296-8(Nov. 1961).

181.

12684 EXPONENTIAL MEASUREMENTS OF NATURAL URANIUM RODS IN HEAVY WATER AND COMPARISONS WITH CRITICAL EXPERIMENTS. E. C. Wingfield and E. J. Hennelly (E. I. du Pont de Nemours & Co., Aiken, S. C.). Nuclear Sci. and Eng., 12: 348-58(Mar. 1962).

Buckling measurements of 42 lattices of natural uranium 1-in. rods in heavy water were made in an exponential facility at two moderator purities. The fuel assemblies were single rods and clusters of 3, 7, and 19 rods. Lattice pitches varied from 3.00 to 21.59 in. A comparison was made between the bucklings that were measured in the exponential facility and those that were obtained from critical measurements. On the basis of a constant radial buckling for the exponential, systematic differences between the exponential and critical measurements were noted. Possible causes of these differences were discussed. Changes in buckling produced by changes in moderator purity were also given. (auth)

182.

9346 (HW-69525) MATERIAL BUCKLING MEASUREMENTS ON GRAPHITE–URANIUM SYSTEMS AT HANFORD: A SUMMARY TABULATION. D. E. Wood (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). May 1961. Contract AT(45-1)-1350. 99p.

Measurements of material bucklings for graphite–uranium systems are summarized. A comprehensive listing and guide to the original data sources is provided. Complete information on physical and nuclear properties of the lattice and the geometry of the exponential assembly is included, along with some of the auxiliary data taken. The fuel sizes vary from 0.925 to 2.5 in. in diameter for five different fuel geometries. The lattice spacings vary from $4^3/_{16}$ to 15 in. Over 300 measurements of material buckling are included. (auth)

183.

6920 BUCKLING MEASUREMENTS IN URANIUM–DIPHENYL LATTICES. G. B. Zorzoli (CISE, Segrate, Italy). Energia nucleare (Milan), 8: 780(Dec. 1961). (In English)

An experiment is described that yields a positive buckling value for a natural U-diphenyl lattice. It is shown that systematic errors, insufficient measurements, and unfavorable vessel dimensions render the experimental results incorrect. (T.F.H.)

184.

933 (AEC-tr-4692) CRITICAL EXPERIMENTS WITH THE ORGANIC MODERATOR, MONOISOPROPYLDIPHENYL. A translation of "Kriticheskie Opyty s Organicheskim Zamedlitelem, Monoizopropildifenilom." N. V. Zvonov, Yu. N. Alekseenko, V. A. Strogonov, M. N. Meshcheryakov, V. I. Buinitskaya, and B. E. Yaroslavtsev (Akademiya Nauk S.S.S.R. Ordena Lenina Institut Atomnoi Energii). 1960. 38p.

The results of critical experiments on EK-10 fuel ele-

ments with the use of the organic compound, monoisopropyldiphenyl, as a moderator are presented. The experiments were carried on a square array at spacings of 16, 19, 22, and 25 mm. The critical loadings at these spacings, and also the change in critical loading on heating the assembly to 50°C were obtained. Measurements of the distribution of the thermal neutron flux were also carried out along the radius and the height of the active zone. The experimental data were compared with calculated values. Moreover, for comparison, experimental determinations of the critical loadings for the EK-10 elements were carried out at lattice spacings of 16, 19, and 22 mm with ordinary water as the moderator. In the second part of the program, experimental determinations of the effectiveness of regulating rods of different configuration were carried out as well as determinations of the effect of replacing part of the reflector with a water–beryllium reflector on the magnitude of the critical loading and on the thermal neutron distribution. (auth)

1963

185.

24258 (IG-Report-142) A COMPARISON OF A SIMPLE INTERACTION THEORY WITH EXPERIMENT. R. T. Ackroyd, E. J. Burton, and M. A. Perks (United Kingdom Atomic Energy Authority. Industrial Group H. Q., Risley, Lancs, England). 1959. 36p.

A simple interaction theory is presented for calculating safe arrays of fissile units of arbitrary size, shape, and distribution. At every stage of the theory an endeavor was made to err on the side of safety. Mixed arrays of fast and thermal units are considered. The reflecting material and void present between units may be well defined, as for example in certain kinds of fissile store, or it can be random as in plants where personnel and portable equipment may be in close proximity to vessels. The theory is compared, for the case of well defined reflectors and fissile units of one type, with the results of three interaction experiments made; on a pair of 20-kg spheres of uranium immersed in water, on arrays of similar spheres in a concrete vault, and for 3- and 7-unit arrays of cylinders containing aqueous solutions of uranyl fluoride. The theoretical results are generally more conservative than the experimental ones, but not to an unduly pessimistic degree. It is not feasible to test a novel feature of the theory, in that it can deal with the case of random or unknown distributions of reflecting material. Since the experiments for well-defined reflectors indicated that the assumptions of the theory do err on the side of safety, the predictions for random reflectors are considered to be conservative. These pitches are somewhat greater than those for well-defined reflectors, but the increases are insufficient to cause real difficulties in the design and cost of plant. To those dealing with large numbers of interaction problems the theory may prove to be attractive on four accounts. The good agreement obtained so far with experiment suggests that the theory is reasonably accurate. Unlike Monte Carlo methods, which in principle are more accurate, the method requires very little computation; (as many as thirty cases of an interaction problem can be dealt with in five minutes on a medium size computer such as Mercury). Unlike the interaction parameter method it does not require an experiment on a simple assembly of units in order to make a prediction for a complicated array of units. The theory makes predictions using the same core and reflector constants as employed in ordinary reactor calculations for single units. (auth)

186.

42223 INVESTIGATION OF PHYSICAL PROPERTIES OF THE ACTIVE ZONE IN THE WW-PR AT CRITICALITY. G. Ya. Andrianov, V. A. Voznesenskii, A. N. Kamyshan, L. V. Komissarov, V. A. Kuz'micheva, G. L. Lunin, V. N. Semenov, and V. I. Khalizev. Vienna, International Atomic Energy Agency, 1963, Preprint SM-42/89, 40p. (CONF-133-11). (In Russian)

From International Atomic Energy Agency Symposium of Exponential and Critical Experiments with Calculations, Amsterdam, Sept. 1963.

Critical experiments in the active zone of VVER showed that homogeneous multiplying lattices are well described by the critical equation up to the material parameter $x_0^2 = 60 \times 10^{-4}$ cm^{-2}. The magnitude η, describing the deviation of critical conditions from one-group, was equal to 1.04 ± 0.08. Results of experimental determinations of K_∞ and x_0^2 for an assembly of 1.5 and 2% enriched fuel elements and for a mixed assembly were in good agreement with theoretical calculations. Variation in absorber efficiency indicates that fuel rod distribution effects reactor geometry parameters. (R.V.J.)

187.

18995 ANALYSIS OF LIGHT-WATER, HIGHLY-ENRICHED URANIUM LATTICES, WITH AND WITHOUT HAFNIUM, TESTED IN THE ALIZÉ II INSTALLATION. J. Bailly, R. Delayre, Y. Desesquelles, Y. Girad, and C. Guionnet (Commissariat a l'Énergie Atomique, Centre d'Études Nucléaires, Saclay, France). p.167-75 of "Light Water Lattices." Vienna, International Atomic Energy Agency, 1962. (In French)

The extent to which the reactivity of cores uniformly loaded with natural Hf may be calculated is evaluated. Two series of critical lattices are used; in one lattice Al is employed alone, while in the other, Zircaloy plates are used instead of those made of Al. (J.R.D.)

188.

4231 (DP-758) EXPONENTIAL MEASUREMENTS ON 19-ROD CLUSTERS OF UO_2 RODS IN D_2O. Norman P. Baumann (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). Aug. 1962. Contract AT(07-2)-1. 33p.

Buckling measurements were made in the Savannah River Laboratory Exponential facilities on D_2O-moderated, 19-rod clusters of natural uranium oxide rods. Four triangular lattice pitches from 7.00 to 12.12 inches and a single square pitch of 8.00 pinches were used. Measurements

were made at room temperature and at selected temperatures up to 218°C. Buckling changes resulting from voids inside housing tubes that surrounded the clusters were also measured. Intercomparisons of the results with critical void coefficient measurements demonstrated the validity of calculations for the anisotropic diffusion effects of the voided lattices. Foil activation measurements of the lattice parameters f, L^2, p, and ϵ were made for representative lattices. The experimental results are compared with calculations. (auth)

189.

28483 (TID-18895) MEASUREMENTS OF BUCKLINGS AND VOID EFFECTS IN D_2O MODERATED, ORGANIC OR H_2O COOLED, LATTICES OF UO_2 ROD CLUSTERS. F. D. Benton and W. E. Graves (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). June 1963. Contract AT(07-2)-1. 17p.

Buckling and void effects in UO_2 rod clusters were measured. The fuel assemblies and organic and H_2O coolants were chosen to cover an extreme range of hydrogenous coolant properties. The measurements were performed in the D_2O-moderated Process Development Pile. The fuel assemblies consisted of 19, 31, or 37 UO_2 rods. The rods were made up of sintered pellets stacked in aluminum tubes. Bucklings were determined by the substitution method, in which the fuel assemblies to be measured replaced assemblies in a one-region lattice of known nuclear properties. Two analysis techniques were used: a one-group perturbation method and a two-group flux matching method. Results of measurements are given in tabular form. (M.C.G.)

26061 MEASUREMENTS OF BUCKLINGS AND VOID EFFECTS IN D_2O MODERATED, ORGANIC- OR H_2O-COOLED LATTICES OF UO_2 ROD CLUSTERS. F. D. Benton and W. E. Graves (E. I. duPont de Nemours & Co., Aiken, S. C.). Trans. Am. Nucl. Soc., 6: 53-4(June 1963).

190.

26757 (EUR-25.e) HEAVY WATER LATTICE BUCKLING MEASUREMENTS. R. Bonalumi and G. B. Zorzoli (Centro Informazioni Studi Esperienze, Milan). 1962. 31p.

A set of buckling measurements on Aquilon II by replacement method is described. The tested configurations were made up of concentric annuli elements fuelled by natural uranium metal and having polystyrene as simulating a hydrogenous coolant. The measured bucklings were compared with the calculated ones; the agreement was fairly good. (auth)

191.

2460 (BNL-731(p.1-17)) REACTOR PHYSICS DIVISION. (Brookhaven National Lab., Upton, N. Y.).

Review papers on the correlation of theory and experiment for U–H_2O lattices were completed. The theoretical results, based strictly on available cross sections and rigorous calculations, generally yield slight overestimates of criticality. Progress is also reported in areas of power reactor concepts, reactor dynamics, Graphite Research Reactor support, and resonance absorption of neutrons. Calculations were carried out in support of the AEC program on the liquid fluidized bed reactor and were continued on the "cold moderator" concept for high-conversion-ratio, Pu-fueled, thermal reactors. With large clusters of variable enrichment, Pu-fueled rods in a cold, internally pressurized, D_2O-moderated core, these reactors have initial conversion ratios greater than unity. The first bare U-graphite assembly was brought critical during the period. Measurements were performed on the UO_2 lattice with nominal water to oxide volume ratio of 3 : 1. Bucklings were measured at increasing poisons. Experimental tech-

192.

3739 K_∞ MEASUREMENTS ON D_2O-MODERATED AND COOLED LATTICES CONTAINING 19-ROD CLUSTERS OF UO_2. W. L. Brooks, G. G. Foster, and R. D. Schamberger (United Nuclear Corp., White Plains, N. Y.). Trans. Am. Nucl. Soc., 5: 340(Nov. 1962).

193.

19586 (UNC-5051) PAWLING LATTICE TEST RIG—TEST REPORT NO. 6. THE EFFECT OF HYDROGENEOUS COOLANTS ON D_2O MODERATED LATTICES. W. L. Brooks, M. R. Fleishman, G. G. Foster, C. W. Monroe, R. D. Schamberger, H. Soodak, and R. Sullivan (United Nuclear Corp. Development Div., White Plains, N. Y.). Mar. 21, 1963. Contract AT(30-1)-3023. 48p.

A series of k_∞ and void coefficient measurements in the PLATR to investigate the effects of hydrogeneous coolants on the k_∞ value of D_2O-moderated UO_2^- and UC-fueled lattices is described. The ROCKLAND-A code was revised to provide calculated results for the experiments. A computational technique for tubular fuel elements was also established. The program was conducted as part of an evaluation of the future potential of natural U, D_2O-moderated reactors. (J.R.D.)

194.

20825 (HW-75319) NUCLEAR SAFETY PARAMETERS FOR 0.5-IN. DIAMETER, 1.8 W/O Pu–Al ALLOY RODS IN LIGHT WATER. C. L. Brown (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Dec. 29, 1962. Contract AT(45-1)-1350. 15p.

The buckling values for 0.5-in.-dia, 1.8 wt % Pu–Al alloy rods in light water, reported in HW-74190, were used to estimate the critical parameters for water-moderated and -reflected arrays of these elements. The values obtained are: minimum critical mass, 2.28 kg Pu; minimum critical volume, 445.4 liters; minimum critical slab thickness, 9.3 in.; minimum critical cylinder diameter, 17.6 in., and minimum critical mass/unit area, 41.8 lbs of alloy/ft^2. The maximum material buckling obtained from the measurements was 0.006130 cm^{-2} at a water-to-alloy volume ratio of 1.55. The minimum critical mass occurs at a water-to-alloy volume ratio of 2.0. The above values will be used as the basis for nuclear safety evaluations concerning the handling and storage of these fuel rods. (auth)

195.

40279 (TID-19028) INTERACTION IN ARRAYS—A COMPARISON OF THEORY AND EXPERIMENT. H. K. Clark (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). [1963]. Contract AT(07-2)-1. 13p. (CONF-39-42)

From American Nuclear Society 9th Annual Meeting, Salt Lake City, June 1963.

A simple method for computing the interaction in a critical grouping of units, each of which would be subcritical if isolated from the others, is reviewed. Calculations by this method are compared with experimental results for reflected and unreflected cubic arrays with various numbers of units. (D.C.W.)

26193 INTERACTION IN ARRAYS: A COMPARISON OF THEORY AND EXPERIMENT. H. K. Clark (E. I. du Pont de Nemours & Co., Aiken, S. C.). Trans. Am. Nucl. Soc., 6: 168-9(June 1963).

196.

20827 CRITICALITY OF INTERACTING ARRAYS OF FISSILE MATERIAL. I. GENERAL THEORY. D. C. Dowson (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). J. Nucl. Energy, Pt. A & B, 17: 1-6(Jan. 1963). (AHSB(S)R-28). (In English)

A criterion is described for testing whether an array of fissile units is subcritical, critical, or supercritical. A precise meaning is given to the term "degree of criticality." Formulas are derived for the multiplication of an array immersed in a nonabsorbing medium for both external and internal neutron sources. The criterion is shown to apply no matter in what detail the neutron emission is considered. (auth)

20828 CRITICALITY OF INTERACTING ARRAYS OF FISSILE MATERIAL. II. UNREFLECTED AIR-SPACED ARRAYS OF SPHERES. D. C. Dowson and F. Abbey (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). J. Nucl. Energy, Pt. A & B, 17: 7-13(Jan. 1963). (AHSB(S)R-29). (In English)

An initial application of the general method formulated in Part I to some unreflected air-spaced arrays of similar fissile spheres is described. (auth)

197.

42400 ROLE OF CRITICAL, EXPONENTIAL AND SMALL LATTICE EXPERIMENTS IN DESIGN OF SPECTRAL SHIFT CONTROL REACTOR. T. C. Engelder, L. G. Barrett, R. H. Clark, and N. L. Snidow (Babcock and Wilcox Co., Lynchburg, Va.). Vienna, International Atomic Energy Agency, 1963, Preprint SM-42/41, 55p. (CONF-133-19)

From International Atomic Energy Agency Symposium of Exponential and Critical Experiments with Calculations, Amsterdam, Sept. 1963.

The Small Lattice Experiment (SLE) consists of a 0.5-ft^3 test region containing the experimental lattice surrounded by a 2-ft diameter buffer region having similar nuclear properties. This in turn is enclosed in an 8-ft diameter driver region which is a graphite honeycomb loaded with sufficient graphite and/or U^{235} to achieve criticality. By suitable adjustment, the buffer produces an asymptotic spectrum throughout the test region, permitting standard cell measurements at comparatively high power levels. Furthermore, k_∞ of the test lattice can be derived from poisoned test region-void reactivity comparisons. SLE and critical experiment measurements of ρ_{28}, δ_{25}, Φ_m/Φ_f and k_∞ in a common epithermal lattice (4% enriched UO_2 rods, 70% D_2O in H_2O moderator) agree within ½%. A generalized theoretical method for deriving k_∞ from experimental data is given. Spectral mismatch errors are analyzed to show that accurate k_∞ measurements can be made with less than 1% of the test fuel required for an equivalent critical experiment. The application of SLE techniques to Th recycle SSCR design problems, particularly U^{233} effects, is discussed. The application to reactor design and the relative cost of direct k_∞ measurements vs k_{eff} and B^2 from critical and/or exponential experiments are evaluated. A comprehensive experimental program developed to support the design of the Spectral Shift Control Reactor is discussed. (M.P.G.)

198.

28134 EXPERIMENTAL BUCKLINGS AND VOID EFFECTS IN HEAVY WATER LATTICES OF NATURAL URANIUM OXIDE ROD CLUSTERS. W. E. Graves, H. R. Fike, and G. F. O'Neill (E. I. du Pont de Nemours and Co., Aiken, S. C.). Nucl. Sci. Eng., 16: 186-95(June 1963).

The material bucklings of twenty-five D_2O moderated lattices of natural UO_2 rod clusters were measured in the Process Development Pile (PDP). The measurements were made in one-region loadings, and should therefore be subject to little systematic error. A number of the lattices employed voided housing tubes around the fuel assemblies. Values of migration areas inferred from measurements of positive periods are also presented. An evaluation of the errors in the buckling measurements indicated that the bucklings should be accurate to about 1%. The migration areas are compared with theoretical values obtained from the Benoist theory, and the agreement is shown to be good. (auth)

199.

29727 (TID-18967) COMPARISON OF THE WORTH OF CRITICAL AND EXPONENTIAL MEASUREMENTS FOR HEAVY WATER MODERATED REACTORS. W. E. Graves and E. J. Hennelly (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). July 1963. Contract AT(07-2)-1. 50p.

Operation of critical, exponential, and pressurized-exponential heavy-water-moderated reactors is described. The applicability of these facilities in the following fields of reactor experimentation is discussed: buckling measurements in uniform lattices; anisotropic and void effects; evaluation of control systems; temperature coefficients; mixed lattices; and subcriticality studies of fuel for heavy-water-moderated reactors. (M.C.G.)

200.
17402 (AECL-1684) HIGHLIGHTS OF CHALK RIVER WORK ON THE PHYSICS OF HEAVY WATER LATTICES SINCE THE 1959 IAEA PANEL MEETING. R. E. Green, H. Maerki, C. B. Bigham, A. Okazaki, J. W. Hilborn, and E. Critoph (Atomic Energy of Canada Ltd., Chalk River, Ont.). Dec. 1962. 61p.

Results of experiments on ZED-2 (enlarged ZEEP) are presented. Analyses of these results were made by calculating lattice parameters and by comparing bucklings obtained from these parameters with experimental bucklings. Other work is reported on reaction rate measurements in a hot ZEEP reactor channel, reactor fuel changes resulting from long-time irradiation, startup and operation of NPD-2, and conversion of NRX to natural UO_2 rods on power step-up to 42 Mw. (J.R.D.)

201.
42394 (HW-77607) PLUTONIUM RECYCLE CRITICAL FACILITY LIGHT WATER DILUTION. V. W. Gustafson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). May 20, 1963. Contract AT(45-1)-1350. 4p.

A study was made of the reactivity status in the PRCF following a cell flooding event which would add H_2O to the core. A 55-inch D_2O moderator level was assumed, and a fuel loading of 17 Pu–Al Mark I type fuel elements on an 8-inch equilateral triangular lattice was calculated to have a critical moderator level at 55 inches. The effects of H_2O dilution of D_2O on the reactivity were calculated up to 50% dilution (110-inch moderator level) and are shown graphically. A continuous reactivity decrease was noted with increasing H_2O addition. (D.L.C.)

202.
9848 (KFK-108) NEUTRON SPECTRA IN HETEROGENEOUS ASSEMBLIES, A SUMMARY OF THE WORK OF THE KARLSRUHE GROUP. Wolf Haefele (Kernreaktor Bau- und Betriebs-Gesellschaft m.b.H., Karlsruhe). Apr./May 1962. 54p.

The influence of heterogeneity on neutron spectra in nuclear reactors and assemblies was investigated. The heterogeneous reactor model, a principal classification of neutron spectra, slowing down spectra in a plate-type reactor, equilibrium spectra in a plate-type reactor, reactor spectra in the thermal range, the representation of reactor spectra by simple sets of functions, and the iterated multigroup method are discussed. Graphs are included. (M.C.G.)

203.
21241 (TID-7658(p.96-8)) BORAX-V COLD CRITICAL EXPERIMENTS. J. Hagen (Argonne National Lab., Ill.).

Rod calibrations in BORAX-V were carried out as boric acid was added in discrete steps so as to achieve criticality with all rods in a bank at several different heights. The total available excess reactivity was found to be about 12% and the shutdown margin to be about 7%. These compare with calculated values of 13.5 and −14.4% respectively. A measurement was made of the temperature coefficient with boric acid present and with the aid of electric heaters. Over the range of 79 to 101°F, a value of −0.005% Δk/k was obtained. Steam voids were simulated by the use of ¼-in. OD × 0.035-in. thick walled aluminum tubing with ends sealed and inserted into water channels. Reactivity effects of removing fuel rods and entire assemblies were assessed. (M.C.G.)

204.
7549 (HW-74761(p.5.1-35)) V. PHYSICS. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

The use of lutetium as a spectral index detector was studied. Data are presented for correcting lutetium activity measurements from thick foils to that with thin foils. Subcritical experiments with 1.82 wt% Pu–Al rods in light water were conducted as part of the series of experiments which utilize plutonium containing various concentrations of the isotope Pu^{240}. The experiments furnish critical mass data which allow more accurate nuclear safety specifications to be set for the reprocessing of plutonium fuels and which can be used as check points in the calculation of light water moderated lattices containing plutonium. The experiments were conducted using critical approach and exponential measurement techniques. A brief study was conducted to examine the applicability of the Westcott scheme to the analysis of some plutonium-aluminum-H_2O assemblies and to suggest some possible prescriptions for survey calculations of reactor types not readily described by means of the Westcott cross section routines. Many of the simpler reactor surveys employ point reactor calculations. Calculations of this type can be meaningful if they are normalized to experimental values or are calibrated by more refined models. In addition to the point reactor systems, one-dimensional calculations were made for the same series of experiments using a three-group diffusion theory physics model. (auth)

205.
35119 MIGRATION AREA IN SLIGHTLY-ENRICHED-UO_2 LIGHT-WATER-MODERATED LATTICES. M. Imai, T. Kanai, and H. Kobayashi (Hitachi Ltd., Tokyo). 3p. (CONF-39-74)

From American Nuclear Society 9th Annual Meeting, Salt Lake City, June 1963.

Migration areas are calculated from a one-group critical equation, using experimental variations of buckling coefficients with lattice size. The assembly is a fully reflected, slightly enriched-UO_2-pin-fueled, light-water-moderated facility. Buckling values are tabulated for cylindrical, rectangular, and parallelepipedal core geometries. (T.F.H.)

206.
9836 (CVNA-133) PHYSICS PROPERTIES OF CLUSTERED UO_2–D_2O LATTICES AT VARIOUS LATTICE PITCHES. J. Jedruch, J. D. Cleary, R. D. Leamer, and

J. A. Roll (Westinghouse Electric Corp. Atomic Power Div., Pittsburgh). Oct. 1962. For Carolinas Virginia Nuclear Power Associates, Inc., Columbia, S. C. Contract [AT(30-1)-2289]. 344p.

An experimental and an analytic investigation of the effects of changes in the lattice of a core consisting of slightly enriched UO_2 rod clusters moderated by D_2O is described. Quantities related to thermal utilization and resonance escape probability are obtained for different water-to-U volume ratios. Thermal spectrum and conversion ratios are investigated by integral experiments. Critical masses and bucklings are given for different lattice pitches as well as spatial and total void coefficients. An optimum computational model is selected which gives good agreement with experimental values of selected parameters. (auth)

207.

9853 (ORNL-TM-433) NEUTRON MULTIPLICATION BY EXPERIMENTAL GAS COOLED REACTOR FUEL ASSEMBLIES. E. B. Johnson and R. K. Reedy, Jr. (Oak Ridge National Lab., Tenn.). Nov. 27, 1962. Contract [W-7405-eng-26]. 6p.

Experiments were conducted to determine the source neutron multiplication of EGCR fuel assemblies under conditions which might exist in transportation and storage. Twenty-eight fuel assemblies were arranged to yield maximum nuclear reactivity under conditions which might be expected for out-of-reactor environments. It was found that these 28 assemblies cannot be made critical when water moderated and reflected. (auth)

208.

31267 (NAA-SR-Memo-8522) MATERIAL BUCKLING MEASUREMENTS FOR GRAPHITE MODERATED, URANIUM CARBIDE-FUELED LATTICES. E. J. Johnson (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). May 22, 1963. Contract [AT(11-1)-Gen-8]. 29p.

Material buckling values for a series of graphite-moderated uranium carbide-fueled lattices were obtained by exponential experiment using the AE-6 thermal column as the neutron source. Three lattice pitches with center-to-center fuel spacings of 9.02, 11.015, and 12.056 in. were examined. The unit cell of the lattices was hexagonally shaped with a central control channel surrounded by 120° segments of six fuel elements located at the corners of the hexagon. The fuel consisted of 0.5-in.-diam. pins of 3 wt.% enriched uranium carbide arranged in an 18-pin cluster to mockup the element proposed for a 500 Mwe power reactor design. The fuel process tubes also contained solid sodium to mockup core coolant. The central control channel can be changed to simulate either the wet or dry control concepts. The measurements were compared with theoretical values obtained from four-group theory with fast-flux weighting and thermal pin disadvantage factors. There is considerable disagreement between the two sets of results. Critical experiments performed for the 11-in. spacing only tend to bear out the validity of the exponential measurements. (auth)

209.

18991 SOME LIGHT WATER LATTICE EXPERIMENTS IN THE UNITED KINGDOM. I. Johnstone (Atomic Energy Establishment, Winfrith, Dorchester, Eng.). p.73-97 of "Light Water Lattices." Vienna, International Atomic Energy Agency, 1962. (In English)

Critical and exponential measurements were made in a number of light water moderated lattices, fueled with highly enriched U. The results are compared with theoretical predictions based on the MUFT-SOFOCATE few-group model and shown to be in good agreement. Further work is reported on an experimental comparison between U-water and Pu-water lattices which may point to a breakdown in the methods of calculation or to a discrepancy in the nuclear data in current use for Pu. (auth)

210.

17419 (MITNE-26) HEAVY WATER LATTICE PROJECT ANNUAL REPORT. Irving Kaplan, A. E. Profio, and T. J. Thompson, eds. (Massachusetts Inst. of Tech., Cambridge). Sept. 30, 1962. Contract AT(30-1)-2344. (NYO-10208). 70p.

Research on material buckling for lattices with natural uranium involved developing configurations to provide an incoming flux appropriate for macroscopic and microscopic measurements. Analytical and experimental methods giving accurate results were devised; computer codes for deriving buckling from measured flux traverses were arranged

211.

42237 MIT EXPONENTIAL LATTICE STUDIES (LOW ENRICHMENT-HEAVY WATER LATTICES). I. Kaplan, D. D. Lanning, A. E. Profio, and T. J. Thompson (Massachusetts Inst. of Tech., Cambridge). Vienna, International Atomic Energy Agency, 1963, Preprint SM-42/31, 62p. (CONF-133-17)

From International Atomic Energy Agency Symposium of Exponential and Critical Experiments with Calculations, Amsterdam, Sept. 1963.

An experimental and theoretical research program on exponential lattices of normal and partially enriched U rods moderated by heavy water is discussed. The special technique developed to supply neutrons from the horizontal thermal column of M.I.T. Research Reactor to the exponential tank, which has a vertical axis, is described. Several volume ratios can be investigated. Measurements of buckling, resonance capture, conversion ratio, and fast fission are made, as well as intracell thermal flux distributions with spectrum-sensitive detectors. The experiments have shown where improvements in calculations are necessary, e.g., in treatment of the cell boundary in the thermal flux space-energy distribution. Most measurements are made by foil activation. A number of techniques have been developed, including a coincidence counting method for the measurement of Np-239 activity in resonance capture experiments, La^{140} fission-product counting for fast fission experiments, and thermal flux plots using foils of Au, Lu, and Eu. Results, including bucklings, are

available for 2.5 cm normal U metal and 0.65 cm 1.027% enriched metal. Preliminary and future work is outlined. (M.P.G.)

212.

4339 (TID-17270) ATR CRITICAL EXPERIMENT FACILITY DESCRIPTION AND INITIAL CRITICAL CONDITIONS. A. L. MacKinney and J. W. Poston (Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.). July 1962. For Ebasco Services, Inc., New York. Contract [AT(10-1)-1075]. 51p. (ATRCE-100)

The Advanced Test Reactor Critical Experiment (ATRCE) is described and data are presented from chemical analyses of components. Core design, loading, and criticality conditions are given. Preliminary control cylinder sensitivities and the prompt neutron decay constant at the critical conditions were measured. The decay constant was used to determine the worth of a variety of fully inserted rods. (H.D.R.)

213.

31692 PULSED NEUTRON EXPERIMENTS ON SUBCRITICAL HEAVY WATER NATURAL URANIUM LATTICES. H. Meister (Kernforschungszentrum Karlsruhe, Ger.). J. Nucl. Energy, Pt. A & B, 17: 97-114(June 1963). (In English)

Prompt neutron decay in several subcritical D_2O-moderated natural uranium lattices is investigated by the pulsed technique. Employing various detector positions inside the multiplying medium, a separation of flux harmonics is achieved by means of a Fourier transform. From the modal decay curves the prompt neutron decay constant α is found as a function of geometrical buckling B^2. In the lower buckling region, $B^2 < 25\ m^{-2}$, experimental α vs. B^2 curves agree with simple two-group calculations on the basis of lattice parameters determined in exponential experiments. Deviations for higher B^2 are shown to be produced by spectrum shifts towards lower neutron energies. In addition, the change of radial buckling $\Delta\mu^2$ caused by cadmium rods fully inserted into the lattice is determined. The results are in agreement with stationary experiments as well as two-group calculations. With single fuel rods withdrawn from the lattice, however, the corresponding radial buckling change $\Delta\mu^2$ turns out to be strongly dependent on the buckling of the axial flux distribution. (auth)

214.

18997 MULTI-REGION LATTICE STUDIES: CORRELATION OF THEORY AND EXPERIMENT. G. H. Minton (Westinghouse Electric Corp., Pittsburgh). p.285-301 of "Light Water Lattices." Vienna, International Atomic Energy Agency, 1962. (In English)

The research and development programs at Westinghouse Atomic Power Division have emphasized the utilization of machine methods of computation and the evaluation of such methods by means of interpretation of critical experiments. The most recent program of critical experiments was the Multi-region Lattice Studies carried out under the Fuel Cycle Development Program. This program involved the measurements of criticality, neutron flux distribution, and reaction rates (microscopic parameters) in lattices of light water moderated, low enriched, UO_2 fuel clad in stainless steel. Fuel rods of UO_2 enriched to 1.6%, 2.7%, and 3.7% were utilized in lattices of 2.5 : 1 and 4.5 : 1 water/U ratio. Currently, these experiments are being supplemented by measurements in the same lattices with boric acid dissolved in the moderator. The present status of a number of experimental and analytical problems is summarized and areas where further effort is needed are noted. An attempt is made to point out inter-relationships between different measurements or calculations where significant points of agreement or contradiction can be useful in interpretation of results or in planning future work. (auth)

215.

28481 (TID-18880) EXPERIMENTS WITH WATER-REFLECTED, UNDERMODERATED, ZIRCONIUM HYDRIDE CRITICAL ASSEMBLIES. L. I. Moss (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). 1963. Contract AT(11-1)-Gen-8. 38p.

A series of critical experiments was conducted with water-reflected, undermoderated zirconium hydride assemblies. The purpose of these experiments was to evaluate the consequences of water immersion of SNAP 2/10A-type reactor cores. Critical loadings were measured with several combinations of lucite rods in vacant lattice locations, beryllium inserts, internal water, neutron poison annuli at the core-reflector interface, and ammonium pentaborate in the water supply. The reactivity worth of the upper tank water as a function of height and the incremental worth of substitution of fuel rods for lucite rods was measured by the pulsed neutron method as the loading was continued past the critical point. From these measurements, an extrapolation for the excess reactivities of fully loaded, fully water-reflected assemblies was obtained. The ratio of the effective delayed neutron fraction to the effective prompt neutron generation time (ℓ^*) was measured for several unpoisoned configurations, employing both pulsed neutron and noise analysis methods. The two methods were in satisfactory agreement, giving a best value of $4.76 \times 10^2\ sec^{-1}$. The corresponding value for the beryllium-reflected reactor was measured as $1.38 \times 10^3\ sec^{-1}$. The much greater ℓ^* for the water-reflected assembly is attributable to reflector delayed neutrons. (auth)

216.

26192 CRITICAL INTERACTION POTENTIALS OF WATER-MODERATED UF_4 ARRAYS. C. E. Newlon (Oak Ridge Gaseous Diffusion Plant, Tenn.). Trans. Am. Nucl. Soc., 6: 168(June 1963).

217.

26810 THE MEASUREMENT OF THE LATTICE PARAMETERS OF A GAS COOLED REACTOR LATTICE. P. F. Nichols, J. R. Worden, F. C. Engesser, and R. E. Heineman (General Electric Co., Richland, Wash.). Nucl.

Sci. Eng., 15: 233-44 (Mar. 1963).

A series of experimental measurements was made on the Experimental Gas Cooled Reactor (EGCR) lattice in the Physical Constants Test Reactor (PCTR). The measurements provide a broad basis for normalization of reactor calculations for lattices of this type. The fuel assembly is a cluster of seven uranium oxide rods, enriched in the U^{235} isotope and clad with stainless steel. The fuel is spaced on an eight-inch square pitch in a graphite moderator. Values of the lattice parameters k_∞, f, p, and ϵ were obtained for 1.8% enrichment of the uranium oxide fuel. The values of k_∞ and f were also obtained for 2.6% enrichment fuel. The techniques of using the PCTR were extended so that supercell measurements may be made. The values of the strength of a boron carbide control rod and a stainless steel loop tube were obtained in this way. The strength of such an inhomogeneous poison in the lattice is expressed as the difference in the supercell multiplication factor k_∞ with and without the poison in the supercell. This difference is the same quantity which is obtained in the usual reactor cell calculation. The fuel temperature coefficient of k_∞ for this cluster was also measured between 50 and 500°C. The coefficient obtained is temperature dependent. The more important of the lattice parameters for the 1.8% enriched fuel are $k_\infty = 1.146 \pm 0.004$, $f = 0.809 \pm 0.005$, $p^{28} = 0.824 \pm 0.006$, $\epsilon = 1.019 \pm 0.002$, Δk (control rod –16 cell supercell) = -0.157 ± 0.012, Δk (empty loop tube –9 cell supercell) = -0.117 ± 0.011, and $(1/k_\infty)(dk_\infty/dT) = -(0.68 \pm 0.05) \times 10^{-3} T^{-1/2}(°K)^{-1}$. For the 2.6% enriched fuel, results are $k_\infty = 1.256 \pm 0.009$ and $f = 0.845 \pm 0.006$. (auth)

218.

24656 MEASUREMENT OF THE MULTIPLICATION FACTOR k_∞ OF A WATER–UO_2 REACTOR LATTICE. Gerhard Riesch (AEG-Kernenergieversuchsanlage, Grosswelzheim, Ger.). Nukleonik, 5: 104-10(Apr. 1963). (In German)

The multiplication factor for light-water lattices was determined with a zero-reactivity measurement. By mixing B_2O_3 with the moderator water, k_∞ of the lattice was made equal to one, which was determined with the help of a test reactor. The reactivity change of the reactor had to be zero if the exactly poisoned insertion was exchanged for air in a place in the reactor in which the flux gradients of all examined neutron groups were zero. This condition was produced by surrounding the measurement insertion in the reactor with an adapter. For the apparatus used a measurement accuracy of ±1% for k_∞ could be calculated. (truth)

219.

28452 (TID-7650(p.76-87)) COMPARISON OF CRITICAL EXPERIMENTS AND THEORY FOR SOME THORIUM–WATER LATTICES. D. H. Roy (Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.).

An extensive series of critical and exponential experiments was performed with uniform lattices of fuel rods composed of fully enriched UO_2 in ThO_2. These lattices were moderated by either H_2O or mixtures of H_2O and D_2O ranging up to 82 mole % heavy water. Lattice spacings ranged from those yielding a somewhat over-moderated core to those yielding an extremely tight lattice in which more than half of the fissions were epithermal. The experimental measurements included buckling, ratio of epithermal to thermal captures in Th^{232}, ratio of epithermal to thermal fissions in U^{235}, the thermal disadvantage factor, and water height reactivity worths leading to $\partial K_{eff}/\partial B^2$. (auth)

220.

34719 (RFP-325) TWO EXPERIMENTAL SUB CRITICAL ARRAYS OF $Pu(NO_3)_4$ SOLUTION. C. L. Schuske (Dow Chemical Co. Rocky Flats Div., Golden, Colo.). [July 1963]. Contract AT(29-1)-1106. 7p.

Calculations were made of the criticality of $Pu(NO_3)_4$ solutions in two subcritical arrays. An extrapolation of the inverse multiplication curve for a shielded array of 3-in.-dia. tanks of stainless steel indicated that the 5 tanks could be of infinite length and remain subcritical. Results for the 30-in.-dia. Rasching-ring-filled stainless steel tank indicated that the vessel would not be critical at any height. (M.C.G.)

221.

26282 (RFP-315) EMPIRICAL ANALYSIS OF CRITICAL BARE ARRAYS OF CYLINDERS CONTAINING ENRICHED $UO_2(NO_3)_2$. C. L. Schuske, B. B. Ernst, and H. W. King (Dow Chemical Co. Rocky Flats Plant, Denver). May 29, 1963. Contract AT(29-1)-1106. 21p..

Empirical equations were found that relate the critical heights of cylinders in arrays with the array base dimensions for various surface-to-surface separations of the individual cylinders in the array. The methods were applied to critical arrays of cylindrical vessels containing aqueous solutions of $UO_2(NO_3)_2$. These critical data were generated by the critical mass facility of the Oak Ridge National Laboratory. (auth)

222.

24620 (EACRP-U-12) REMARKS ON THE PHYSICS OF HEAVY WATER LATTICES. B. I. Spinrad and R. W. Meier (European-American Committee on Reactor Physics). Feb. 1963. 13p.

Gathering and disseminating information on D_2O lattices, integral parameters required to specify heavy water lattice performance, status of the method used for D_2O lattice calculations, data needed for the operating conditions of reactors, use of the substitution method to study lattice buckling with interpretation of the results by the one-group perturbation or the two-group two-region method, experiments to get information on burn-up in D_2O lattices, and determination of neutron spectra and cross sections relevant to burn-up studies are examined. (D.C.W.)

223.
5576 (BNL-5985) CONVERSION RATIOS IN SLIGHTLY ENRICHED URANIUM, WATER MODERATED LATTICES. S. Tassan (Brookhaven National Lab., Upton, N. Y.). [nd.]. Contract [AT(30-2)-Gen-16]. 8p.

An experiment is described in which the conversion ratios were measured using highly enriched U-Al foils as catchers. Data are included on the ratios of epi-cadmium to sub-cadmium fission rates of U^{235} in 1% enriched U light water moderated lattices, and on conversion ratios of 1% enriched U light water moderated lattices. (J.R.D.)

224.
26194 STUDIES OF CRITICAL CUBIC ARRAYS OF FISSILE MATERIALS. J. T. Thomas (Oak Ridge National Lab., Tenn.). Trans. Am. Nucl. Soc., 6: 169-70(June 1963).

225.
41729 (HW-77871(p.22-36)) EXPERIMENTAL STUDIES IN HIGH EXPOSURE Pu–Al LATTICES. T. B. Thornbury and L. C. Davenport (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Measurements were performed in the PCTR on graphite-moderated, air-cooled lattices fueled with clusters of nineteen 0.5-inch-diameter rods of Pu^{240}-enriched Pu–Al. The cells were arranged on a square pitch with lattice spacings of 6.5, 8.375, and 10.5 inches. For each lattice spacing, the mass of copper required to poison the central cell to a multiplication of unity was determined. Foil activation data were obtained at various positions throughout the lattices with gold, copper, U^{235}–Al, and Pu–Al detectors both bare and cadmium covered. The reactivity coefficients of copper, U^{235}–Al, and Pu–Al were measured in the fuel in the central cell. Relative activation rates, activation rates for a detector with a cross section varying inversely with velocity, cadmium ratios, and ratios of effective cross sections were calculated. (auth)

226.
24609 (AEEW-R-235(Pt.I)) BRITISH INDUSTRIES COLLABORATIVE EXPONENTIAL PROGRAMME. VOLUME I. INTRODUCTION; EXPONENTIAL EXPERIMENTS ON RODS AND TUBES OF NATURAL URANIUM METAL. (United Kingdom Atomic Energy Authority. Reactor Group, Atomic Energy Establishment, Windfrith, Dorset, England). Mar. 1963. 264p.

The experimental techniques used in the performance of exponential buckling measurements are described. The results are given on buckling measurements in a wide range of graphite lattices in which the fuel elements consisted of rods or tubes of natural U metal. The observed buckling values are correlated with theory. (auth)

227.
24610 (AEEW-R-235(Pt.II)) BRITISH INDUSTRIES COLLABORATIVE EXPONENTIAL EXPERIMENTS ON ENRICHED URANIUM METAL RODS AND ON CLUSTERS OF URANIUM METAL AND URANIUM OXIDE RODS. (United Kingdom Atomic Energy Authority. Reactor Group. Atomic Energy Establishment, Winfrith, Dorset, England). Mar. 1963. 239p.

Results are given of exponential buckling measurements on graphite lattices containing clusters of U metal rods, clusters of uranium oxide rods, and single U metal rods with a range of near-natural U^{235} contents. In the case of the metal fuel elements the results of the measurements are correlated with theory. (auth)

228.
24611 (AEEW-R-235(Pt.III)) BRITISH INDUSTRIES COLLABORATIVE EXPONENTIAL PROGRAMME. VOLUME III. FINE STRUCTURE EXPERIMENTS; MISCELLANEOUS EXPONENTIAL EXPERIMENTS. (United Kingdom Atomic Energy Authority. Reactor Group. Atomic Energy Establishment, Winfrith, Dorset, England). Mar. 1963. 181p.

Results are given of neutron flux fine structure measurements in graphite lattices containing wide ranges of U metal and uranium oxide fuel element numbers. In the case of the metal fuel elements the observed fine structure parameters are correlated with theory. Measurements of neutron diffusion area in graphite, and of buckling and flux distribution in assemblies containing super lattices of empty channels are included. A correlation with theory of migration area asymmetry measurements in U-graphite lattices is also given. (auth)

229.
11938 (BAW-1257) ANALYSIS OF WATER MODERATED UO_2 AND ThO_2 LATTICES. David B. Wehmeyer (Babcock and Wilcox Co., Lynchburg, Va.). May 1962. Contract AT(30-1)-2602. 49p.

A series of critical and exponential experiments was performed with lattices of fuel rods composed of low-enriched UO_2 and fully enriched UO_2 in ThO_2. These lattices were moderated by H_2O and by mixtures of D_2O and H_2O. Measurements made included buckling, ratio of epithermal to thermal capture in U^{238} and Th^{232}, ratio of epithermal to thermal fissions in U^{235}, ratio of thermal flux in the moderator to that in the fuel, and water height reactivity worth. (J.R.D.)

230.
2158 (MND-C-2502) AN EXPERIMENTAL PROGRAM ON WATER-MODERATED ARRAYS OF URANIUM AND STAINLESS STEEL. L. Welshans (Martin Co. Nuclear Div., Baltimore). May 1962. Contract [AT(30-1)-243]. 134p.

An experimental study on critical arrays of water-moderated, flat-plate, laminated fuel bundles is described. The U loading per bundle was about 54 g U^{235}; stainless steel and boron steel contents were varied to provide the desired metal-to-water ratio and the proper boron loading. The experiments were performed in support of the ANPP code development program. This program is aimed at development of an automated general prupose nuclear design code. A total of 12 cores were studied. (J.R.D.)

231.

3790 MEASUREMENTS OF NATURAL-URANIUM, HEAVY-WATER LATTICES BY THE PULSED-NEUTRON TECHNIQUE. F. R. Westfall and A. W. Waltner (North Carolina State Coll., Raleigh). Trans. Am. Nucl. Soc., 5: 386(Nov. 1962).

232.

26818 MEASUREMENTS OF NATURAL URANIUM-HEAVY WATER LATTICES BY THE PULSED NEUTRON TECHNIQUE. Fredrick Robert Westfall. Thesis, Raleigh, N. C., North Carolina State Coll., 1962. 82p.

Natural U lattices in heavy water were investigated with the pulsed-neutron-source method. The systems were very small; buckling ranged from 300 to 900 m^{-2}. In the experiments, the mean life of the decay of thermal neutrons in systems of well-defined geometry was measured as a function of buckling. From a least-squares fit of the derived expression for the decay constant to the experimental data, quantitative information was obtained regarding the diffusion and lattice parameters characteristic of the system. Three sets of measurements were run in both of 2 different-sized cylindrical tanks; heavy water, lead-mercury stacks in heavy water, and U rods in heavy water. This enabled an extension of the range of bucklings and the investigation of two lattices of particular interest: a single rod centrally located in a small test volume and a triangular array of rods on a 7-inch pitch in a larger container. By observations and analysis on this sequence of 3 media, the important parameters for a multiplying system were deduced. (Dissertation Abstr.)

233.

26060 BUCKLINGS AND DISADVANTAGE FACTORS OF SEVERAL WATER-MODERATED, 1.25%-ENRICHED URANIUM METAL SLAB LATTICES. Henry Windsor (Brookhaven National Lab., Upton, N. Y.). Trans. Am. Nucl. Soc., 6: 52-3(June 1963).

234.

36866 (GAMD-2873) EXPERIMENTS AND ANALYSIS ON BERYLLIUM OXIDE MODERATED CORES CONTAINING HYDROGEN. W. B. Wright and R. G. Bardes (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Jan. 22, 1962. Contract AT(04-3)-187. 31p.

The reactivities of a number of cores containing BeO and polyethylene, which simulated water flooding, were measured and calculated. Other determinations included core symmetries, reactivity worths of rods and rod voids, and perturbations on assembly reactivities. The experimental arrangements and measurements, the calculational methods used in the analysis, and the results of the analytical-experimental comparison are described. (D.C.W.)

1964

235.

38316 (A/CONF.28/P/669) EXPERIMENTAL AND THEORETICAL STUDIES OF URANIUM OXIDE LATTICES MODERATED BY MIXTURES OF LIGHT AND HEAVY WATER. E. Andersen, D. Babala, K. Bryhn-Ingebrigtsen, et al. (Norway. Institutt for Atomenergi, Kjeller). May 1964. 15p.

Reactor physics measurements on UO_2 lattices moderated by mixtures of light and heavy water have been performed in the NORA zero power reactor. The fuel used for these experiments has been UO_2 enriched to 3% U^{235} clad in stainless steel. A series of measurements on different H_2O/D_2O ratios and different moderator-to-fuel volume ratios has been completed. Macroscopic and microscopic neutron flux distributions have been measured using different types of foil materials resulting in sets of B_m^2 values and cell parameters for thermal and resonance neutrons. The ratio of epi-cadmium to sub-cadmium capture in U^{238} (ρ^{28}), the ratio of epi-cadmium to sub-cadmium fission in U^{235} (δ^{25}), and initial conversion ratios have been measured. The ratio of Lu-foil activity to Cu-foil activity has been measured in different positions inside the fuel and in the moderator and was compared to calculated values of the same activation ratio and used as a spectral index for the lattices studied. Supplementary to experiments on clean critical cores some measurements on two-region cores have been performed to make it possible to extend the range of measured buckling values considerably. Analysis of the experimental results using multigroup methods both for the fast and thermal neutron region is presented. The neutron energy spectrum in the fast region is calculated by a code BIGG. Resonance absorption and fission are given a correct distribution in energy and are governed by input resonance integrals. BIGG is well adapted to handle slowing down in mixtures of light and heavy water. The thermal part of the neutron spectrum is calculated by the K-7 THERMOS code. (D.L.C.)

236.

40801 TEMPERATURE OPERATION OF AN UNPRESSURIZED NATURAL URANIUM HEAVY WATER ASSEMBLY. D. G. Andrews (Univ. of Toronto) and I. F. Freibergs. 19p. (CONF-446-15)

From American Nuclear Society 10th Annual Meeting, Philadelphia, June 1964.

Buckling measurements have been made on subcritical lattices consisting of 1-inch-diameter natural-uranium rods in square lattices in heavy water, using traversing BF_3 counters. Lattice pitches of $3\frac{1}{2}$, $5\frac{1}{4}$, and 7 in. have been used, at temperatures ranging from 18 to 90°C, leading to figures for the temperature coefficient of buckling $-(dB^2/dT)$ over the range 20 to 80°C. (auth)

237.

21591 (AEEW-M-425) A STUDY OF SOME CANADIAN EXPERIMENTS USING THE WINFRITH FIVE-GROUP SCHEME. J. R. Askew (United Kingdom Atomic Energy Authority. Reactor Group. Atomic Energy Establishment, Winfrith, Dorset, England). Jan. 1964. 18p.

As part of the justification of the Winfrith five-group scheme, a series of Canadian critical experiments was analyzed. These include 19- and 7-rod clusters, the latter with air and organic coolant as well as heavy water. Agreement of calculation with experiment on fast fission and conversion ratio is very good. Discrepancies of up to 2% in reactivity are traced to errors in fine structure which are due partly to the use of diffusion theory and partly to the method used to simulate end-regions. When fine structure is corrected the deviation of calculated k_{eff} from unity is less than 0.5% in practical cases. (auth)

238.

28986 EXPERIMENTAL BUCKLINGS AND NEUTRON-DENSITY RATIOS OF D_2O-MODERATED MIXED LATTICES. F. D. Benton and W. E. Graves (E. I. du Pont de Nemours & Co., Aiken, S. C.). Trans. Am. Nucl. Soc., 7: 74-5(June 1964).

239.

36777 (DP-873) MEASUREMENTS OF BUCKLINGS AND VOID EFFECTS IN D_2O-MODERATED, ORGANIC- OR H_2O-COOLED LATTICES OF UO_2 ROD CLUSTERS. Frederick D. Benton (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). June 1964. Contract AT(07-2)-1. 24p.

Material bucklings for a variety of UO_2 rod clusters on a triangular 9.33-inch lattice pitch in D_2O were determined by substitution experiments in the Process Development Pile. Measurements were made with clusters of 19, 31, and 37 half-inch UO_2 rods in which the coolant media were varied to include D_2O, H_2O, organics, or air. The bucklings, which were determined both by a one-group perturbation method and by a two-group flux matching method, ranged from 0.80 to 5.60 m^{-2}. (auth)

240.

3116 REACTIVITY EFFECT OF HYDROGENOUS COOLANTS IN D_2O-MODERATED LATTICES. W. L. Brooks, M. Fleishman, G. G. Foster, C. W. Monroe, and R. D. Schamberger (United Nuclear Corp., White Plains, N. Y.). Trans. Am. Nucl. Soc., 6: 244(Nov. 1963).

241.

6452 BUCKLING MEASUREMENTS OF 3% ENRICHED STAINLESS-STEEL CLAD UO_2 FUEL ELEMENTS IN D_2O MODERATOR. K. Bryhn-Ingebrigtsen, J. Smit, and C. H. Millar (Nora Project, Kjeller, Norway). p.251-4 of "Heavy Water Lattices: Second Panel Report." Vienna, International Atomic Energy Agency, 1963.

Buckling measurements were carried out in the NORA reactor for a series of uniform, square, D_2O-moderated lattices with moderator-to-fuel ratios ranging from 99 : 1 to 12 : 1 for cylindrical fuel rods of the following specifications: Fuel: 3% enriched, unsintered UO_2, density = 9.41 g/cm^3, diam. = 1.128 cm. Cladding: 304 stainless steel, integral diam. = 1.128 cm, outer diam. = 1.270 cm, thickness = 0.071 cm. The buckling values obtained are compared with different theoretical estimates. (auth)

242.

16739 (NAA-SR-7541) CRITICAL EXPERIMENTS ON SLIGHTLY ENRICHED URANIUM METAL FUEL ELEMENTS IN GRAPHITE LATTICES. R. W. Campbell, R. J. Doyas, H. C. Field, C. A. Guderjahn, R. L. Guenther, D. F. Hausknecht, M. S. Mayer, and H. A. Morewitz (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). June 30, 1963. Contract AT-11-1-GEN-8. 38p.

A series of clean critical experiments was performed in the SGR critical facility utilizing 2 wt % enriched, uranium metal, hollow cylinder, fuel elements in AGOT graphite moderator. Six lattice spacings were used, varying from 6.93 to 16.0 in. on a triangular pitch. Critical loadings and fuel element worths were determined and compared to the results of 4-group diffusion theory. Calculations utilized TEMPEST, S_4, FORM, and AIM-5 programs on the IBM 7090. The calculated K_{eff} compared well with experiments over the full range of moderator-to-fuel volume ratios when using a 2200 m/sec graphite absorption cross section of 4.07 mb. The sensitivity of the calculation to variations in the graphite absorption cross section was examined and the experimental error due to inventory uncertainties was assessed. The differential worths of both the central and peripheral fuel elements were obtained and agreed in general with AIM-5 calculations. The thermal flux traverse of a unit cell was shown to agree best with a Wilkins' spectrum option of TEMPEST. Details of both the experimental and theoretical methods are given. (auth)

243.

4688 (BNL-799(p.1-25)) REACTOR PHYSICS DIVISION. J. Chernick and H. Kouts (Brookhaven National Lab., Upton, N. Y.).

Theoretical research was mainly in the fields of neutron thermalization and reactor dynamics. The analysis of the asymptotic characteristics of the uranium plate lattices was completed with excellent agreement between buckling measurements and the theory. Alternate cadmium ratio techniques of determining resonance capture in lattices were compared. A new uranium–graphite assembly was brought critical in the Minimum Reflection Critical Facility. Kinetics of the High Flux Beam Reactor were investigated. Measurements in uranium slabs in water were completed. For the Settled Bed Reactor program, uranium carbide fuel was analyzed. The characteristics of the settled bed are studied. Work is continuing on the magnetohydrodynamic systems as alternatives to the SNAP turbogenerators for generating electricity. Pulsed fission plasma reactor studies have indicated that UBr_3 or UBr_4 would be much more suitable gaseous fuels than UF_6 be-

cause of their lower disassociation temperatures. The characteristics of a thermally regenerated cell utilizing the Pb–$PbBr_2$–Br_2 system are being investigated. A study of the characteristics of a boiling oxygen chemonuclear reactor for ozone production was carried out. The feasibility study of gaseous-moderated reactors was continued and a number of cases using HF, CF_4, DF, and HF · CF_4 moderators with UF_6-fueled cores were investigated. Studies of the stability of selected condensed aromatic compounds to heat, gamma radiation, and reactor radiation are continuing. Fouling tests are evaluated for organic coolants. Neutron and reactor cross section studies are reported. (N.W.R.)

244.

32506 (DP-868) COMPARISON OF A SIMPLE THEORETICAL TREATMENT OF CRITICAL ARRAYS OF FISSIONABLE UNITS WITH EXPERIMENT. H. K. Clark (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). Jan. 1964. Contract AT(07-2)-1. 28p.

Calculations of the critical sizes of cubic arrays of interacting fissionable units are compared with critical experiments. The units were of two types: vessels containing 5 liters of an aqueous solution of highly enriched uranium and cylinders of highly enriched uranium metal. The arrays were surrounded by various thicknesses of hydrogenous reflectors. Agreement between calculation and experiment is reasonably good when consideration is given to the simplicity of the method of calculation. Curves are presented for computing critical sizes of cubic arrays of 8, 27, 64, or 125 identical units as a function of the albedo of the reflector surrounding the array and of the reactivity of an individual unit. (auth)

245.

3114 EFFECTS OF CONTROL BLADES IN LATTICES MODERATED BY D_2O/H_2O MIXTURES. R. H. Clark, E. J. DeRoche, G. T. Fairburn, J. W. Hallam, and T. C. Engelder (Babcock and Wilcox Co., Lynchburg, Va.). Trans. Am. Nucl. Soc., 6: 221-2(Nov. 1963).

246.

44969 (CEA-R-2478) MESURES GLOBALES DE RESEAUX A GRAPHITE DANS L'EMPILEMENT CRITIQUE MARIUS. (Integral Measurements of Lattice Properties in the Natural Uranium–Graphite Critical Facility MARIUS). Francois Cogne (France, Commissariat â l'Énergie Atomique. Centre d'Etudes Nucléaires, Saclay). Aug. 1964. 40p.

A systematic study of natural uranium-graphite lattices was undertaken in the critical facility MARIUS, which was built in 1959 in Marcoule. Integral measurement of lattice properties is carried out by the progressive replacement method. The report describes the experimental methods, the analysis of the experiments, and the results obtained for lattices with pitches ranging from 192 to 317 mm and fuel elements with cross sections ranging from 6 to 20 cm^2. The principles of correlation of the results are also outlined. Additional experimental results are also given, pertaining to the determination of the anisotropy, of both the axial and the radial migration areas, and of the age in graphite. (auth)

247.

33014 (A/CONF.28/P/263) LATTICE STUDIES AND CRITICAL EXPERIMENTS IN D_2O MODERATED SYSTEMS. John L. Crandall (DuPont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.), Walter L. Brooks, Irving Kaplan, Francis L. Langford, Jr., and Loren C. Schmid. 16p.

Prepared for the United Nations Third International Conference on the Peaceful Uses of Atomic Energy, 1964.

A review is presented, dealing with heavy-water-moderated assemblies, of measurements of buckling and reactivity; calculations of reactivity; detailed measurements and calculations of lattice parameters; determinations of the effects of voids, heterogeneities, and other special features on the physics parameters; and the incorporation of the measured and calculated physics parameters into the design of heavy water reactors. Examples are given. A bibliography of 103 references is included. (T.F.H.)

248.

6457 BSQ, AN IBM-704 CODE TO CALCULATE HEAVY WATER LATTICE PARAMETERS. F. E. Driggers (E. I. du Pont de Nemours & Co., Aiken, S. C.). p.551-72 of "Heavy Water Lattices: Second Panel Report." Vienna, International Atomic Energy Agency, 1963.

A code is described that is used at the Savannah River Laboratory to calculate the buckling and other lattice parameters of D_2O-moderated lattices. A comparison is made of calculated and measured bucklings for a variety of lattices. Two adjustments are made in the calculations to improve the agreement. The amount of resonance absorption is reduced by about five percent for lattices containing uranium metal and the value of η is assumed to be a linear function of the ratio of moderator volume to fuel volume and both of the coefficients in this expression are adjusted. The agreement between measured and calculated bucklings is within ± 0.25 m^{-2} for almost all of the lattices of single rods or of tubes and is within ± 0.50 m^{-2} for almost all of the lattices of rod clusters. (auth)

249.

19315 THE INTERACTIONS OF SUBCRITICAL REACTORS. V. G. Dubovskii, A. V. Kamaev, G. M. Vladykov, F. M. Kuznetsov, V. Z. Nozik, Yu. D. Palamarchuk, G. A. Popov, and V. V. Vavilov. At. Energ. (USSR), 16: 16-20 (Jan. 1964). (In Russian)

When determining the nuclear safety of systems containing fissionable materials, the interaction of all the components must be taken into consideration; however, this is often impossible to do in the case of subcritical systems. Approximations must therefore be used for estimating the safety margin of the interacting system. In order to facilitate the problem, an attempt was made to replace the given subcritical array consisting of materials possessing given nuclear properties and geometric parameters by a reactor that is equivalent to the first one with regard to the buckling, but has different nuclear properties; equations were obtained for determining the equivalent geometrical dimensions. The equations derived were experimentally verified on subcritical arrays containing cylinders and square

parallelipipeds, made of stainless steel containers with a wall thickness of 1.5 to 2 mm and radii or sides of 30 cm, filled with an aqueous solution of 90%-enriched $^{235}UO_2(NO_3)_2$. Similar systems were also built with 6-liter capacity glass cylinders with 0.5-cm thick walls. Comparison of the experimental and theoretical values indicated that the latter contain a suitable safety margin and therefore may be used for evaluating the safety of an interacting system; this was proved by recalculating the data of E. Woodcock and U. Paxton (Progr. Nucl. Energy Series Vol. 4, 213 (1961)) concerning a system of U spheres. (TTT)

250.

3117 EXPERIMENTAL BUCKLINGS AND MIGRATION AREAS IN D_2O-MODERATED LATTICES OF NATURAL-URANIUM METAL TUBES. A. E. Dunklee and W. E. Graves (E. I. du Pont de Nemours & Co., Aiken, S. C.). Trans. Am. Nucl. Soc., 6: 244-5(Nov. 1963).

251.

4740 (CONF-187-24) BUCKLING MEASUREMENTS—HEAVY NATURAL URANIUM TUBULAR FUEL ASSEMBLIES. A. E. Dunklee and W. E. Graves (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). [Nov. 1963]. Contract AT(07-2)-1. 16p.

From American Nuclear Society Meeting, New York, Nov. 1963.

One-region buckling measurements that were made on a series of D_2O-moderated lattices of heavy uranium metal tubes in the Process Development Pile at Savannah River Laboratory are presented. The purposes of these measurements are to provide normalization points for lattice bucklings and to extend the study of natural uranium–D_2O systems. The dependence of buckling on the moderator-to-fuel ratio is studied for two types of lattices. (R.E.U.)

252.

38311 (A/CONF.28/P/645) THEORETICAL AND EXPERIMENTAL INVESTIGATIONS OF SLIGHTLY ENRICHED ZIRCONIUM-HYDRIDE MODERATED LATTICES. K. Einfeld, G. Memmert, W. Ullrich, and S. Wiesner (Internationale Atomreaktorbau G.m.b.H., Bensberg, Germany). May 1964. 20p.

In connection with the development of a heterogeneous zirconium-hydride moderated reactor, several critical and sub-critical measurements on lattices have been performed. These tests were performed in a fixed aluminum lattice by varying the arrangement of rods containing UO_2 or zirconium-hydride with equivalent diameters. The time-dependent neutron flux behavior after a neutron pulse was measured. The higher harmonic effects were eliminated through a numerical evaluation method which permitted the determination of the characteristic decay constants, as eigenvalues can also be directly calculated for a given system through theoretical methods, developed and employed for this purpose. The reliability of these methods can be illustrated with an example of a UO_2–H_2O experimental core with a large reflector. Measurements were performed on various zirconium-hydride lattices up to 450°C, in order to determine the temperature dependence of the reactivity. This dependence is of special interest in the case of zirconium-hydride moderation. In this connection several theoretical expressions for the scattering kernel and for the heterogeneous spectrum methods were investigated in order to theoretically describe the experimental results as well as possible. The critical experiments were performed on a split-table assembly of the Argonne "ZPR" type and were limited a maximum temperature of 200°C. The results of the subcritical measurements were in essence verified by the critical assembly. In addition measurements with a pile-oscillator, threshold detectors and resonance detectors were made to obtain a very detailed determination of the cell-parameters. These measurements will, in turn, make possible the determination of the temperature dependence of heterogeneous zirconium-hydride lattices. (auth)

253.

4786 (BAW-1273) SPECTRAL SHIFT CONTROL REACTOR BASIC PHYSICS PROGRAM. MEASUREMENT AND ANALYSIS OF UNIFORM LATTICES OF SLIGHTLY ENRICHED UO_2 MODERATED BY D_2O–H_2O MIXTURES. T. C. Engelder, R. H. Clark, M. N. Baldwin, E. J. DeRoche, G. T. Fairburn, J. W. Hallam, D. H. Roy, N. Vutz, and T. M. Schuler (Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.). Nov. 1963. Contract AT(30-1)-2602. 150p.

Experimental results and theoretical interpretation of a series of twenty uniform lattice critical experiments in which the neutron spectrum is varied over a fairly broad range are summarized. Two types of fuel rods were studied: 4.02%-enriched UO_2 in stainless steel tubes; and 2.46%-enriched UO_2 in aluminum tubes. Lattice nonmoderator-to-moderator volume ratios ranged from 0.65 to 1.2. The moderators were mixtures of light and heavy water ranging in composition from zero to 77% D_2O, with and without boric acid. Measurements include critical size and composition, $\partial\rho/\partial h$, buckling and reflector savings, thermal disadvantage factor, and cadmium ratios of U^{235} and U^{238}. Theoretical methods used to analyze the data are given, and results are compared. (auth)

254.

3169 (BAW-1274) SPECTRAL SHIFT CONTROL REACTOR BASIC PHYSICS PROGRAM. MEASUREMENT AND ANALYSIS OF PERTURBED LATTICES OF SLIGHTLY ENRICHED UO_2 MODERATED BY D_2O–H_2O MIXTURES. T. C. Engelder, R. H. Clark, E. J. DeRoche, G. T. Fairburn, J. W. Hallam (Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va.); D. R. Hostetler, and C. B. Woodhall (Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.). Oct. 1963. Contract AT(30-1)-2602. 97p.

Critical experiments and theoretical calculations of reactivity and flux perturbations were performed. Three major critical assemblies were studied; the basic cores had 1.206-cm diameter, aluminum clad, 2.46%-enriched UO_2 fuel rods on a square lattice pitch of 1.511 cm. Mod-

erator compositions for the three major cores had 0, 50, and 72 mole % D_2O. Boric acid was added to keep the core radius constant at 61.11 cm. Data on a zone-loaded core with an inner zone of 93%-enriched aluminum clad UO_2-ThO_2 fuel rods surrounded by an outer zone of 4%-enriched stainless steel clad UO_2 fuel rods are also included. The moderator composition was 81% D_2O. Perturbers were introduced axially by removing fuel pins to create moderator gaps and by inserting perturbing blades and cruciform rods. The blades were 10 inches wide. The cruciform rods were 6 inches from tip to tip. The materials were Boral, cadmium, or aluminum. Most perturbers were fully inserted, but some measurements were made while the Boral cruciform rod was $\frac{1}{3}$ and $\frac{2}{3}$ withdrawn. Thermal flux distributions were measured with gold and dysprosium foils. Reactivity was determined by using $\partial\rho/\partial h$ techniques and by measuring the differential boron worth and the boron concentration required for criticality at the reference moderator height. Reactivity and thermal neutron flux shapes were calculated using PDQ-4, a four-group, two-dimensional neutron diffusion code for the Philco 2000 computer, and TKO, a four-group, three-dimensional code for the IBM-704 computer. (auth)

255.

4787 (BAW-1283) SPECTRAL SHIFT CONTROL REACTOR BASIC PHYSICS PROGRAM. MEASUREMENT OF k_∞ AND OTHER LATTICE PARAMETERS BY THE SMALL LATTICE EXPERIMENT TECHNIQUE. T. C. Engelder, N. L. Snidow, D. M. Roberts, G. T. Fairburn (Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va.); and D. H. Roy (Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.): Nov. 1963. Contract AT(30-1)-2602. 124p.

The Small Lattice Experiment (SLE) is a technique for measuring k_∞ and other infinite-medium lattice parameters using much less test fuel than is required for equivalent critical or exponential experiments. The theoretical basis for applying such an experiment to epithermal lattices is discussed, an analysis of spectral mismatch error is given, and the facility for the measurements is described. SLE measurements are compared to those in an equivalent critical assembly to test the validity of the technique. In an epithermal lattice of 4.02%-enriched UO_2 fuel rods having a nonmoderator-to-moderator volume ratio of 1.0 and moderated by a D_2O-H_2O mixture containing 70% D_2O, values of k_∞, $\overline{\phi}_m/\overline{\phi}_f$, δ_{25}, and ρ_{28} agree within statistical errors (0.5 to 1%). The SLE technique was also used to measure these parameters in two similar epithermal lattices containing ($U^{235}-ThO_2$) fuel rods ($N_{Th}/N_{25} = 15$). (auth)

256.

4739 (CONF-187-14) MEASUREMENTS OF BUCKLINGS AND DIFFUSION COEFFICIENT CHANGES IN D_2O-MODERATED LATTICES OF TUBULAR UO_2 ASSEMBLIES. H. R. Fike and W. E. Graves (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). [Nov. 1963]. Contract AT(07-2)-1. 10p.

From American Nuclear Society Meeting, New York, Nov. 1963.

Diffusion coefficients and bucklings are measured for UO_2 fuel tube assemblies in D_2O moderated lattices. The effects of H_2O fogs and air in the coolant channels in place of D_2O are studied. (T.F.H.)

3118 MEASUREMENTS OF BUCKLINGS AND DIFFUSION-COEFFICIENT CHANGES IN D_2O-MODERATED LATTICES OF TUBULAR UO_2 ASSEMBLIES. H. R. Fike and W. E. Graves (E. I. du Pont de Nemours & Co., Aiken, S. C.). Trans. Am. Nucl. Soc., 6: 245-6(Nov. 1963).

257.

38294 (A/CONF.28/P/25) LATTICE STUDIES AT CHALK RIVER AND THEIR INTERPRETATION. R. E. Green, C. B. Bigham, I. H. Gibson, R. G. Jarvis, M. J. Halsall, F. E. Driggers, and C. H. Miller (Atomic Energy of Canada Ltd. Chalk River Nuclear Labs., Chalk River, Ont.). May 1964. 15p.

Lattice parameters are studied in natural-uranium-fueled heavy-water-moderated pressure-tube type reactors. The ZED-2, ZEEP, and CANDU Reactors are used in this study and their results are obtained by experimental techniques and analyzed theoretically by computer codes LATREP and LATTICE ANALYSIS. The analytical methods are outlined and correlated with the experiments. (N.W.R.)

258.

3186 (KR-50) COOLANT VOID REACTIVITY MEASUREMENTS IN A D_2O-MODERATED LATTICE WITH 7 ROD UO_2 FUEL CLUSTERS. F. W. A. Habermann (Norway. Institutt for Atomenergi, Kjeller). July 1963. 69p.

A series of measurements was carried out in the zero power facility NORA with a core of 7-rod UO_2 cluster fuel elements of the Halden Boiling Heavy Water Reactor to investigate the reactivity effect of void formation in a coolant channel. The shroud of one fuel element was subdivided into sections in which the water level could be depressed by air. The reactivity effect of voids created in this way was measured for various combinations of channels. The measured effect was always negative and largest in absolute magnitude in the center of the fuel cluster. The interaction between voided channels was such as to increase the absolute magnitude of the reactivity effects. With a first order perturbation expression based on two-group theory for an anisotropic reactor, curves were fitted to the measured reactivity effects as a function of void depth. For each case two parameters were calculated representing, respectively, multiplication and leakage properties of the lattice. It is then possible to make a comparison with lattice parameter calculations. The comparison is made only for the case of 100% void in a coolant channel by calculating the parameters for this case with the Swedish burnup 5 code. (auth)

259.
21594 (BNL-6975) A REVIEW OF THE REACTOR PHYSICS DATA FOR SLIGHTLY ENRICHED URANIUM, WATER-MODERATED LATTICES. Robert L. Hellens and Glenn A. Price (Brookhaven National Lab., Upton, N. Y.). [May 1, 1963]. Contract [AT(30-2)-Gen-16]. 168p.

The methods commonly used in making buckling, critical loading, and cell parameter measurements are reviewed, and the results of such measurements are summarized. Only simple uniform and multiregion lattice experiments are considered. The experimental work done at each of the laboratories involved is summarized. The methods of interpreting the various lattice measurements are briefly discussed, with emphasis on the sources of systematic errors. (D.C.W.)

260.
35113 (TID-5540(p.529-609)) REACTOR-PHYSICS DATA FOR WATER-MODERATED LATTICES OF SLIGHTLY ENRICHED URANIUM. Robert L. Hellens and Glenn A. Price (Brookhaven National Lab., Upton, N. Y.).

A compilation of measurements for light-water lattices is presented. Measurement techniques, interpretation of results, and actual experimental results are described. Bucklings and critical dimensions are tabulated in the appendix. A bibliography of 109 references is presented. (R.E.U.)

261.
3093 (APED-4236) CONTROL AUGMENTATION CRITICAL EXPERIMENTS. T. R. Jaworowski, P. G. Aline, and H. D. Kosanke (General Electric Co. Vallecitos Atomic Lab., San Jose, Calif.). July 15, 1963. Contract [AT(04-3)-189]. 74p.

Critical experiments are reported for clumped heterogeneous light water moderated lattices using 3 wt % enriched UO_2 fuel clad with stainless steel. Reactivity worth utilizing water height and pulsed neutron techniques, temperature and void coefficients, flux and power distributions, and spectral indicator activations were measured for a wide range of configurations. These configurations were typical of boiling water reactor lattices and included assemblies with black and gray absorbing slabs and control rod cruciforms. Differential reactivity and spectrum effects were investigated for various slab, cruciform and void tube perturbations in these experiments. The resonance spectral indicator method was used in particular to study the spatial variation of the neutron spectrum and density in a typical boiling water reactor lattice. The observed spatial neutron spectrum variations show that this is a very useful technique. (auth)

262.
3104 CRITICAL SIZE AND FLUX DISTRIBUTION MEASUREMENTS IN HIGHLY ENRICHED, LIGHT WATER MODERATED LATTICES. I. Johnstone, W. H. Taylor, and S. K. Wallace (Atomic Energy Establishment, Winfrith, Eng.). J. Brit. Nucl. Energy Soc., 2: 379-90(Oct. 1963).

The experimental determinations of critical size and flux distributions in critical and sub-critical assemblies using fuel packs consisting of Al–U and stainless steel plates are described. The results show that few-group models are adequate for calculating the critical size of highly absorbing light water moderated lattices and that material buckling and thermal fine structure measurements can be made in light water moderated assemblies as small as 12 in. square and 24 in. high. (D.C.W.)

263.
42923 (MITNE-46) HEAVY WATER LATTICE PROJECT ANNUAL REPORT. Irving Kaplan, D. D. Lanning, and T. J. Thompson, eds. (Massachusetts Inst. of Tech., Cambridge). Sept. 30, 1963. Contract AT(30-1)-2344. 136p. (NYO-10212)

An experimental and theoretical program on the physics of heavy water-moderated, partially enriched lattices is reported. Experimental methods were adapted or developed for research on buckling, fast fission, resonance capture, and thermal capture. After being successfully tested on lattices of one-inch-diameter natural-uranium rods in heavy water, the methods were applied to three lattices of ¼-inch, 1.03% enriched uranium rods, moderated by heavy water. Research programs are also under way to take and correlate data from single-rod measurements, two-region lattice measurements, miniature lattice measurements, and pulsed neutron methods. In addition, a program is under way to measure the effect of control rods in the lattice assembly. A listing of topical reports, generated during the report period, is included. (auth)

264.
12847 (NAA-SR-Memo-9349) NUCLEAR SAFETY SRE STORAGE CELL (CORE III FUEL). N. Ketzlach (Atomics International, Div. of North American Aviation, Inc., Canoga Park, Calif.). Dec. 27, 1963. Contract AT-11-1-GEN-8. 4p.

It was determined that it is safe to store the SRE Core III elements (both enrichments) in an infinite array when positioned at the spacing dictated by the presently designed storage cell. This safety is independent of the degree of water flooding. However, it is dependent on the presence of the four-inch schedule 40 steel pipe positioned in each storage hole. Although the array is calculated to be subcritical ($k_\infty \sim 0.96$) even in the absence of the steel pipe, the accuracy of the method of calculation and the margin of safety are inadequate for assuming it to be safe under these conditions. (auth)

265.
9735 URANIUM-OXIDE GRAPHITE-MODERATED SUBCRITICAL EXPERIMENTS. [PART] II. Toyohiko Kikuchi and Ryushi Kubota (Fuji Electric Mfg. Co., Ltd., Japan). Fuji Jiho, 36: 706-12(Aug. 1963). (In Japanese)

Experimental results on material bucklings and extrapolation distances for various lattices and fuel arrangements

are reported. The results are compared with theoretical values calculated by simple but realistic models. By use of the experimental results on material buckling, the number of neutrons per absorption in fuel η and the effective resonance integral is reduced. (auth)

266.
9580 (WAPD-T-1632) NON-ASYMPTOTIC LATTICE PARAMETERS IN A DRY UO_2 BLANKET-MEASUREMENTS AND CALCULATIONS. J. Korsmeyer, C. Emert, P. Johnson, S. Milani, J. Mitchell, R. Serenka, and S. Weiss (Westinghouse Electric Corp. Bettis Atomic Power Lab., Pittsburgh). Sept. 1963. Contract AT(11-1)-Gen-14. 19p. (CONF-187-47)

From American Nuclear Society Meeting, New York, Nov. 1963.

The ratios of epicadmium to subcadmium capture in U^{238}, of U^{238} capture to U^{235} fission, of U^{238} fission to U^{235} fission, and of epicadmium to subcadmium fission in U^{235} were measured in a UO_2 blanket in a UO_2-Zr-H_2O seed blanket-seed slab core with a water to nonwater volume ratio of 0.142. The results were compared with calculations by a design model using a one-dimensional few group diffusion program and with calculations by a one-dimensional, multigroup P-3 transport code. (D.C.W.)

267.
38321 (A/CONF.28/P/691) NATURAL AND SLIGHTLY ENRICHED URANIUM LATTICE STUDIES OF VOIDED METAL ROD CLUSTERS IN D_2O. H. R. Lutz (Switzerland. Eidgenossisches Institut fur Reaktorforschung, Wurenlingen), R. W. Meier, and J. P. Schneeberger. May 1964. 20p.

A series of lattices consisting of voided rod cluster elements was investigated in the subcritical facility MINOR. This experimental program has formed the basis to check calculation methods for heavy water-moderated and gas-cooled pressure-tube reactors. A detailed analysis was undertaken for an element consisting of 7 uranium metal rods enriched to 0.960%. This rod cluster is supported by a graphite structure, which is surrounded by a pressure tube. An annular air gap separates the pressure from the calandria tube. This cluster presents a physical mockup of the fuel element to be used in the first charge of the Swiss experimental power station presently under construction at Lucens. In order to obtain calculation methods that are of a more general usefulness and have a wider field of application, the experiments specific to the Lucens reactor were extended by varying a number of parameters of the lattice arrangements; the most important change consisted in a replacement of the enriched rods by natural uranium rods, leaving all structural material the same. The void content per cell was decreased by removing the calandria tube and increased by removing the graphite fillers. Some measurements with 4-rod clusters were performed. For all lattices, several moderator-to-fuel volume ratios were investigated. For some or all of the lattices the following parameters were determined: the material buckling; flux disadvantage factors; the epithermal component; Cd-ratio of ^{238}U; and axial flux peaking due to Al spacer pieces. Theoretical interpretations of the experiments have been made on the basis of the two group four factor model. (auth)

268.
44972 (MIT-2344-1) STUDIES OF REACTIVITY AND RELATED PARAMETERS IN SLIGHTLY ENRICHED URANIUM, HEAVY WATER LATTICES. B. K. Malaviya, I. Kaplan, D. D. Lanning, A. E. Profio, and T. J. Thompson (Massachusetts Inst. of Tech., Cambridge). May 25, 1964. Contract AT(30-1)-2344. 311p. (MITNE-49)

The kinetic behavior of a neutron transport medium irradiated by a burst of fast neutrons was investigated on the basis of several theoretical models. Expressions were derived for the prompt-neutron decay constant of the asymptotic thermal flux in a subcritical multiplying system. These expressions relate the decay constant of a subcritical assembly to various parameters of interest. Pulsed neutron experiments were made with subcritical assemblies to measure lattice parameters. The pulsed neutron method was also applied to the measurement of absolute reactivity and the reactivity worth of control rods in far subcritical assemblies. Concurrently with the pulsed neutron studies, steady-state exponential experiments with control rods were also undertaken. Die-away experiments on pure moderator assemblies were made to measure the thermal-neutron diffusion parameters of heavy water at room temperature and the effect of thermally black rods inserted axially in a cylindrical moderator assembly. Pulsed neutron runs on unperturbed lattices were used to evaluate such lattice parameters as k_∞, L^2, k, f_m, B^2_m, etc. These values are in agreement, within experimental uncertainties, with the results of steady-state exponential experiments and of calculations based on the THERMOS code. Pulsed neutron experiments on pertubed lattices were made to find prompt-neutron lifetime and the absolute negative reactivity of the assembly. The worths of control rods were also measured. The pulsed neutron and steady-state experiments for the measurement of the reactivity effect of control rods give results that agree within the experimental uncertainties. Two-group theory, with no allowance for absorption in the fast group, is found to underestimate the worth of the rod by a few per cent. The conditions for the validity of control rod experiments in exponential assemblies were considered. Suggestions for extending the techniques developed in this work and for refining the results are also included. (auth)

269.
26880 EXPONENTIAL EXPERIMENTS WITH WATER AND NATURAL URANIUM LATTICES. W. K. Mansfield and J. M. Kim (London Univ.). p.457-67 of "Exponential and Critical Experiments. Vol. I." Vienna, International Atomic Energy Agency, 1964.

A report is given of a series of exponential experiments performed on the Queen Mary College subcritical assembly. This assembly consists of a 4½-ft aluminum core tank in which up to 4 tons of aluminum-clad natural uranium in the

form of 1.2-in. dia. bars, 43.25-in. long, are supported vertically by a series of lattice plates. Initially a 10-c, Po–Be neutron source (2.5×10^7 n/s) was used to provide the neutron flux, but later measurements were made using a SAMES, 150 kv neutron generator (5×10^9 n/s). Buckling measurements were made for moderator-fuel ratios of 1.73, 2.48, and 3.18 using indium foils, BF_3 counters and ^{238}U fission chambers as detectors. Migration areas were determined for the same ratios of moderator-to-fuel by means of the boron poisoning technique. The novel features of the results were the apparent variation of the material buckling with fuel loading and with the type of detector employed. The reasons for these variations are discussed and their general significance to exponential experiments in hydrogeneous media developed theoretically. (auth)

270.
5970 (ORNL-3499(Vol.I)(p.62-3)) CRITICAL EXPERIMENTS AND CALCULATIONS WITH ANNULAR CYLINDERS OF U(93.2) METAL. J. T. Mihalczo (Oak Ridge National Lab., Tenn.).

Critical experiments were performed on annular cylinders of enriched (93.2% U^{235}) U metal to verify the adequacy of the S_n method of solving the transport equation for this geometry. The measured reactivities agree well with the calculated multiplication constants. Measurements of prompt-neutron decay constants by the pulsed-neutron technique were included. (auth)

271.
21684 EFFECT OF SLIGHT SHIFT OF FUEL ELEMENTS ON THE REACTIVITY OF UTR-B. Katsuichi Nakamura, Hidebumi Shimizu, Hiroshi Tanaka, and Takeo Niwa (Kinki Univ., Tokyo). Nippon Genshiryoku Gakkaishi, 6: 129-34(Mar. 1964). (In Japanese)

The core-tank of UTR-B is separated in six sections. Each section has a length of 83.3 mm in east-west direction, and holds a fuel element that has a length of 76.2 mm in the same east-west direction. It may be expected that the reactivity of the reactor would be changed, if each fuel element is shifted towards the center line of the core-tank or in the opposite direction. The possible reactivity change due to the slight shift of the fuel elements was examined. It was found that if each fuel element is shifted toward the center line of the core-tank, the reactivity of the reactor increases and the critical mass decreases. If each fuel element is shifted outward in the opposite direction, the reactivity of the reactor decreases and the critical mass increases. This effect is smallest near the center line of the core-tank, and the farther the position is off the center line, the larger the effect becomes. But near the side of the core-tank, the effect comes small again. This may be affected by the thermal neutron flux distribution and the importance function. The thermal neutron flux distribution in the core-tank was measured, and the importance function was calculated. The curve of the statistical weight vs position of the core-tank was found to have the same tendency as the experimental results. (auth)

272.
6456 A THEORETICAL INTERPRETATION OF THE PRESSURE-TUBE, HEAVY-WATER, ZERO-ENERGY EXPERIMENT IN DIMPLE. D. A. Newmarch (Atomic Energy Establishment, Winfrith, Dorset, Eng.). p.445-502 of "Heavy Water Lattices: Second Panel Report." Vienna, International Atomic Energy Agency, 1963. (AEEW-R-187)

An analysis is made of the lattice physics measurements performed in DIMPLE during 1960 to 61 on pressure-tube, heavy-water cores with various coolants in the pressure tube. The experimental reactivities and reaction rates are compared with Carlson S_n and Monte Carlo calculations in order to reveal any discrepancies between experiment and basic theory. Predictions of the Winfrith five-group scheme were compared with experiment and the more sophisticated theories. It is shown that agreement between this simple method and experiment is to some extent fortuitous and results from errors canceling each other. (auth)

273.
28921 (TID-20649) PROGRESS REPORT OF NORA PROJECT, SEPTEMBER 15, 1963–MARCH 15, 1964. (Norway. Institutt for Atomenergi, Kjeller). Contract AT (30-1)-2760. 37p.

Experimental results are summarized on: characteristics of H_2O-D_2O mixtures as reactor moderators; lattice parameters and criticality characteristics of assemblies moderated with these water mixtures; thermal neutron distributions and transport in assemblies moderated with these mixtures; neutron lifetimes in subcritical reactors; neutron spectra in JEEP; measurement of the zero-power transfer function of NORA with H_2O-D_2O moderator; the analysis of the reactivity parameters of NORA by the BIGG-K-7 THERMOS code; and the analysis of thermal neutron distributions in cells by the K-7 THERMOS code. (T.F.H.)

274.
31036 (TID-20767) PROGRESS REPORT OF NORA PROJECT, SEPTEMBER 15, 1963–MARCH 15, 1964. (Norway. Institutt for Atomenergi, Kjeller). Contract AT(30-1)-2760. 42p. (NC-56)

The development and operation of NORA research reactor are summarized along with experimental and theoretical activities associated with the reactor. Information is given on lattice parameters for H_2O/D_2O moderated cores, critical water levels, buckling measurements, intracell thermal neutron density distribution, spectrum index, capture-to-fission ratios, initial conversion ratios, zero-power transfer function measurements using impulse channel method, neutron lifetime measurement by means of reactor noise analysis, analysis of the NORA experiments using BIGG and K-7 THERMOS calculations, theoretical studies of the thermal neutron density distribution inside the cell, and calculation of disadvantage factors for small cells. (N.W.R.)

275.

17284 EXPERIMENTAL DETERMINATION OF THE RATIO β^{eff}/l ON THE ZR-1 CRITICAL ASSEMBLY. L. Pál, L. Turi, and J. Vigassy (Central Research Inst. of Physics, Budapest). J. Nucl. Energy, Pt. A & B, 17: 473-80(Dec. 1963).

The parameter β^{eff}/l ($l = \tau/k$) was experimentally determined on the light-water moderated, enriched-uranium fueled, heterogeneous critical assembly ZR-1 using both the Feynman and the Rossi methods. It was found that the Rossi method can be used in the case of thermal reactors with hydrogenous moderator and enriched fuel provided the average neutron lifetime is not larger than 10^{-4} sec. The latter of the two methods proved to be faster and more reliable and even the effect of the delayed neutrons does not disturb the evaluation of the data measured. (auth)

276.

6454 EXPERIMENTAL REACTOR PHYSICS WORK ON HEAVY WATER LATTICES IN SWEDEN. R. Persson, E. Hellstrand, and E. Johansson (Aktiebolaget Atomenergi, Stockholm). p.305-42 of "Heavy Water Lattices: Second Panel Report." Vienna, International Atomic Energy Agency, 1963.

The material buckling of various lattices in the exponential (ZEBRA) and the critical (RO) facilities, resonance integrals of various fuel geometries and materials, and neutron spectra with a fast chopper are discussed. In the exponential experiments, a leakage effect was found. The leakage results in a background that introduces systematic errors when changes in the radial buckling are estimated purely by theoretical calculations. In the critical facility the progressive substitution technique was extensively used. A modified one-group perturbation theory combined with a new cell definition is found to be a powerful tool for the interpretation of substitution experiments. A summary of the work on resonance integrals is given together with a discussion of factors limiting the experimental accuracy. The neutron spectrum in the central channel of the reactor R1, where the resonance integrals were studied, was subject to detailed investigations with a fast chopper. The energy range extended from 0.02 ev up to about 10 kev. (auth)

277.

9601 (ANL-6691) CRITICAL EXPERIMENT WITH BORAX-V. Internal Superheater. K. E. Plumlee, Q. L. Baird, G. S. Stanford (Argonne National Lab., Ill.); and P. I. Amundson (Argonne National Lab., Idaho Falls, Idaho). Nov. 1963. Contract W-31-109-eng-38. 36p.

A critical experiment was performed with 12 BORAX-V superheater subassemblies in a central voidable region plus 1228 to 1525 UO_2 fuel pins (3 wt % enriched) in a peripheral region. Removing water (28% of superheater volume) at room temperature decreased reactivity by 2.2%. The midplane (two-dimensional) peak-to-average power distribution in the voided superheater was approximately 1.24, mostly attributable to flux depressions within insulated fuel boxes. Cadmium ratios are also reported. The experiment was initiated to supplement computational information which might have affected plans for loading the superheater zone into the BORAX-V reactor. No changes were indicated by the experiment. (auth)

278.

38303 (A/CONF.28/P/361) INVESTIGATION ON SYSTEM WITH ZIRCONIUM HYDRIDE MODERATOR. N. N. Ponomarev-Stepnoi, O. N. Smirnov, and R. V. Kuleva (U.S.S.R.). May 1964. 16p.

The clusters studied are a system of alternating layers of fuel and moderator. The flat-type fuel elements consist of teflon-4 mixed with 90%-enriched uranous-uranic oxide. The ZrH_n moderator blocks are shaped and sized to ensure moderator arrangement in layers of different thickness. Critical experiments and measurements on critical clusters were performed. Results with unreflected ZrH_n + ^{235}U systems were generalized and presented in the form of a B–C diagram (B is material buckling, C, concentration). The effect of fissionable-material self-shielding on the reactor critical parameters was studied. Efficiency of various reflectors (Be, BeO, ZrH_n) was analyzed. The effect on the critical parameters of various absorbers uniformly distributed in the core volume was studied. Neutron spectral characteristics in the tested clusters were measured. Measurement results for determination of the neutron spatial distribution in the reactor volume and core cell are cited. The proper method for carrying out neutron physical calculations of these quasi-homogeneous reactor systems was selected. (M.J.T.)

279.

33012 (A/CONF.28/P/264) LATTICE STUDIES IN LIGHT WATER MODERATED SYSTEMS. Glenn A. Price (Brookhaven National Lab., Upton, N. Y.) and Robert L. Hellens. 15p.

Prepared for the United Nations Third International Conference on the Peaceful Uses of Atomic Energy, 1964.

A review is presented of developments in the study of slightly enriched, light water-moderated lattices. Material buckling, fission ratios, resonance neutron capture, and thermal neutron distributions are considered. Representative results are given for 3 wt % $^{233}UO_2$–ThO_2 triangular lattice. (R.E.U.)

280.

38653 SUBCRITICAL ASSEMBLY FOR INVESTIGATION OF REACTOR PHYSICS. I. Purika, T. Roshesku, M. Sabzu, G. Fodor, V. Ieremiya, P. Dzheordzhesku, K. Karageorgiev, and M. Pavelesku. Rev. Phys., Acad. Rep. Populaire Roumaine, 6: Suppl., 423-30(1961). (In Russian)

The design of an enriched-uranium, graphite-moderated subcritical assembly is presented. Measurements of the multiplication constant as a function of lattice spacing are given. The critical mass, critical radius, and multiplication constants for infinite media are tabulated. The design of the neutron source is given, and the reactor physics experiments for which the assembly will be used are outlined. (C.E.S.)

281.
26886 DETERMINATION OF D_2O-2% ENRICHED URANIUM LATTICE PARAMETERS BY MEANS OF A CRITICAL SYSTEM. N. Raisic, S. Takac, H. Markovic, and T. Bosevski (Inst. of Nuclear Science "Boris Kidrich", Belgrade). p.509-26 of "Exponential and Critical Experiments. Vol. II." Vienna, International Atomic Energy Agency, 1964.

A series of experiments performed to determine the basic parameters of a D_2O-2% enriched-uranium lattice are described. The fuel elements were hollow cylinders, canned in aluminum and wetted by heavy water both on the inside and outside. The experiments were performed on the RB critical assembly of the Boris Kidrič Institute of Nuclear Sciences in Belgrade. The clean geometry of this system enables a rather simple theoretical interpretation of experimental results, and straightforward comparison with two-group diffusion-theory calculations. Measurements performed for ten different lattice configurations included determination of buckling, water-level reactivity coefficient, and disadvantage factors for thermal- and epithermal-neutron flux inside a reactor cell. Techniques for these measurements are described. The experimental data are then used to derive the lattice parameters defined by the four-factor formula and two-group diffusion-theory treatment of the reactor core. The results are compared with the standard calculations of the same parameters for measured lattice configurations. Deviations are found in the value of n and resonance integral for ^{235}U. An analysis is made of the usefulness and accuracy of information obtained from critical experiments for design and exploitation of a research reactor, composed of D_2O and enriched uranium. (auth)

282.
44983 MATERIAL BUCKLING AND CRITICAL MASSES OF URANIUM RODS CONTAINING 3 WT% U^{235} IN H_2O. W. B. Rogers, Jr. and F. E. Kinard (E. I. du Pont de Nemours & Co., Aiken, S. C.). Nucl. Sci. Eng., 20: 266-71 (Nov. 1964). (TID-19880)

Exponential experiments with light-water moderator were conducted to determine criticality standards for the handling of uranium metal enriched to 3 wt % in ^{235}U. These measurements, made with massive rods 2 and 3 in. in diameter, were combined with Hanford measurements with smaller rods to provide critical bucklings and masses for H_2O-moderated lattices over a range of rod diameters from less than 0.15 to more than 3 in. Subcritical buckling measurements are compared with the more conventional approach-to-critical method. (auth)

283.
36779 (HW-81659(p.4-9)) EXPERIMENTS WITH PuO_2-UO_2 FUEL ELEMENTS IN LIGHT WATER. L. C. Schmid, W. P. Stinson, R. C. Liikala, and J. R. Worden (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Approach-to-critical, exponential, and critical experiments are carried out for H_2O-moderated UO_2-PuO_2-fueled lattices having moderator/fuel volume ratios of 2.71 to 5.14. The center-to-center spacing of the fuel elements varies from 0.71 to 0.90 in. Experimental results are compared with the predictions of one-dimensional, 4-group diffusion theory. (T.F.H.)

284.
3352 MEASUREMENT OF k_∞ AND OTHER PARAMETERS IN EPITHERMAL LATTICES BY SLE TECHNIQUE. N. L. Snidow, D. M. Roberts, G. T. Fairburn, and T. C. Engelder (Babcock & Wilcox Co., Lynchburg, Va.). Trans. Am. Nucl. Soc., 6: 246-7(Nov. 1963).

285.
21599 (NAA-SR-Memo-9382) STATUS OF ASCR CRITICAL AND SUBCRITICAL EXPERIMENT CORRELATION. E. R. Specht (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). Feb. 12, 1964. Contract AT-11-1-GEN-8. 53p.

Material bucklings for UC fuel elements, enriched to 3.02% are inferred from critical experiments in which 24 UC fuel elements were surrounded by uranium metal fuel elements. The UC fuel elements were placed on triangular pitches of 9, 11, and 12 in. and the number of uranium metal drivers changed to obtain criticality for each case. Four-group theory was employed to infer the material bucklings of each critical configuration by modifying the thermal absorption cross section of the UC region until the calculated value of k_{eff} was unity. Comparisons of theoretical calculations and "measured" material bucklings (B_m^2) indicate that multigroup theory (15 groups) does much better than few-group theory (2 to 4 groups). Because of the inherent simplicity of few-group theory (as compared to multigroup theory), an extensive comparison was made between various few-group theoretical calculations. These calculations determined the effect on the curve of material buckling versus lattice pitch when all flux shapes were calculated in each of the four energy groups (with diffusion theory) and with transport theory. Also, various treatments were employed to obtain cross sections for the thermal energy group. These included calculations with and without flux depressions across each fuel pin, with the fuel pins homogenized in various ways (over the process tube volume, over the rubber band surface of the pin cluster, and into several bands of UC), and a thermal multigroup calculation (using THERMOS). These few group calculations caused significant variations in the magnitude of the material buckling but did not produce the same shape for the curve of B_m^2 versus pitch as was obtained from the measurements. Measurements on subcritical assemblies of B_m^2 for ASCR lattices were found to vary considerably from the values of B_m^2 that were inferred from the critical assemblies. When the subcritical material bucklings are corrected by using calculated values of diffusion anistropy, most of the difference between the critical and subcritical values of B_m^2 disappears. (auth)

286.
40784 (NAA-SR-9772) MATERIAL BUCKLINGS OF CRITICAL AND SUBCRITICAL URANIUM CARBIDE FUELED GRAPHITE ASSEMBLIES. PART II. THEORETICAL INTERPRETATION. E. R. Specht (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). Aug. 1, 1964. Contract AT(11-1)-Gen-8. 47p.

Material bucklings for uranium carbide fuel elements, enriched to 3.02 wt % of uranium-235, were obtained from critical and subcritical assemblies. The fuel elements were placed in a graphite assembly on triangular lattice pitches of 9, 11, and 12 inches. Four-group theory was used to infer material bucklings from 12 "two-region" critical mass measurements. Material bucklings were obtained from 22 subcritical assemblies consisting of 31, 23, and 19 cells set at pitches of 9, 11, and 12 inches. The large differences observed (up to 3.4 m^{-2}) between the critical and uncorrected subcritical values of B_m^2 are shown to be due to diffusion anisotropy and the use of calculated rather than measured radial bucklings in the subcritical experiments. Multigroup (15 energy group) cell calculations give material bucklings that agreed very well with those inferred from the critical assemblies. (auth)

287.
11397 (HW-79054(p.43-6)) SUBCRITICAL MEASUREMENTS WITH 2.0 WT% Pu–Al RODS IN LIGHT WATER. W. P. Stinson, J. A. Marks, and L. C. Schmid (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Experiments were conducted with 2.0 wt % Pu–Al rods in light water using critical approaches and exponential measurements. The critical Pu masses, critical rod numbers, critical bucklings, and extrapolation lengths obtained are given. The critical mass and buckling are plotted as functions of the H : Pu ratio. (D.C.W.)

288.
21596 (EUR-597.e) CALCULATION OF D_2O-MODERATED LATTICES BY BASIC METHODS. B. Sturm (European Atomic Energy Community. Nuclear Research Center, Ispra, Italy). 1964. 32p. (CONF-469-2)

From the Conference on Physics of Reactors, Milan, Nov.-Dec. 1963.

For the calculation of D_2O-moderated, organic-cooled natural uranium lattices, a calculational procedure is proposed based on the most refined reactor codes available at Ispra. For the calculation of fast group constants, the GAM-I multigroup program is used. Weighting factors to take into account the heterogeneity of the lattice cell are calculated by the DSN program for macro- and microcells. For the calculation of thermal cross sections the THERMOS-code is used, and the disadvantage factors for the different materials in the cell are obtained by a one-group calculation using the DSN program. On the basis of this procedure, an analysis of critical experiments for 7-UO_2-rod, organic-cooled fuel elements was performed. Reasonable agreement was found for material buckling as well as for the measured thermal disadvantage factors, initial conversion ratio, and fast fission ratio. (auth)

289.
38314 (A/CONF.28/P/650) LIGHT-WATER-ENRICHED URANIUM LATTICE EXPERIMENTS. F. Szabó (Magyar Tudományos Akadémia Központi Fizikai Kutató Intézete, Budapest). May 1964. 18p.

Light water-moderated 10%-enriched uranium-fueled lattices were investigated with different $H/^{235}U$ ratios. One of the aims of these experiments was to test the results of calculation methods used. For this purpose the measurements were carried out in a wide interval of the $H/^{235}U$ ratio in order to get significant effects. Fuel elements of the VVR-S-type research reactor were used and in most cases triangular lattice patterns applied. Critical mass values were determined over the range 190 to 775 of the $H/^{235}U$ ratio. This range involves nearly equal lengths of under- and over-moderated regions. Temperature dependence of reactivity was determined in the interval 10 to 40°C, in over-moderated, under-moderated, and in a nearly optimum lattice. Temperatures of fuel and moderator were the same. In this temperature interval both positive and negative values of temperature coefficients were found in all the lattices. Spatial dependence of the reactivity effect of vertical, cylindrical voids—the lengths of which were identical with that of the fuel elements—was determined in four cores with different lattice pitches. The values of the void-to-cell-volume ratios were kept nearly the same. Positive void effect was reached in a strongly overmoderated lattice. Thermal neutron distributions were determined in lattices containing central water traps of different diameters. These measurements were carried out both in a strongly under-moderated and in a nearly optimum lattice. The value of β_{eff}/l was measured using both the Feynmann and the Rossi method in a nearly optimum lattice. The two methods yielded nearly the same results. The value of β_{eff}/l was determined in an over- and an under-moderated lattice by the Rossi method only, which proved to be faster and more accurate. The results of a simple calculation method were compared with the experimental data. Rather good agreement was found for the two-group model when taking into account fission under slowing down and the intracell dependence of the thermal spectrum in the calculation of the thermal utilization factor. (auth)

290.
38335 MEASUREMENTS OF NEUTRON TEMPERATURE IN SYSTEM ZR-2. Geza Szerphalmi, László Turi, and József Vigassy (MTA Kozponti Fizikai Kutatointezet, Budapest). Energia Atomtech., 17: 146-52(Mar. 1964). (In Hungarian)

After reviewing the main methods used for the determination of neutron spectra of reactors, neutron spectrum measurements carried out with the critical assembly (zero-power reactor) ZR-2 are discussed. The fuel elements of the system consisted of 10% enriched UO_2 in a Mg system, clad with Al; natural water served as modera-

tor and reflector. Three different lattice arrangements were used for the measurements which included the determination of the reactivity change caused by Cd, B, Dy and Ag specimens introduced into the assembly as aqueous solutions of $Cd(NO_3)_2$, H_3BO_3, $Dy(NO_3)_2$ and $AgNO_3$ in Plexiglas containers. The reactivity change due to the absorbent was calculated from the measured change in the asymptotic period of the system brought to the supercritical state; the concentration of the solution was so chosen that reactivity change values of about 4×10^{-4} were obtained. The results differed from those of similar experiments carried out with the VVR-S type reactor having a similar spectrum and burnout; this is probably due to differences in the experimental technique. (TTT)

291.
2565 (ORNL-TM-719) CRITICAL THREE-DIMENSIONAL ARRAYS OF NEUTRON-INTERACTING UNITS. J. T. Thomas (Oak Ridge National Lab., Tenn.). Oct. 1, 1963. Contract [W-7405-eng-26]. 13p.

As many as 125 five-liter units of concentrated aqueous uranyl nitrate solution were assembled in critical arrays. The solution, at a concentration of 415 g of uranium per liter, and having a specific gravity of 1.555, was contained in right circular cylinders of methacrylate plastic having a 0.64-cm-thick wall. The U^{235} content of the uranium was 92.6 wt %. The dependence of the number of units required for criticality as a function of the spacing was determined. The critical number, N, as a function of spacing, within the range of these experiments, has been determined to be $N = N_0(\rho/\rho_0)^{-s}$, where N is the number of units in the critical array, ρ and ρ_0 are, respectively, the uranium density in the array and in the unit, and N_0 and s are constants depending upon the neutron reflector surrounding the array. For arrays with no reflector N_0 and s are 2.506 ± 0.014 and 1.928 ± 0.028. Experiments with 8 and 27 unit arrays at lower U^{235} concentration indicate that the value of s increases and that of N_0 decreases with decreasing concentration. (auth)

292.
5968 (ORNL-3499(Vol.I)(p.50-7)) EFFECTS OF BOROSILICATE GLASS RASCHIG RINGS ON THE CRITICALITY OF AQUEOUS URANYL NITRATE SOLUTIONS. J. T. Thomas, J. K. Fox, and E. B. Johnson (Oak Ridge National Lab., Tenn.).

The natural-boron content of the glass varied from 0.5 to 5.7 wt %, and the volume of the vessel occupied by the glass ranged from 20.9 to 30%. Results from exponential experiments, using a critical layer of solution above the column of solution-ring mixture as a neutron source, provided estimates of the material buckling of the mixture as a function of solution concentration, boron content of the glass, and the glass volume present. It was shown, for example, that the buckling is negative if glass containing 4 wt % boron occupies more than 22% of the mixture volume, whereas the same concentration of glass containing only 0.5 wt % of boron results in positive values of the buckling except for solutions more dilute than about 72 g of uranium per liter. (auth)

293.
5969 (ORNL-3499(Vol.I)(p.58-62)) CRITICAL ARRAYS OF U(93.2) METAL CYLINDERS. J. T. Thomas (Oak Ridge National Lab., Tenn.).

Critical experiments were performed with three-dimensional arrays of individually subcritical units of U(93.2) metal with various thicknesses of paraffin reflector. The units consisted of cylinders containing about 21 and 26 kg of U(93.2) supported on stainless steel rods and separated by stainless steel spacers. In some arrays neutron-moderating materials surrounded the individual units. Within the range of the experiments, the effect of placing a 15.24-cm-thick paraffin reflector around an array of a particular density was to reduce the number of units required for criticality by a factor of ~12. The critical number was further reduced by a factor of ~2 by the addition of 1.270-cm-thick Plexiglas between adjacent units. (auth)

294.
30471 (ORNL-TM-868) CRITICAL THREE-DIMENSIONAL ARRAYS OF NEUTRON-INTERACTING UNITS. PART II. U(93.2) METAL. J. T. Thomas (Oak Ridge National Lab., Tenn.). July 1964. Contract W-7405-eng-26. 33p.

Criticality studies were made of three-dimensional arrays of uranium metal cylinders enriched to 93.2 wt % ^{235}U. Four weight groups of units, ranging from 10.5 to 26.2 kg of uranium in five sizes, were employed to determine the critical surface separation between units as a function of the number in an array. The critical uranium density (the ratio of the mass of a unit to the volume of the lattice cell it occupies in a critical array) was observed to depend upon a number of factors. Changes in k_{eff} of the individual units, caused by altering their shape, produced inverse changes in the density, provided the array shape was unchanged. The density of the array varied with its shape in a manner similar to that resulting from corresponding changes in a single critical unit. Addition of an hydrogenous reflector reduced the density of the array by factors much greater than those observed for similar reflection of a single critical unit. The density was found to be minimal when the units were separated by an hydrogenous moderator 4.9-cm thick. The separate effects of moderation and reflection of arrays are not additive. In several experiments it was observed that an array comprised of one-half of each of two different critical arrays was subcritical. Part I of the study is ORNL-TM-719. (auth)

295.
11403 LATTICE-PARAMETER MEASUREMENTS FOR SOLID AND CONCENTRIC TUBES OF URANIUM FUEL IN GRAPHITE LATTICES. D. E. Wood, K. R. Birney, and E. Z. Block (General Electric Co., Richland, Wash.). Nucl. Sci. Eng., 18: 116-25(Jan. 1964). (HW-SA-3013)

Lattice parameters were measured for natural uranium concentric tubes and solid fuel elements of 2.5 inches outside diameter in the Physical Constants Testing Reactor (PCTR). The primary quantities measured are the mass of

copper required to reduce k_∞ to one, copper activations throughout the lattice, and cadmium ratios for U^{238} capture, U^{235} and U^{238} fission, copper, gold, and lutetium. The results derived from these measurements are k_∞, f, p, ϵ, and the effective neutron temperature for the concentric tube in a 10.5-inch graphite lattice with both water and air in the coolant channels; k_∞ and f for the concentric tube in an 8.375-inch lattice, water cooled only; and k_∞, f, and p for the solid fuel in a 10.5-inch graphite lattice, water and air cooled. The effective fraction of the internal surfaces for resonance capture was determined to be 0.50. The average value of η for natural uranium in these lattices was found to be 1.30. (auth)

296.

36782 (HW-81659(p.33-44)) AN ANALYSIS OF PLUTONIUM–LIGHT WATER CRITICAL EXPERIMENTS. J. R. Worden, R. C. Liikala, and W. A. Reardon (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A detailed analysis of three sets of critical experiments, at 1.8, 2, and 5 wt % Pu in Pu–Al alloys, has been made. These experiments all involved critical approaches using PuAl rods in light water at various lattice spacings. Three areas of potential difficulty in the analysis of plutonium systems were studied: the basic cross section data; calculation of the space-energy distribution of thermalized neutrons in a lattice cell; and methods of calculating the neutron absorption rate in the 1.056-ev resonance of ^{240}Pu. (auth)

1965

297.

5477 (HW-80206) APPROACH TO CRITICAL AND CALIBRATION EXPERIMENTS IN THE PLUTONIUM RECYCLE CRITICAL FACILITY. R. A. Bennett and L. C. Schmid (General Electric Co. Hanford Atomic Products, Operation, Richland, Wash.). July 1964. Contract AT(45-1)-1350. 54p. Dep.; $3.00(cy), 2(mn) OTS.

The Plutonium Recycle Critical Facility (PRCF) is designed for investigations of reactor phenomena associated with either D_2O or H_2O moderator. The report describes the initial approach to critical experiment and power level, control rod, and moderator level calibration experiments conducted in the PRCF moderated with D_2O. Results of measurements of the longitudinal and radial flux distribution are included separate from the power calibration. Control rod calibration results include integral and differential worth measurements as measured by positive periods. Moderator level calibration results include the integral and differential worth of moderator level changes between 93 and 105 in. In a few cases the experimental results are compared to analytical results using three-group diffusion theory. The reactor is critical at a moderator level of 105 in. with 25 UO_2 fuel clusters surrounded by 30 Pu-Al fuel clusters at an 8 in. lattice spacing. Diffusion theory adequately describes the spatial distributions of the thermal-neutron density. The maximum flux in the reactor is 4.6×10^8 neutrons/cm^2 sec at 100 w total power. The control rods are worth approximately 2 mk. The moderator level coefficient of reactivity is 0.16 mk/in. at 105 in. (auth)

298.

10658 INITIAL EXPERIMENTS ON THE CORNELL UNIVERSITY ZERO POWER REACTOR CORES. Selwyn Solomon Berg. Thesis, Ithaca, N. Y., Cornell Univ., 1964. 264p.

The reactors studied are heterogeneous, normal water-moderated, uranium–dioxide-fueled cores with water-to-fuel ratios of 1:1, 1.5:1, 2:1, 3:1, and 4:1. The fuel rods are held in a hexagonal array by upper and lower grid plates. The control rods are clusters of three poison-fuel elements. Each control element is 0.660 in. in outside diameter and has an upper section of packed boron carbide and a lower section of uranium dioxide fuel. These control rods are designed to be located anywhere in the core and are guided by the grid plates; with the control rods at their upper limits, the core is uniformly fuel and moderator. The initial experiments measured critical sizes, flux shapes, cadmium ratios, peripheral fuel rod worths, water gap worths, and control rod worths. The values and probable errors for these parameters are tabulated. Hand-computational methods of determining k by a multigroup technique, which uses four energy groups, and a multifactor technique, which uses a resonance fission factor in addition to the usual four factors, are shown. Control rod worths are calculated by both a simple two-group model and a modified two-group model that takes the flux perturbation into account. Agreement between theory and experiment in the determination of k was as good as expected; k was shown to be within 6% of one for the multifactor method and within 4% of one for the multigroup method. The poor agreement found between theory and experiment for the determination of control rod worths indicates the need for a better theoretical model. (Dissertation Abstr., 25: No. 3, Sept. 1964).

299.

35752 (BNWL-52) EXPONENTIAL EXPERIMENTS AND NEUTRON MULTIPLICATION MEASUREMENTS WITH 1.25 WT PERCENT ENRICHED N-REACTOR FUEL ELEMENTS IN LIGHT WATER. Brown, C. L.; Lloyd, R. C.; Bierman, S. R.; Clayton, E. D. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Mar. 1965. Contract AT(45-1)-1830. 12p. Dep.(mn); $1.00(cy), 1(mn) CFSTI.

Subcritical experiments were performed with N-Reactor spike enriched fuel elements in light water to provide data for nuclear safety guidance. Exponential and neutron multiplication measurements were performed on 1.25 wt % enriched outer tubes (2.41 in. OD, 1.76 in. ID) and on tube-in-tube assemblies comprised of the outer tube and an 0.95 wt % enriched inner tube (1.25 in. OD, 0.44 in. ID) in each of four lattice pitches. Maximum bucklings were found to be 4680 microbucks for the outer tubes and 3900 microbucks for the tube-in-tube assemblies. Figures showing material buckling, critical mass, volume, slab thickness, and cylinder diameter as a function of water to fuel volume ratio are presented. Safe values for nuclear safety guidance are also presented. Pulsed neutron source measurements on two of the 1.25 wt % enriched outer tube lattices indicate the difference between delayed criticality and prompt criticality to be about five fuel columns for these lattices. (auth)

300.

10495 MEASUREMENT OF THE INFINITE MULTIPLICATION FACTOR IN A NATURAL URANIUM, LIGHT-WATER LATTICE. T. W. T. Burnett and T. G. Williamson (Univ. of Virginia, Charlottesville). Nucl. Sci. Eng., 21: 201-5(Feb. 1965).

The infinite multiplication factor, k_∞, is one of the basic

parameters of a subcritical assembly. Usually, these assemblies are designed for maximum k_∞; however, it is difficult to conduct laboratory experiments which yield a value of k_∞ to reasonable accuracy. Common methods, such as the loading technique and exponential experiment, are of doubtful validity or require apparatus not always available. Pulsing techniques are widely accepted, but are difficult to apply to reflected assemblies. An alternative approach is used. It is based on the integration of the thermal-neutron flux over the equivalent infinite medium. Use of variations in the method with poisoned assemblies eliminates the need for accurate determinations of the source strength, the absolute thermal-flux calibration, and the epithermal parameters of the medium. The theory is general and can be applied with a minimum of equipment. The results obtained from this method (and its variations) were checked by pulse measurements on the bare assembly and by a four-factor formula calculation. All results agree to within 2%. (auth)

301.

47955 (CONF-651103-6) MAXIMUM SAFE LIMITS FOR SLIGHTLY ENRICHED URANIUM AND URANIUM OXIDE. Clark, H. K. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Lab.). Sept. 9, 1965. Contract AT(07-2)-1. 21p. (SM-70/31). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

For Presentation at IAEA Symposium on Criticality Control of Fissile Materials, Stockholm.

A simple, approximate method is used for the calculation of safe limits for lattices of slightly enriched uranium and uranium oxide in water. The material buckling is calculated by an asymptotic multigroup transport code and the extrapolation distances by a two-group diffusion theory code. Data from exponential and critical experiments on these lattices in water are reviewed briefly. The calculations and experimental data were used to calculate critical and safe dimensions for spheres, infinite cylinders, and infinite slabs. (D.L.C.)

302.

42044 (MIT-2344-3, pp 37-54) STUDIES OF EPITHERMAL NEUTRONS IN URANIUM, HEAVY WATER LATTICES. D'Ardenne, Walter H. (Massachusetts Inst. of Tech., Cambridge. Dept. of Nuclear Engineering).

Measurements related to reactor physics parameters were made in three heavy water lattices. The three lattices studied contained .25 inch diameter, 1.03% ^{235}U uranium metal rods in triangular arrays spaced at 1.25, 1.75, and 2.50 inches. The following four microscopic parameters were measured in each of the three lattices studied: (1) the ratio of the average epicadmium ^{238}U capture rate in the fuel rod to the average subcadmium ^{238}U capture rate in the fuel rod, (2) the ratio of the average epicadmium ^{235}U fission rate to the average subcadmium ^{235}U fission rate in the fuel rod, (3) the ratio of the average ^{238}U capture rate to the average ^{235}U fission rate in the fuel rod, and (4) the ratio of the average ^{238}U fission rate to the average ^{235}U fission rate in the fuel rod. (M.O.W.)

303.

3521 (TID-21238) PROGRESSIVE SUBSTITUTION EXPERIMENTS IN UO_2 LATTICES MODERATED BY D_2O/H_2O MIXTURES. H. R. Frazen (Norway. Institutt for Atomenergi, Kjeller). Aug. 1964. 28p. (KR-90; NORA-21). Dep.(mn); $2.00(cy), 1(mn) OTS.

Buckling measurements for cores of uranium oxide (3% enriched) in different mixtures of D_2O/H_2O were performed in the NORA reactor by means of a progressive substitution technique. In order to check the results, some experiments were also carried out by the substitution technique in critical lattices for which the material buckling was already known. Some subcritical experiments were also performed to give additional information about the buckling obtained by substitution experiments. The analysis was done by three-region, two-group theory, and a correction was introduced in order to take into account the effect of the reflector. For a D_2O concentration of 99.50% and a lattice pitch of 6.544 cm, the material buckling with void was obtained by three-region, one group theory. All the results were found to agree satisfactorily with the results from critical experiments. Axial flux distributions were measured in some cases by using copper foils and a small fission chamber. (auth)

304.

33573 COMPARISON OF LATTICE PHYSICS EXPERIMENTS IN HEATED GRAPHITE STACKS CONTAINING PLUTONIUM–URANIUM FUEL WITH THEORETICAL PREDICTION. Gibson, M.; Harper, R. G. (Atomic Energy Establishment, Winfrith, Dorset, Eng.). J. Nucl. Energy, Pt. A & B, 19: 343-56(May 1965).

Measurements of buckling and flux fine structure and fission-rate distributions in graphite-moderated lattices fuelled with plutonium–uranium metal rods at temperatures up to 400°C are summarized. The results are of general interest in the development of rigorous methods of calculation for plutonium-bearing systems. The measured values are compared with theoretical calculations which determine a detailed representation of the neutron energy spectrum and incorporate fundamental nuclear data libraries. There is general overall agreement between the measured values and the theoretical predictions. (auth)

305.

42048 (MIT-2344-3, pp 125-36) TWO-REGION LATTICES. Gosnell, J. (Massachusetts Inst. of Tech., Cambridge. Dept. of Nuclear Engineering).

Investigations of the validity of lattice parameter measurements in the center of two-region assemblies are being conducted by using the M. I. T. lattice facility. Determination of spectral changes across the two regions was the main object of the experiments. In addition, axial buckling measurements were made at various radial positions in the assemblies to ascertain whether or not the axial relaxation length is a function of radius. (M.O.W.)

306.

29547 EXPERIMENTAL LATTICE PARAMETERS FOR 28-ROD CLUSTERS OF NATURAL UO_2 IN D_2O. R. E. Green, K. J. Serdula, and C. B. Bigham (Atomic Energy of Canada Ltd., Chalk River, Ont.). Trans. Am. Nucl. Soc., 8: 253-4(May 1965).

307.

6841 (NAA-SR-9771) MATERIAL BUCKLINGS OF CRITICAL AND SUBCRITICAL URANIUM-CARBIDE-FUELED GRAPHITE ASSEMBLIES. PART I. EXPERIMENT. O. R. Hillig and D. W. Latham (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). Dec. 1, 1964. Contract AT(11-1)-Gen-8. 71p. Dep.; $1.50(OTS).

A series of critical and subcritical experiments were

performed to determine values of material buckling and k_{eff} for graphite assemblies fueled with slightly enriched (3.02 wt %) uranium carbide fuel elements. Lattice arrangements were studied which included control channel mockups for triangular lattice pitches of 9, 11, and 12 inches. Experiments were performed to determine critical mass for two-region (uranium–carbide and uranium-metal-fueled) assemblies and k_{eff} for subcritical uranium–carbide-fueled assemblies. Exponential experiments were carried out to determine material buckling for subcritical assemblies containing uranium carbide fuel elements. Additional assemblies of varying size were used to study diffusion anisotropy and the apparent dependence of material buckling upon assembly size. The exponential studies were corrected for these effects. Material buckling and k_{eff} values were also inferred from the two-region critical mass measurements using four-group theory, and these values were compared to the results obtained from subcritical measurements. The values inferred from the critical mass measurements provided a more consistent set of results. These inferred values also were compared to calculated results. The agreement with results obtained using multigroup theory is very good while the agreement with results from few-group calculations is poor. (auth)

3566 MATERIAL BUCKLINGS OF URANIUM–CARBIDE-FUELED GRAPHITE ASSEMBLIES—MEASUREMENTS. O. R. Hillig and D. W. Latham (Atomics International, Canoga Park, Calif.). Trans. Am. Nucl. Soc., 7: 295(Nov. 1964).

308.

1203 (K-1619) THE ELEMENTS OF NEUTRON INTERACTING ARRAYS. PART I. UNREFLECTED, HOMOGENEOUS ARRAYS OF URANIUM METAL AND URANIUM SOLUTION MATERIALS. C. E. Newlon (Oak Ridge Gaseous Diffusion Plant, Tenn.). Oct. 13, 1964. Contract W-7405-eng-26. 42p. Dep.; $2.00(cy), 1(mn) OTS.

Consideration was given specifically to arrays whose units are individually safe for diffusion plant operations and to the derivation of optimum spacing criteria for these units. Part I of the report covers that portion of the work concerned with homogeneous arrays of interacting uranium solution units, in air, and also includes results for arrays of high-density uranium metal units. From the mass-volume relations developed from the calculations, it appears that, for equal volume arrays, the minimum critical U-235 mass will occur at an intermediate U-235 enrichment appreciably lower than the 93% level at which the single unit minimum critical mass is normally found. (auth)

309.

10496 CRITICALITY OF SLIGHTLY ENRICHED URANIUM IN WATER-MODERATED LATTICES. C. R. Richey, R. C. Lloyd, and E. D. Clayton (General Electric Co., Richland, Wash.). Nucl. Sci. Eng., 21: 217-26(Feb. 1965).

Exponential and approach-to-critical experiments were conducted with slightly enriched uranium in light water. The uranium was fabricated into elements of 0.175 to 1.66 in. diam with enrichments of 1.007 to 3.063 wt % ^{235}U. Exponential measurements were also made with natural uranium rods (0.925 in. diam) in light water. Analytical methods were used to correlate the experimental results and extend the data to include uranium rods containing 4.0 and 5.0 wt % ^{235}U. (auth)

310.

38126 (HW-84608(p.55-66)) SUMMARY OF RESULTS OF EBWR CRITICAL EXPERIMENTS. Schmid, L. C.; Lauby, J. H.; Stinson, W. P.; Uotinen, V. O. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

Critical experiments are carried out in PRCF with fuel rods containing 1.5 wt. % PuO_2–UO_2 and with water moderator. The center-to-center lattice spacing of the rods in the experiments is 0.71 in., which corresponds to a volume moderator-to-fuel ratio of 2.71. This ratio is the same as will be used in EBWR with these fuel rods, although the geometry in EBWR will differ from that in PRCF. (T.F.H.)

311.

8603 (WCAP-3269-25) CALCULATION OF LATTICE PARAMETERS AND CRITICALITY FOR UNIFORM WATER MODERATED LATTICES. L. E. Strawbridge (Westinghouse Electric Corp. Atomic Power Div., Pittsburgh). Sept. 1963. Contract AT(30-1)-3269. 57p. Dep.(mn); $2.00(cy), 2(mn) OTS.

A method of calculating criticality is described. The entire method has been programmed in the LEOPARD code. The method has the features described below. The loop model is eliminated in which the U-238 absorption was divided into two parts with the assumption that all U-235 epithermal fissions occurred at the breakpoint. Two empirical constants were used to describe the split in U-238 absorption and fast leakage. These constants are no longer necessary. A more general U-238 resonance integral correlation is introduced that is applicable to both uranium metal and uranium dioxide and that inherently includes the temperature effect. Thermal calculations are included in which modified Amouyal-Benoist disadvantage factors are calculated at every SOFOCATE energy level. A heterogeneous fast fission effect is introduced to replace the homogeneous calculation. In evaluating each change to the calculational procedure, the neutron multiplication factor (K) was calculated for 49 critical experiments using UO_2 fuel, and, in some cases, 20 exponential experiments using uranium metal fuel. The experiments cover a wide range of parameters and represent a severe test of any calculational procedure. The proposed calculations represent an improvement over previous methods as evidenced by a reduction in the standard deviation about the mean for several groups of experiments of particular interest in the design of pressurized water reactors. For example, the standard deviation of 10 experiments performed at Westinghouse Atomic Power Division using steel-clad UO_2 rods was reduced from 0.71 to 0.32%. The proposed method was found to agree with a more rigorous calculation which requires approximately 50 times as much computer time. (auth)

312.

21645 EXPERIMENTAL STUDIES ON NATURAL URANIUM–LIGHT WATER EXPONENTIAL PILE. [PART] II. Eiichi Takeda, Eiichi Arai, Keiji Kanda, and Shinichi Wakabayashi. Bull. Tokyo Inst. Technol., No. 61, 59-66 (1964).

The material buckling of the natural uranium-light water exponential pile was measured in two lattice piches on the basis that the values of material buckling and reflector saving are independent of the core size with the same core composition. The neutron multiplication factor calculated was compared with the theoretical values calculated by the four-factor formula. (auth)

313.

29559 DETERMINATION OF K_∞ FOR A PuO_2-UO_2 GRAPHITE LATTICE. T. B. Thornbury and N. A. Hill (Battelle-Northwest, Richland, Wash.). Trans. Am. Nucl. Soc., 8: 268(May 1965).

314.

24187 EXPERIMENTAL DETERMINATION OF THE ISOTHERMAL TEMPERATURE COEFFICIENTS OF REACTIVITY AND TEMPERATURE DEPENDENT THERMAL DISADVANTAGE FACTORS OF THE CORNELL UNIVERSITY ZERO POWER REACTOR. W. P. Wynn, Jr. (Cornell Univ., Ithaca, N. Y.). 121p. (CONF-654-76)

From American Nuclear Society Meeting, San Francisco, Nov.-Dec. 1964.

A comparison between experiment and theory was made for two-temperature dependence studies on the Cornell University Zero Power Reactor; a light water moderated, 2.1% enriched, UO_2 ceramic fueled, aluminum clad, triangular pitched critical assembly with nominal water-to-fuel ratios of 1:1, 1.5:1, 2:1, 3:1, and 4:1. The isothermal temperature coefficient of reactivity was calculated using two-group, two-region perturbation theory for an infinite cylinder. The temperature coefficient was predicted to be a linear function of the volumetric coefficient of thermal expansion for water, with intercept A and slope B. For four water-to-fuel ratios, excess reactivity was determined as a function of temperature from clean-core period measurements converted by the inhour equation. Reactivity differences were divided by temperature differences to obtain the isothermal temperature coefficient of reactivity. Values of the temperature coefficient were least-squares fitted to linear functions of the coefficient of expansion of water to determine A and B. Disagreement between measured and calculated values of A and B indicates a need for refinement of the calculations. Thermal disadvantage factors were measured over a temperature range from 5 to 65°C for three water-to-fuel ratios using dysprosium, manganese, and vanadium foils. Two integral foil counting techniques were used. One was to dissolve the manganese foils in nitric acid and gamma-count aliquots of the solutions in a well-type scintillation counter. The other method was to beta-count the solid foils in a proportional flow counter. Results of the two methods agreed within experimental error. Measured values of thermal disadvantage factors were compared to theory and found to be a few per cent low. Further experimental studies are suggested to investigated the disagreement. Thermal disadvantage factors were least-squares fitted to a linear function of temperature to determine the temperature dependence. Measured and calculated values of the temperature dependence were found to agree within the experimental errors. (auth)

1966

315.

16034 THE CRITICALITY OF INTERACTING ARRAYS. Abbey, F. (United Kingdom Atomic Energy Authority, Risley, Eng.). Vienna, International Atomic Energy Agency, 1965, Preprint No. SM-70/6, 39p. (CONF-651103-16). ORINS.

From IAEA Symposium on Criticality Control of Fissile Materials, Stockholm.

Central to an understanding of the criticality of interacting arrays is a knowledge of the quantitative relation between the reactivity and separation of units. An examination of this relation for regular arrays of spheres is made in terms of a simple theoretical model which focuses attention on the interaction probabilities between pairs of units, i.e., the proportion of the neutrons emitted by one unit which reach another. Formulation of balance equations connecting the fluxes entering and leaving each unit then allows assessment of the criticality of the array. The interaction probabilities are taken as mean fractional solid angles. Attention is drawn to the possibility of using Monte Carlo methods for the evaluation of interaction probabilities for systems of complex geometry and the GEM Monte Carlo code for the IBM 7090 is used to evaluate cases where other spheres in an array partially intervene between source and target spheres. The values obtained are separately tabulated as being of interest outside the particular arrays considered. A new code incorporating a specially simplified variant of GEM and provision for solution of the neutron balance equations is described as a ready means for estimation of the criticality of arrays for which the unit reactivities are known. A related discussion of factors affecting the criticality of arrays is presented. Factors considered include the size, shape, composition and density of units, proximity to reflectors and the mixing of units of differing types. Comparison is made between the theoretical model and published critical data for both unreflected and reflected arrays. Recently reported data for systems of this kind from both experimental and calculational sources allow a reasonably comprehensive comparison. New results are also given on the effect of moving units out of the plane of a two dimensional array and on the interaction of small numbers of slab tanks. The latter is seen as a special case and is investigated in some detail; a simple empirical correlation is given for air spacing and it is shown that there is little to be gained by attempting decoupling with shielding materials. (auth)

316.

26427 (AECL-2330, pp 218-32) CRITICAL EXPERIMENTS WITH THE ORGANIC COOLANTS, MONO-ISOPROPYLDIPHENYL AND GAS OIL. Aleksenko, Yu. N.; Buinitskaya, V. I.; Zaslavskii, V. V.; Zvonov, N. V.; Koslov, V. N.; Meshcheryakov, M. N.; Rogorzhkin, I. V.; Stolpnik, V. P.; Strogonov, V. A.; Yaroslavtsev, B. E. Translated from [pp 182-93] of Issledovaniya po Primeneniyu Organicheskikh Teplonositelei-Zamedlitelei v Energeticheskikh Reaktorakh, Atomizdat, Moscow, 1964.

Critical experiments were performed using fuel elements of type EK-10 (these are standard fuel elements used in reactors of type VVR-S) and the organic moderators monoisopropylbiphenyl (which is a product of the alkylation of diphenyl) and gas-oil (a straight distillation fraction of natural oil representing a mixture of aromatic naphthene and paraffin hydrocarbons) as well as an aqueous moderator. The main results of these critical experiments and a description of the reactor are included. (M.O.W.)

317.

5071 STUDY OF A SLIGHTLY ENRICHED UO_2 LATTICE WITH H:U = 0.42—MEASUREMENT AND ANALYSIS. Baer, W.; Hardy, J. Jr.; Klein, D.; Volpe, J. J.; Palowitch, B. L.; Frantz, F. S. Jr. (Westinghouse Electric Corp., Pittsburgh). Nucl. Sci. Eng., 23: 361-7(Dec. 1965). (WAPD-T-1631).

Parameter measurements in a 1.3% enriched UO_2 lattice with H:U = 0.42 were performed. The measurements are an extension of an experimental program in the TRX critical facility of the Bettis Atomic Power Laboratory. Earlier

measurements were made for a wide range of water-to-uranium (H_2O:U) volume ratios (1:1 to 8:1) using 4-ft (1.2-m)-high slightly enriched, 0.387-in. (0.98-cm)-diam uranium metal or oxide fuel rods clad with aluminum. The new data were compared with current analytic techniques, using both P-1 and P-3 multigroup analysis in the epithermal-neutron energy range and Monte Carlo multigroup methods for thermal neutrons. This extremely undermoderated lattice provides a very stringent test for both the computational methods and the neutron cross sections used. The quantities measured were: the ratio of epithermal-to-thermal radiative captures in ^{238}U (ρ^{28}); the ratio of captures in ^{238}U to fissions in ^{235}U (the modified conversion ratio, CR*); the ratio of ^{238}U fissions to ^{235}U fissions (δ^{28}); and the ratio of epithermal-to-thermal ^{235}U fissions (δ^{25}). In addition, activations were obtained with thermal-neutron detectors of widely different spectral response. The results indicate that the calculational methods predict the parameters very well, except for δ^{28}. The discrepancy in δ^{28} may be due to inadequate ^{238}U inelastic scattering cross sections, but this conclusion requires additional study. Monte Carlo calculations of thermal-neutron detector activations show that use of either the Nelkin or Koppel kernel gives results that agree with the data. (auth)

318.

14065 (MIT-2344-4, pp 4-16) MATERIAL BUCKLING AND INTRACELLULAR THERMAL NEUTRON DISTRIBUTIONS. Barch, J.; Berube, N.; Kelley, B.; Supple, A.; Clikeman, F. M. (Massachusetts Inst. of Tech., Cambridge. Dept. of Nuclear Engineering).

Standard methods are developed to measure the material buckling and the intracellular thermal neutron distributions for lattices studied in the M. I. T. Lattice Facility. Results are obtained by these techniques on three lattices of uranium rods in D_2O. Results are given for 1.143% ^{235}U-enriched, 0.25-inch-diameter rods in triangular lattice spacings of 1.75 and 2.50 inches, and 0.947% ^{235}U-enriched, 0.75-inch-diameter uranium rods in a 2.50-inch triangular spacing. (auth)

319.

32787 STUDIES OF SUBCRITICAL LATTICES WITH DISTRIBUTED NEUTRON ABSORBER BY MEANS OF THE PULSED NEUTRON SOURCE TECHNIQUE. Bliss, H. E. 10p. (CONF-651101-76). ORAU. Gmelin, AED-CONF-65-332-106.

From 13th Conference on Remote Systems Technology, Washington, D. C.

The theoretical base for pulsed-neutron measurements of the multiplying properties of subcritical lattices, using the addition of a distributed neutron absorber, are outlined; and results are presented for two clean lattice configurations. The results of steady-state measurements in the same lattices and in modified lattices are also presented. (D.C.W.)

320.

22375 (BNWL-193, pp 45-56) MEASUREMENTS OF MATERIAL BUCKLINGS FOR 1.002, 1.25, AND 1.95 WT PERCENT ^{235}U ENRICHED URANIUM TUBE LATTICES IN LIGHT WATER. Brown, C. L.; Lloyd, R. C. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

Material bucklings and extrapolation distances were measured for light water lattices of 1.002, 1.25, and 1.95 wt % ^{235}U enriched uranium tubes, and 1.002 wt %/1.002 wt % and 1.25 wt %/0.95 wt % ^{235}U enriched uranium tube-in-tube assemblies. Detailed results are presented. (M.O.W.)

321.

3393 RECENT EXPONENTIAL AND CRITICAL APPROACH EXPERIMENTS WITH SLIGHTLY ENRICHED URANIUM METAL TUBES IN LIGHT WATER. Brown, C. L.; Lloyd, R. C. (Battelle-Northwest, Richland, Wash.). Trans. Amer. Nucl. Soc., 8: 443-4(Nov. 1965).

322.

34926 CRITICAL EXPERIMENTS AND CALCULATIONS IN SOME VERY UNDERMODERATED, SLIGHTLY ENRICHED, URANIUM-OXIDE/WATER LATTICES. Boynton, A. R.; Pennington, E. M. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 9: 129(June 1966).

323.

26425 (DP-1014) CRITICAL AND SAFE MASSES AND DIMENSIONS OF LATTICES OF U AND UO_2 RODS IN WATER. Clark, Hugh K. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Lab.). Feb. 1966. Contract AT(07-2)-1. 60p. (CONF-651103-29). Dep. mn. CFSTI $3.00 cy, $0.75 mn.

From IAEA Symposium on Criticality Control of Fissile Materials, Stockholm.

A survey is given of available critical and exponential data data obtained with water-reflected lattices of slightly enriched uranium and uranium oxide rods. Calculations are made for these lattices by an asymptotic multigroup buckling code and by a two-group diffusion theory code employing parameters generated in the first code. Comparison between calculations and experiments is made in terms of a k_{eff}, which is the ratio of the calculated k to that calculated from the experimental dimensions and flux traverses and from the calculated migration areas and extrapolation distances. For some of the data, comparisons are also made between the present method of calculation and a more highly sophisticated method. Reasonable care is taken in the present calculations to take account of all important effects; but since the experiments are used to normalize the calculations, absolute accuracy in the calculations is not necessary. A least squares treatment is given to k_{eff}, the parameter relating calculation and experiment, in order to obtain an average curve of k_{eff} as a function of the ratio of water to uranium. At low enrichments, k_{eff} is also allowed to vary with rod diameter and enrichment. Nearly all values of k_{eff} lie within ±0.01 of the average curves. Critical masses and dimensions are calculated with the buckling and two-group codes to correspond to values of k_{eff} lying on the average curves; and safe masses and dimensions are calculated to correspond to values of k_{eff} lying 0.02 below the average curves. Tables of minimum critical and maximum safe values are presented as a function of enrichment. (auth)

324.

30836 (BAW-3492-1) LUMPED BURNABLE POISON PROGRAM. Final Report. Clark, R. H.; Batch, M. L.; Pitts, T. G. (Babcock and Wilcox Co., Lynchburg, Va. Critical Experiment Lab.). Jan. 1966. Contract AT(30-1)-3492. 60p. Dep. mn. CFSTI $3.00 cy, $0.75 mn.

A series of critical experiments was conducted to study the physics effects of cylindrically shaped, lumped poison

distributed throughout a light-water-moderated nuclear reactor. The lumped poisons were borosilicate glass rods (12.6% B_2O_3) of 0.460- and 0.326-inch diameters, silica glass rods (3.0% B_2O_3) of 0.326-inch diameter, and aluminum-clad B_4C rods. The fuel was 4 and 2.5%-enriched UO_2 rods of 0.475-inch OD. Lattice non-moderator/moderator volume ratios were 0.750. Measurements included critical size and composition, poison rod reactivity worth, $\partial\rho/\partial h$ and excess reactivity, gross power distribution, and thermal flux distribution around the central poison rod. Measurements are tabulated, and results are summarized and discussed. (auth)

325.

32783 METHODS OF MATERIAL BUCKLING DETERMINATION IN THE HELENA SUBCRITICAL ASSEMBLIES. Dabrowski, Cyryl; Mikulski, Andrzej T. 33p. (CONF-651123-7). ORAU.

From Conference on Reactor Physics and Engineering, Budapest.

47072 METHODS OF MATERIAL BUCKLING DETERMINATION IN THE HELENA SUBCRITICAL ASSEMBLIES. Dabrowski, Cyryl; Mikulski, Andrzej T. (Inst. of Nuclear Research, Swierk, Poland). Nukleonika, 11: 121-35(1966).

A detailed analysis of the material buckling determination in the natural uranium fuelled graphite moderated HELENA subcritical assemblies was carried out by means of the following methods: thermal neutron flux measurement along the vertical axis with an assumption of the point thermal neutron source either in one-region or two-region assembly, measurement of the thermal neutron flux along the vertical axis and in one horizontal plane, measurement of the thermal neutron flux in two horizontal planes, and measurement of the thermal neutron flux in one vertical plane of the assembly. Three assemblies of 24 cm lattice pitch and the following dimensions: 180 × 180 × 237, 180 × 240 × 237 and 240 × 240 × 237 cm were investigated. The following results have been obtained: migration area anisotropy coefficient $\alpha_n = 1.056 \pm 0.045$ and material buckling $(73.5 \pm 4.3) \times 10^{-6}$ cm^{-2}. (auth)

326.

3397 REACTOR PHYSICS MEASUREMENTS IN SLIGHTLY ENRICHED URANIUM D_2O LATTICES. D'Ardenne, W. H.; Kaplan, I.; Lanning, D. D.; Thompson, T. J. (Massachusetts Inst. of Tech., Cambridge). Trans. Amer. Nucl. Soc., 8: 447-8(Nov. 1965).

327.

46993 (DP-1054) LATTICE MEASUREMENTS ON TUBULAR FUEL ASSEMBLIES IN D_2O. Fike, Harold R. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Lab.). Aug. 1966. Contract AT(07-2)-1. 16p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

Substitution measurements in the Process Development Pile (PDP) were used to determine bucklings and diffusion coefficients for D_2O-moderated lattices of tubular natural UO_2 fuel assemblies. Four types of assemblies were investigated at triangular lattice pitches of 9.33, 11.10, and 12.12 inches. Coolants within the different assemblies included D_2O, air, and mockups of H_2O fog and organic liquids. (auth)

328.

40381 UTILIZATION AND EFFECTIVENESS OF LINEAR ARRAYS OF NEUTRON-ABSORBER-FILLED TUBES AS CONTROL RODS IN WATER MODERATED NUCLEAR REACTORS. Fisher, John Raymond. College Park, Md., Univ. of Maryland, 1966. 167p.

Thesis.

An experimental and analytical investigation was made with the University of Maryland reactor to determine the reactivity worth of linear arrays of absorber-filled tubes as compared to solid slabs of the same material. Measurements were made as a function of tube diameter and spacing between the tubes in both an air and water environment. An analytical model was devised to correlate the experimental results; this was then applied to a typical water-moderated power reactor to determine the effect of spacing the absorber-filled tubes. Solid slabs of Ag-In-Cd, Boral, and cadmium and stainless steel tubes filled with Ag-In-Cd and B_4C were used for the measurements. The slabs varied in thickness and could be arranged in a close-packed linear array to form a slab, or could be arranged in a linear array with gaps between adjacent tubes. The reactivity worth of the samples was measured by placing them in a lucite box inserted into the center of the University of Maryland Reactor. Reactivity worths were determined from the difference in critical positions of the reactor regulating rod and reduced to the reactivity per unit area of an infinitely long and wide sample by means of a correction factor. The difference between the solid slab and close-packed array of tubes was found experimentally to be small, 5 percent for the smaller diameter tubes and negligible for the larger ones, in both an air and water environment. The effect of spacing was found to be a function of tube diameter as well as the width of the gap. The analytical model was applied to a typical large water reactor controlled by a solid cruciform of Ag-In-Cd and a cruciform of spaced tubes filled with Ag-In-Cd. For the particular case analyzed, it was determined that a reduction of 45 percent in absorber material could be accomplished with a reduction in the shutdown capability of only 0.8 percent Δk for the stuck rod condition. (TCO)

329.

46994 (ORNL-3272) CRITICAL MASS STUDIES. PART XI. CRITICAL PARAMETERS OF URANIUM-ALUMINUM ALLOY SLUGS. Fox, J. K.; Gilley, L. W. (Oak Ridge National Lab., Tenn.). May 28, 1962. Contract W-7405-eng-26. 27p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

Critical and near-critical experiments were performed with cylindrical pieces of two alloys of 93%-^{235}U-enriched U and Al to establish safe practices in fabrication and in other procedures. In one set of experiments the U content of the alloy was 5.4 wt % and the dia of the pieces was 1.015 in. The minimum critical number of 12-in.-long slugs in a single water-reflected and -moderated tier was 134; the minimum number in a double tier was about 165. Experiments with 8-ft-long, optimumly spaced rods gave the dia of an infinitely high, unreflected cylindrical array as between 15 and 17.5 in. and as approximately 12 in. for a totally water-reflected cylindrical array. The second alloy, which was studied in detail earlier, contained 7.6 wt % U and was made into cylinders 1.35 in. in dia. The minimum number of 8-in.-long slugs which could be made critical in a square lattice was 101, arranged in two tiers, and this agreed with the earlier results. The minimum

critical number in a similar hexagonal lattice was 100. A double tier of 1000 closely packed 8-in.-long slugs was subcritical when submerged in water. The dia of an infinitely long, unreflected critical cylindrical array of optimumly spaced rods was between 15 and 17.5 in.; the dia of a water-reflected cylindrical array was about 12 in. It was shown that randomly spaced slugs produce less source neutron multiplication than do lattices of the same overall density. (auth)

330.

38532 (NASA-TN-D-3555) CRITICAL MASS STUDIES WITH THE NASA ZERO POWER REACTOR II. III. HETEROGENEOUS ARRAYS OF CYLINDRICAL VOIDS. Fox, Thomas A.; Mueller, Robert A.; Ford, C. Hubbard (National Aeronautics and Space Administration, Cleveland, Ohio. Lewis Research Center). Aug. 1966. 24p. CFSTI $1.00 cy, $0.50 mn.

The ZPR-II was used to determine several critical cylindrical configurations of aqueous fuel solutions that contain heterogeneous arrays of voids. These voids are cylindrical, symmetrically arranged parallel to the axis of the reactor, and extend the height of the core. A wide range of highly enriched aqueous uranyl fluoride fuel concentrations are studied. Data are also presented on the thermal neutron flux distributions in the central radial plane and on the variation of void reactivity importance with radial position. (H.D.R.)

331.

34921 EXPERIMENTAL BUCKLINGS FOR SEVEN-ROD CLUSTERS OF NATURAL UC IN HEAVY WATER MODERATOR. Green, R. E. (Atomic Energy of Canada Ltd., Chalk River, Ont.); Kay, R. E.; Okazaki, A.; Casini, G.; Metzdorf, H. J.; Paillon, M. Trans. Amer. Nucl. Soc., 9: 123-4 (June 1966).

332.

45078 (MIT-2344-6) USE OF NEUTRON ABSORBERS FOR THE EXPERIMENTAL DETERMINATION OF LATTICE PARAMETERS IN SUBCRITICAL ASSEMBLIES. Harrington, J.; Lanning, D. D.; Kaplan, I.; Thompson, T. J. (Massachusetts Inst. of Tech., Cambridge. Dept. of Nuclear Engineering). Feb. 1966. Contract AT(30-1)-2344. 202p. (MITNE-69). Dep. mn. CFSTI $6.00 cy, $1.25 mn.

A new method for measuring values of the multiplication factor for an infinite medium, k_∞, was developed. The method involves addition of neutron absorbers to subcritical assemblies. Measurements of the values of k_∞ for three lattices of U metal rods in heavy water were made. Solid copper rods were used as absorbers. The values of k_∞ were found to be in agreement, within experimental uncertainties, with values obtained with the four-factor formula. The method used is closely related to the Hanford (PCTR) technique for measuring k_∞ which requires a critical reactor. Since it is nearly a measurement of $(k_\infty - 1)$, and because the measured values are independent of a number of sources of error in the four-factor formula, the uncertainties in the measured values should be smaller than those obtained with the four-factor formula. The work shows that a 3% uncertainty in $(k_\infty - 1)$ may be obtained with the method, comparable with the uncertainty in PCTR measurements. Measured values of the neutron regeneration factor were also determined. They agree well with values computed with the THERMOS code. The results of exploratory experiments indicate that it may be possible to do the measurements in two-region subcritical assemblies. (auth)

333.

10331 (UJV-1412/65) VLIV PRIMESI H_2O NA REAKTIVITU TEZKOVODNIHO REAKTORU S PRIRODNIM URANEM. (Effects of H_2O Addition on the Reactivity of Heavy Water Moderated Reactor Fueled with Natural Uranium). Hron, M. (Ustav Jaderneho Vyzkumu, Ceskoslovenska Akademie Ved, Rez). 1965. 16p. ORAU.

The reactivity effects of H_2O addition into the moderator of a heavy water moderated reactor with natural uranium are investigated. The values for various radii of cylindrical fuel rods and various water-uranium ratios were calculated as a function of light water concentration in the moderator. The maximum values of the buckling which can be attained by addition of H_2O at various parameters of the lattice are discussed. It appears that the reactivity increases only in systems with a low moderator-to-fuel volume ratio in the cell. (auth)

334.

3379 (ORNL-TM-1207) CRITICAL EXPERIMENTS WITH SPERT-D FUEL ELEMENTS. Johnson, E. B.; Reedy, R. K. Jr. (Oak Ridge National Lab., Tenn.). July 14, 1965. Contract W-7405-eng-26. 17p. Dep. mn; CFSTI $1.00 cy, $0.50 mn.

The critical dimensions of lattices of SPERT-D fuel elements in several non-reactor environments were determined to establish specifications for use in storage, transportation, and chemical processing operations. Each fuel element contained about 300 g of ^{235}U in 22 aluminum-clad flat plates. In addition to lattices with water moderator and reflector, a dilute aqueous solution of uranyl nitrate was used in some experiments to simulate a dissolver. In still other experiments, varying amounts of boron were added to the uranyl nitrate solution to determine its effect as a soluble neutron absorber in chemical process equipment. It was shown, for example, that a minimum of about 3.5 kg of ^{235}U is required in a critical lattice moderated and reflected by water. This mass is reduced to 2.8 kg (contained in the elements) when U(92.6) solution having a ^{235}U concentration of 3.99 g/liter was substituted for water. It was increased to 13.6 kg when 1.118 g of natural boron was added to each liter of the uranyl nitrate solution. (auth)

335.

38441 (BAW-3492-1(Suppl.1)) ANALYSIS OF LUMPED BURNABLE POISON CRITICAL EXPERIMENTS. Jones, H. M.; Alcorn, F. M.; Paulson, A. E. (Babcock and Wilcox Co., Lynchburg, Va. Atomic Energy Div.). Apr. 1966. Contract AT(30-1)-3492. 29p. (TRG-66-22). Dep. mn. CFSTI $2.00 cy, $0.50 mn.

The developmental work on nuclear analytical methods for distributed control pins is summarized. All the calculations utilized a nuclear code package that was selected after a comparative study of available codes. Four critical experiments have been selected as typical of the B poison pins used in the experiments; B concentration, pin size, and pin distribution were considered. The reported error of the calculated multiplication constant indicates the adequacy of the analytical model. Further work required in some areas is discussed. (auth)

336.

14080 PULSED NEUTRON STUDIES OF BeO-NATURAL URANIUM LATTICES. Joshi, B. V.; Nargundkar, V. R.; Subbarao, K. (Atomic Energy Establishment

Trombay, Bombay). pp 231-37 of STI/PUB/104(Vol. 2).

The use of the pulsed neutron method for determination of the diffusion parameters and multiplication constants of lattices is described. The diffusion parameters and multiplication constants of BeO-natural uranium lattices are obtained by this method. The uranium rods used were 2.92 cm diameter clad in aluminum 0.072 cm thick and were arranged vertically in channels of square cross section 5 × 5 cm, in a square lattice of pitch 15 cm. The neutron bursts were produced from a cascade accelerator by pulsing the ion source and using the Be(d,n) reaction. The detectors were enriched boron trifluoride proportional counters. The space and time harmonics were eliminated. The decay constant of the fundamental mode was determined for several bucklings, far from criticality. In these calculations the effect of diffusion cooling and anisotropy was taken into account. All the diffusion and multiplication constants were determined by the method of least squares fit by three different approaches. The prompt critical buckling was determined by solving the decay constant equation for $\lambda = 0$. (auth)

337.

20199 (AEEW-R-466) A COMPARISON OF MEASUREMENTS ON PuO_2–UO_2 FUELLED GRAPHITE LATTICES WITH CALCULATIONS USING THE ARGOSY METHOD. King, D. C. (Atomic Energy Establishment, Winfrith (England)). Jan. 1966. 48p. Dep. BIS $1.40. HMSO 7s.

The predictions of the ARGOSY method of calculation are compared with buckling and reaction rate measurements on graphite lattices containing plutonium enriched oxide cluster fuels. Most of the measurements were obtained on one lattice pitch giving a graphite-to-fuel volume ratio similar to that in the Windscale A.G.R. Apart from a reduction of 10% in the values of the capture and fission resonance integrals (4 ev to 10 kev) of all nuclides the method as coded in ARGOSY III uses basic nuclear data. It is shown that the buckling predictions of ARGOSY III at room temperature are in agreement with the experimentally determined values within approximately the experimental error, i.e., equivalent to ±0.5% in reactivity. When systematic errors are removed, however, a linear trend with the fraction of fissions occurring in ^{239}Pu is evident; the calculated reactivity being (0.6 ± 0.3)% high for fuel with 80% of fission occurring in ^{239}Pu. The experimentally determined change of buckling between 20 and 390°C is predicted to within the experimental error, i.e., equivalent to ±0.7 mn/°C. Reaction rate ratios and radial power distributions at all temperatures are predicted well by ARGOSY III. (auth)

338.

2938 (UCRL-14245) INTERACTION IN ARRAYS OF FISSIONABLE MATERIALS. Kolar, O. C.; Morton III, J. R.; Pruvost, N. L. (Lawrence Radiation Lab., Univ. of California, Livermore). Oct. 5, 1965. Contract W-7405-eng-48. 32p. (CONF-651103-12). Dep. mn; CFSTI $2.00 cy, $0.50 mn.

From IAEA Symposium on Criticality Control of Fissile Materials, Stockholm.

The interaction effect is studied in arrays of fissionable materials. Both experimental and theoretical efforts are reported. The arrays studied experimentally are composed of Pu metal units. Array geometries are simple; the basic units are cylinders and the arrays are cubical. Bare arrays are studied, as well as those with internal moderation or external reflection. 130 basic units are available so that arrays up to 5 × 5 × 5 in size can be studied. The purposes of the study are two-fold: to provide clean, critical data to use in the development and normalization of computational techniques; and the data as such are expected to be useful for interpolation and extrapolation for application to actual systems of interest. The basic array unit is a 3-kg cylinder of Pu metal, 4.62 cm high, 6.53 cm in diameter, and of density 19.54 g/cm^3. The height-to-diameter ratio is chosen so that a double unit consisting of two basic units, one on top of the other, will have the same shape factor as a single basic unit. A brief description of the canning, assembly equipment, and the experimental building is presented. A short discussion of the safety evaluation (startup accident, step insertion of reactivity) is also given. Pertinent calculations are performed using both the GEM and SORS neutron Monte Carlo electronic computer programs. Experimental results for arrays consisting of two units up to the maximum number are presented. (auth)

339.

42609 (WAPD-T-1886) ^{233}U THORIUM SEED-BLANKET EXPERIMENTAL PROGRAM. Korsmeyer, J. D.; Milani, S.; Raab, H. F. Jr. (Bettis Atomic Power Lab., Pittsburgh, Pa.). Apr. 1966. Contract AT-11-1-GEN-14. 16p. (CONF-660606-18). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From American Nuclear Society Meeting, Denver, Colo.

The lattice characteristics of seed fuel elements containing ^{233}U and ^{235}U were measured in a series of small seed-blanket critical assemblies. The measurements included critical axial bucklings, seed power shapes, the ratio of epithermal to thermal thorium captures in blanket regions, and the ratio of epithermal to thermal ^{233}U fissions through seed and blanket regions. The experimental results are compared with calculations performed with diffusion theory, Monte Carlo, and neutron transport techniques. (H.D.R.)

340.

20210 (WCAP-3269-39) CRITICAL EXPERIMENTS PERFORMED WITH CLUSTERED AND UNIFORM ARRAYS OF RODDED ABSORBERS. Leamer, R. D.; Hanlen, D. F.; Hamilton, G. N.; Taylor, E. G. (Westinghouse Electric Corp., Pittsburgh, Pa. Atomic Power Div.). Nov. 1965. Contract AT(30-1)-3269. 110p. Dep. mn. CFSTI $4.00 cy, $0.75 mn.

A series of critical experiments was undertaken in which the reactivity worths of clusters and uniform arrays of Ag–In–Cd absorber rods were measured in water-moderated uranium-oxide cores of low enrichment. Experimental parameters included enrichment, lattice pitch, diameter of poison and fuel rods, and absorber rod cladding thickness. Reactivity worths were obtained by measuring the critical water heights of the test and reference cases, and integrating the differential water worth curve between those limits. Some of the experiments had very small worths (e.g., absorber rod cladding), so good core reproducibility was required. These experiments were performed using a variety of fuel including stainless steel clad fuel rods of 2.7 wt %, 3.7 wt %, and 5.7 wt % enrichments, and Zircaloy-4 clad rods of 2.72 wt % enrichment. (M.O.W.)

341.
34925 EXPERIMENTAL AND ANALYTICAL RESULTS FOR PuO_2–UO_2–H_2O LATTICES. Liikala, R. C.; Stinson, W. P. (Battelle-Northwest, Richland, Wash.). Trans. Amer. Nucl. Soc., 9: 127-8(June 1966).

342.
8349 (RT/FI(65)14) SITUAZIONE ATTUALE DEI METODI DI CALCOLO DEI RETICOLI CONTENENTI PLUTONIO. (Current Position on Methods for Calculating Plutonium Lattices). Loizzo, P.; Tondinelli, L. (Comitato Nazionale per l'Energia Nucleare, Rome (Italy)). July 1965. 13p. Dep. mn.

Six critical experiments in PuO_2 lattices performed at Hanford are calculated with different methods. (auth)

343.
12370 INFLUENCE OF THEORETICAL CROSS SECTIONS AND THERMAL SPECTRUM MODELS IN THE CALCULATIONS FOR WATER REACTORS CONTAINING PLUTONIUM. Loizzo, P.; Lopez, S.; Tondinelli, L. Comit. Naz. Energia Nucl., Notiz., 11: No. 8-9, 44-7 (Aug.-Sept. 1965). (In Italian).

The problems in the neutron calculation in a reactor containing ^{239}Pu and the difference with ^{235}U are examined. A general scheme for the calculation is proposed and this plan is optimized to the neutron cross section data used. A comparison is made between theory and experimental values for the Hanford critical experiments. (J.S.R.)

344.
45084 SLIGHTLY ENRICHED URANIUM SINGLE-ROD HEAVY-WATER LATTICE STUDIES. Lutz, H. R.; Heer, W.; Meier, R. W. (Federal Inst. for Reactor Research, Wuerenlingen, Switzerland). Neue Tech., 8: 130-42(June 1966).

The results of a series of buckling and microparameter measurements on square lattices of 0.96% enriched U metal rods are given. The experiments were performed in the subcritical facility MINOR. A special chapter deals with approach to criticality experiments. These were necessary in order to determine whether a lattice would reach criticality at finite D_2O levels. The method used gave very reliable extrapolations for the determination of critical D_2O-heights. The experimental bucklings are compared with 4 buckling codes and show considerable deviations in the undermoderated region. (auth)

345.
18143 CALCULATION OF LATTICE PARAMETERS OF THE WWER CORE FOR COMPARISON WITH EXPERIMENTAL RESULTS ON CRITICAL ASSEMBLIES. Mueller, A.; Zenker, W. (VEB Atomkraftwerk, Berlin). Kernenergie, 8: 602-5(1965). (In German).

Lattice parameters for assemblies of fuel element bundles of the WWER type with uniform and different ^{235}U enrichment were calculated by the programs GRUPA and EMMI and compared with the values determined experimentally on critical assemblies. Although the calculational methods are relatively simple, a good agreement resulted between the experimentally determined and the calculated values of the material buckling B_m^2, an effective migration area M_n^2 and the infinite multiplication factor k_∞ of the lattice. (auth)

346.
34923 MEASUREMENT OF LATTICE PARAMETERS IN NATURAL-URANIUM/HEAVY-WATER LATTICES USING A FEW RODS. Okazaki, A.; Craig, D. S. (Atomic Energy of Canada Ltd., Chalk River, Ont.). Trans. Amer. Nucl. Soc., 9: 125-6(June 1966).

347.
8465 (RFX-316) REACTOR PHYSICS MEASUREMENTS IN RO WITH 27-ROD CLUSTERS OF NATURAL UO_2, 1.35 CM IN DIAMETER. Persson, R.; Wikdahl, C.-E. (Aktiebolaget Atomenergi, Stockholm (Sweden)). Sept. 30, 1964. 28p. Dep. mn.

Clusters with three different pin pitches were investigated by means of the substitution technique as regards the material buckling versus lattice pitch and the changes of buckling and diffusion coefficients caused by the introduction of air as "coolant." The buckling change with void is found to be in approximate agreement with that predicted by a modified version of BURNUP-5 calculations, but the radial diffusion coefficient changes less than expected giving rise to a more pronounced anisotropy. The measured changes of the diffusion coefficients can, however, be well correlated by means of a simple formula combining two extreme definitions of the average diffusion coefficient. Activation studies were performed in one lattice configuration on voided and D_2O-filled clusters. The fine structure was measured by means of copper foils and wires. (auth)

348.
3600 NEUTRON SPECTRUM MEASUREMENTS IN HETEROGENEOUS MEDIA [Thesis]. Salah, Sagid. Gainesville, Fla., Univ. of Florida, 1964. 145p.

Neutron spectra were measured both integrally and differentially and then compared with theoretical calculations. Measurements were made in highly absorbing media in the University of Florida Training Reactor (UFTR) thermal column, in the UFTR core, and in the subcritical assembly sitting on top of the graphite pedestal of the UFTR. The subcritical assembly contained some typical D_2O-moderated natural uranium lattices. In the thermal column of the UFTR, thermal neutron spectra were measured integrally using activation detectors such as ^{197}Au, ^{164}Dy, ^{175}Lu, ^{176}Lu, and ^{239}Pu. The measurements were made in a stainless steel rod, a natural uranium slab, and an Al can filled with borated water, and results were interpreted using Westcott's formulation. The effective neutron temperature and the epithermal index, $r\sqrt{T/T_0}$, were measured by using activation detectors in the UFTR core between the two fuel boxes in the North-South center line. The result of the effective neutron temperature measurement was compared with the open-beam differential spectrum and found to be in good agreement. A crystal diffraction spectrometer for differential spectrum measurements in the subcritical assembly was used. The neutron spectrum out of the subcritical assembly was measured for two different lattice pitches using Mark I and Mark V-B natural U fuel elements in D_2O. A neutron beam was extracted from the cell boundary and from the center of the cell for each of the three lattice arrangements. The results were compared with the theoretical spectra calculated with the THERMOS code and found to be in excellent agreement. Integral measurements were also made in the subcritical using ^{176}Lu and ^{197}Au detectors. The experimental activation measurements were compared with the THERMOS calculations and were also found to be in excellent agreement. Finally, the effective neutron temperature change in a unit cell for four U–D_2O

lattice arrangements was measured differentially and integrally and compared with the theoretical calculations. The results were in good agreement. (A.G.W.)

349.
24706 (AECL–2523) LATTICE MEASUREMENTS WITH 19-ELEMENT NATURAL URANIUM METAL ASSEMBLIES. PART II. RELATIVE TOTAL NEUTRON DENSITIES AND HYPERFINE ACTIVATION DISTRIBUTIONS IN A LATTICE CELL. Serdula, K. J. (Atomic Energy of Canada Ltd., Chalk River (Ontario)). Dec. 1965. 55p. Dep. mn. AECL $1.50.

Relative total neutron densities and hyperfine Mn-wire activation distributions are derived from measurements in the central cell of a triangular array of 55 19-element natural uranium metal assemblies in the ZED-2 reactor. Results are presented for lattice pitches of 20, 24, 28, and 40 cm. "Coolants" used were, (1) D_2O of moderator purity, and (2) air to simulate loss of coolant condition. Data indicated that the neutron-density depression in the cell is less upon loss of coolant. (auth)

350.
6584 (LA-DC–7248) CRITERIA AND EVALUATION FOR THE STORAGE OF FISSILE MATERIAL IN A LARGE AND VARIED REACTOR RESEARCH AND DEVELOPMENT PROGRAM. Smith, David R. (Los Alamos Scientific Lab., Univ. of California, N. Mex.). [nd]. Contract W-7405-eng-36. 13p. (SM–70-50; CONF-651103–13). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

From IAEA Symposium on Criticality Control of Fissile Materials, Stockholm.

Due to the large amounts of plutonium and uranium stored at LASL, criticality studies were made for stored arrays. Critical masses of air-spaced arrays of 10.5, 15.7, 20.9, and 26.2 Kg cylinders of 93% enriched uranium are presented. Critical masses are given for spheres with core densities reduced by the fraction ρ/ρ_0; water reflected U(93) metal, $U(93)O_2$, $U(93)F_6$, and unreflected U(93) metal. (M.O.W.)

351.
42599 (UJV–1424) SROVNANI TEORETICKEHO VYPOSTU LAPLASIANU MRIZE TEZKOVODNIHO REAKTORU S MERENIM NA KRITICKEM SOUBORU. (Comparisons of Theoretical Laplacian of Heavy-Water Reactor Lattice With the Measurements of the Critical Assembly). Sulc, P. (Ustav Jaderneho Vyzkumu, Ceskoslovenska Akademie Ved, Rez). 1966. 32p. Dep. mn.

A comparison between the calculated and measured Laplacian of a heavy water gas-cooled lattice cell with natural U fuel is given. The method of the correlation of the program PARAMETRY II is discussed together with the deficiencies of the calculational method. An estimate of the influence of incorrect values of some physical constants and geometrical characteristics on the Laplacian is made. In the conclusion some recommendations concerning the improvement of the calculational method are added and attention is devoted to necessary extension of the experimental material. (auth)

352.
12493 (AECL–2514) COMPARISON OF CALCULATED AND MEASURED CRITICAL SIZES FOR SOME ZED-2 LATTICES. Thibaudeau, J.; Critoph, E. (Atomic Energy of Canada Ltd., Chalk River (Ontario)). Nov. 1965. 30p. (CRRP–1228). Dep. AECL $1.00.

A study of some lattices of 7 and 19-element UO$_2$ clusters, which was measured in ZED-2, was undertaken to compare the predictions of different programs with measurements. In this study HEX, FLAME, and SYMMETRIZE were used. Using these programs, it was possible to assess the errors associated with most of the approximations contained in each of these programs and derive corrections for them. The measured buckling was finally compared with the bucklings obtained from the programs with these derived corrections applied. (auth)

353.
26433 CRITICAL ARRAYS WITH SUBCRITICAL UNITS OF FISSILE MATERIALS. Thomas, J. T. (Oak Ridge National Lab., Tenn.). Chem. Eng. Progr., 61: Symp. Ser. No. 60, 44-53(1965).

Experiments in support of specifications for the safe storage and transportation of fissile materials are described. The effect of hydrogenous reflectors surrounding arrays, as well as the introduction of moderating materials into arrays, are presented. Other factors influencing the number of units in a critical array are illustrated and an estimate of their magnitude is given. (auth)

354.
20840 (BNWL–193, pp 57-67) NEUTRON INTERACTION BETWEEN MULTIPLYING MEDIA SEPARATED BY VARIOUS MATERIALS. White, J. D.; Richey, C. R. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

The quantity of material necessary to effectively isolate a multiplying media is of primary concern to the design of shipping containers and formulation of storage regulations for fissionable materials. The results are given of a series of critical experiments in which the isolating properties of various materials were investigated. Those materials tested were placed at the interface between two arrays of multiplying media that comprised the core of a critical assembly. The critical length of the fuel core was then determined as a function of the test material thickness; the effective isolation thickness of a material was taken to be that thickness for which the fuel core critical length approached an asymptotic value. Comparative data were also obtained on the reflector savings associated with the various test materials, where the reflector savings is defined as the decrease in critical length due to the presence of the reflector. (M.O.W.)

355.
45955 (MLM–1340) NEUTRON MULTIPLICATION DETERMINATION OF ^{238}Pu DIOXIDE. Wolfe, R. A.; Kahle, J. B. (Mound Lab., Miamisburg, Ohio). Oct. 5, 1966. Contract AT(33-1)-Gen-53. 21p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

A neutron multiplication experiment was performed with 21 storage capsules containing high isotopic analysis plutonium dioxide. Each storage capsule contained approximately 80 grams of the ^{238}Pu isotope. The capsules were assembled in a heterogeneous array spaced four inches center-to-center and measurements were made independently in a water and in an air medium. The results indicated that no neutron multiplication was detectable in the fully assembled array in a water medium. However, a slight neutron multiplication of 1.29 existed in the fully assembled array in an air medium. (auth)

356.

26428 (AECL-2330, pp 233-50) A STUDY OF THE PHYSICAL CHARACTERISTICS OF HYDROGEN-CONTAINING REACTORS IN THE CRITICAL REGION. Yurova, L. N.; Polyakov, A. A.; Klenov, G. I.; Morosov, I. G.; Inyutin, E. I.; Sidorov, G. I. Translated from pp 194-207 of Issledovaniya po Primeneniyu Organicheskikh Teplonositelei-Zamedlitelei v Energeticheskikh Reaktorakh, Atomizdat, Moscow, 1964.

The results of an experimental comparison of organic and aqueous moderators under identical conditions on a critical stand are presented. In these studies, monoisopropylbiphenyl was used as the organic moderator, the purpose of the experiment being to determine the critical mass of uranium in systems with organic and aqueous moderators for a given construction of fuel elements. The critical stand used is described, and the distribution of thermal neutrons is analyzed. The results show that the values of the migration area for media with monoisopropylbiphenyl lie below the values for aqueous moderators by 40–70% for identical values of QH/Q_5. Measurements were also carried out at different ratios of the active zone to determine the effect of the geometry of the active zone on the critical masses. These investigations showed that in the region where $D_{equiv} \simeq 1$ when $QH/Q_5 = 200$ to 300 for monoisopropylbiphenyl and $QH/Q_5 = 300$ to 400 for aqueous moderators, the values of the critical masses are essentially independent of the geometry of the active zone. (ATD)

1967

357.

5681 (BAW-3647-1) PHYSICS VERIFICATION PROGRAM. Quarterly Technical Report No. 1, January–June 1966. (Babcock and Wilcox Co., Lynchburg, Va.). Aug. 1966. Contract AT(30-1)-3647. 61p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Measurements are described of the nuclear properties of water-moderated, slightly enriched U-fueled lattices, perturbed by boric acid and Ag–In–Cd control pins. Three critical assemblies were studied in this period. In all cases the lattices consisted of 0.475-in.-OD rods of 2.46%-enriched UO_2, arrayed on a square pitch of 0.644 in. and moderated by H_2O. Only the boric acid concentration in the moderator (and the critical dia) was changed, from zero to about 1.5 g B/l. The critical conditions, reactivity coefficients ($\Delta\rho/\Delta h$ and $\Delta\rho/\Delta N_B$), buckling, and thermal disadvantage factor are reported. Methods being developed for the measurement of ρ_{28}, the modified conversion ratio, and the epithermal neutron spectrum are described. (auth)

358.

5678 (AECL-2606) LATTICE MEASUREMENTS WITH 28-ELEMENT NATURAL UO_2 FUEL ASSEMBLIES. PART I. BUCKLINGS FOR A RANGE OF SPACINGS WITH THREE COOLANTS. Serdula, K. J. (Atomic Energy of Canada Ltd., Chalk River (Ontario)). July 1966. 95p. Dep. mn. AECL $2.00.

Bucklings derived from activation measurements in the ZED-2 reactor are given for 28-element natural UO_2 fuel assemblies. Measurements were made in D_2O-moderated triangular arrays of 55 assemblies at pitches in the range 20 to 40 cm. The three coolants used were D_2O of moderator purity, He to simulate a voided condition, and HB-40, an organic liquid. Bucklings obtained with He as coolant are greater than those for D_2O and HB–40 coolants for the pitches investigated. Macroscopic variations of the In–Cd ratio were measured with D_2O and organic coolants within the range of pitches studied. (auth)

359.

2884 EXPERIMENTAL STUDIES OF MIXED D_2O LATTICES CONTAINING ThO_2 RODS. Sokolowski, E.; Andersson, A. J. W.; Persson, R.; Wikdahl, C. E. (Aktiebolaget Atomenergi, Stockholm). 17p. (CONF-660524-23). ORAU. Gmelin, AED-CONF-66-104-18.

From 2nd International Thorium Fuel Cycle Symposium, Gatlinburg, Tenn.

Measurements were made on D_2O lattices of natural UO_2 rods, 1.2% enriched UO_2 rods, ThO_2 rods, and their admixtures. The data from the buckling, spectral index, and flux distribution measurements are presented. Conversion ratio measurements were also made. (D.L.C.)

3. CROSS SECTIONS

1956

1.
1058 KAPL-813
Knolls Atomic Power Lab., Schenectady, N. Y.
SOME ACTIVATION MEASUREMENTS IN THE INTERMEDIATE ENERGY REGION. I. H. Dearnley, H. E. Soisson, and J. E. Yunker. Sept. 24, 1952. Decl. Oct. 18, 1955. 88p. Contract W-31-109-Eng-52.

Measurements of ratios of isotopic capture cross section to U^{235} fission cross section (σ_a/σ_{f235}) in an intermediate energy region have been made for a number of isotopes. Materials were activated by photoneutrons from a 2-Mev x-ray Be block source or by neutrons from side hole No. 6 of the Argonne CP-3 reactor. The ratio of σ_a/σ_{f235} was measured for Na^{23}, Mn^{55}, As^{75}, Rb^{85}, Sr^{88}, Zr^{96}, Mo^{98}, Ru^{102} Pd^{108}, Cd^{114}, In^{115}, Sn^{120}, Sn^{122}, Sn^{124}, Sb^{123}, Ba^{138}, La^{139}, Pr^{141}, Ce^{142}, Nd^{148}, W^{186}, and Au^{197}. The values obtained were considerably lower than predicted by the statistical theory of nuclei. (auth)

2.
4350 LA-1201
Los Alamos Scientific Lab., N. Mex.
A MEASUREMENT OF THE AVERAGE FISSION CROSS SECTIONS OF Pu^{240} AND Pu^{241} IN THE FAST REACTOR NEUTRON SPECTRUM. Edward T. Jurney. Nov. 8, 1950. Decl. Dec. 7, 1955. 8p. Contract W-7405-eng-36. $0.15 (OTS).

A comparison chamber is described which was designed to be inserted into a reactor port in order to measure fission rates from neutron fluxes which had undergone a minimum of energy degradation. The cross section was inferred from a comparison of the fission counting rates from foils of known Pu^{240} enrichments with the rates from standard foil containing only Pu^{239}. Incidental to the Pu^{240} fission cross section measurement the experiment also yielded a measurement of the average fission cross section for Pu^{241}. (auth)

3.
4359 KAPL-511
Knolls Atomic Power Lab., Schenectady, N. Y.
DEPENDENCE OF α OF U-233 ON TEMPERATURE. J. B. Sampson and H. Hurwitz, Jr. August 14, 1951. Decl. Dec. 6, 1955. 19p. Contract W-31-109-eng-52. $0.20(OTS).

Two lines of reasoning are used in considering the temperature dependence of α (ratio of radiative capture to fission) in U^{235} exposed to a thermal neutron flux. An experiment involving measurement of the ratio of fissions in U^{235} to capture in U^{238} with and without a Ag shell surrounding the U foil, is analyzed. The best value indicated from this experiment is $(d\alpha/dT) \sim -(10 \pm 5) \times 10^{-5}/°C$, in contrast to Wigner's result of $-22 \times 10^{-5}/°C$ by a similar analysis. Indications from neutron velocity selector results are that $d\alpha/dT$ is more apt to be positive than negative, and possibly as large as $10 \times 10^{-5}/°C$. There is no experiment which is definitely inconsistent with the assumption $d\alpha/dT = 0$, and moreover this zero value is a compromise between the qualitative values indicated by the two lines of reasoning considered. (auth)

4.
1642 ORNL-1992
Oak Ridge National Lab., Tenn.
A DIRECT COMPARISON OF SOME NUCLEAR PROPERTIES OF U-233 AND U-235. J. T. Thomas, J. K. Fox, and Dixon Callihan. Dec. 12, 1955. 25p. Contract W-7405-eng-26.

A comparison has been made of some nuclear properties of U^{233} and U^{235} based on data obtained in a series of critical experiments. Aqueous solutions of uranyl oxyfluoride containing uranium enriched to about 90% in each of the two isotopes have been made critical in water reflected spherical reactors having diameters of 26.4 and 32.0 cm. With the hypotheses that the reported nuclear constants for U^{235} are reliably known and that the neutron leakage spectra for U^{233} and U^{235} were equal for the same water reflected critical sphere, the value of $\eta(U^{233})$ at 0.026 ev was determined to be 2.31 ± 0.03. The critical masses of the two isotopes have been measured over a limited temperature range and corresponding values of the

reactivity temperature coefficient are reported. The results are tabulated. The critical mass of U^{233} in the larger sphere, unreflected, was 1.15 kg. The delayed neutron yield of the two isotopes were compared by noting the periods resulting from the withdrawal of a boron poison from the critical spheres. It is shown that the yield from U^{233} is about one third that from U^{235}, in agreement with other determinations. (auth)

1957

5.
2739 KAPL-M-HB-3
Knolls Atomic Power Lab., Schenectady, N. Y.
MEASUREMENT OF α FOR Pu AT 20 KV. Harvey Brooks. Apr. 11, 1947. Decl. Feb. 7, 1956. 8p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

6.
12284 AECU-3527
Westinghouse Electric Corp. Research Labs., [East] Pittsburgh, Penna.
COMPILATION OF TECHNICAL REPORTS ON THE SUBJECT OF FAST NEUTRON SCATTERING. PART III. [1957]. 117p. Project NR-024-028. Sponsored by ONR and AEC under Contract N6-ORL-156, task 2. $18.30(ph OTS); $6.00(mf OTS).

Part I issued as AECU-3387. Part II issued as AECU-3402.

The papers presented are entitled: Resonance Capture Integrals as a Check on the Neutron Absorption Cross Section in Multigroup Tabulations; Multigroup Neutron Cross Sections; Cross Section Calculations for Fast Neutron Scattering; IV. Depth and Radius of the Real Potential in a Diffuse Surface Optical Model; and The Construction of Inelastic Scattering Matrices. (T.R.H.)

7.
9440 AERE-NP/R-2140
Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.
TABLE OF CONSISTENT NUCLEAR DATA FOR U^{233}, U^{235}, AND Pu^{239} FOR 2200 M/SEC NEUTRON. P. A. Egelstaff, K. W. Morton, and J. E. Sanders. Dec. 1955. 11p.

Five quantities relating to the behavior of fissile isotopes are tabulated for U^{233}, U^{235}, and Pu^{239}. These quantities are the average number of neutrons emitted per fission, the average number of neutrons emitted per neutron absorbed, σ_a, the absorption cross section, σ_f the fission cross section and the ratio $(\sigma_a - \sigma_f)/\sigma_f$. (M.H.R.)

8.
12504 KAPL-1346
Knolls Atomic Power Lab., Schenectady, N. Y.
REPORT OF THE NUCLEAR PHYSICS SECTION FOR MARCH, APRIL, MAY 1955. Decl. Mar. 14, 1957. 43p. Contract W-31-109-Eng-52. (Deleted version: AECD-3792). $0.30(OTS).

The neutron capture-to-fission ratio was measured for a 5000 ev shielded U^{235} sample and found to be, for a mean fission energy of 15 kev, 0.406. The schematic diagram of the new three-stage mass spectrometer is given. An anticoincidence scintillation spectrometer, which is being applied to the quantitative analysis of gross fission-product mixtures is described. Plans are being made to build a high-precision magnetic alpha analyzer. Pu^{239} fission and absorption cross sections and temperature coefficients of reactivity are analyzed. The average of a Breit-Wigner resonance over a Maxwellian neutron spectrum is made. An extension has been made to the theory of atomic displacements produced by irradiation in a solid, improving agreement with experiment. The effects of monoenergetic fast neutron on normal and gold-doped Ge was studied. (W.L.H.)

9.
2636 BNL-1575
Brookhaven National Lab., Upton, N. Y.
ε MEASUREMENTS. H. Kouts and G. Price. Feb. 19, 1953. Decl. May 3, 1956. 7p. $1.80(ph OTS); $1.80 (mf OTS).

The experimental method and results of fast fission factor measurements on reactor fuel rods containing U^{238} are discussed. (D.E.B.)

10.
11359 KAPL-1793
Knolls Atomic Power Lab., Schenectady, N. Y.
AN σ49(E) CURVE CONSISTENT WITH EBR-1 MEASUREMENTS AND ALSO WITH MOST FORMER MEASUREMENTS. J. B. Sampson and D. F. Molino. May 16, 1957. 18p. Contract W-31-109-Eng-52. $0.20(OTS).

A critical review is made of experimental information concerning the energy dependence of the capture-to-fission ratio, α, of Pu^{239}. A form of this dependence is obtained which is shown to be consistent with nearly all measurements, including these recently reported for EBR-1. This dependence shows a rapid decrease with energy near 100 kev, and is intermediate in form between those of several previous suggestions. (auth)

11.
713
DETERMINATION OF THE AVERAGE NUMBER OF NEUTRONS ν_{eff} EMITTED IN A SINGLE CAPTURE EVENT BY THE ISOTOPES U^{233}, U^{235} AND Pu^{239} IN THE SUPERTHERMAL REGION OF NEUTRON ENERGIES. P. E. Spivak, B. G. Erozolimsky, G. A. Dorofeev, V. N. Lavrenchik, I. E. Kutikov, and Yu. P. Dobrynin. Soviet J. Atomic Energy,

No. 3, 295-302(1956).

The variation of ν_{eff} for the isotopes U^{233}, U^{235} and Pu^{239} is measured in the superthermal region of neutron energies. For U^{233} ν_{eff} remains constant up to an energy of the order of 100 ev. For Pu^{239} ν_{eff} drops by 12% on going from the thermal spectrum to the spectrum of energies from 0.15 to 0.5 ev, and then remains constant. For U^{235} ν_{eff} remains constant on going from the thermal spectrum to the spectrum of energies from 0.15 to 0.5 ev, and then drops by 18% on going to the spectrum of energies from 8 to 130 ev. (auth)

12.
714

THE AVERAGE NUMBER OF NEUTRONS ν_{eff} EMITTED BY THE ISOTOPES U^{233}, U^{235} AND Pu^{239} ON CAPTURE OF NEUTRONS WITH ENERGIES FROM 30 TO 900 kev. P. E. Spivak, B. G. Erozolimsky, G. A. Dorofeev, V. N. Lavrenchik, I. E. Kutikov, and Yu. P. Dobrynin. Soviet J. Atomic Energy, No. 3, 303-9(1956).

Measurements are made on ν_{eff} for the isotopes U^{233}, U^{235} and Pu^{239} for neutrons with energies from 30 to 900 kev. It is discovered that in this energy region ν_{eff} increases substantially as the neutron energy increases. (auth)

1958

13.
9371

NEW DATA FOR NEUTRON CROSS SECTIONS. P. K. (K. P.). Atomnaya Energiya 3, No. 8, 169-70(1957).

On the basis of data obtained from the U.S.A., the U.S.S.R., and England, the most probable values for the neutron cross sections and criticality constants for U^{233}, U^{235}, and Pu^{239} are tabulated. (J.S.R.)

14.
4573

CAPTURE-TO-FISSION RATIOS FOR FAST NEUTRONS IN U^{235}. B. C. Diven, J. Terrell, and A. Hemmendinger (Los Alamos Scientific Lab., N. Mex.). Phys. Rev. 109, 144-50 (1958) Jan. 1.

The ratio $\alpha = \sigma_c/\sigma_f$, where σ_c is the neutron capture cross section and σ_f the neutron-induced fission cross section, has been measured for U^{235} as a function of neutron energy. A pulsed and collimated neutron beam is passed through a U^{235} sample at the center of a large liquid scintillator. Captures and fissions are detected by means of their prompt gamma rays; elastic and inelastic scattering events are ignored because of smaller pulse heights. Fissions are distinguished from captures by means of delayed pulses from the capture of thermalized fission neutrons. It is found that in the neutron energy range E_n = 0.175 to 1.0 Mev the value of α is given approximately by $\alpha = 0.190$ to $0.116E_n$. The accuracy of the determination of α is 10 to 15% in terms of the standard deviation and individual points. (auth)

15.
11786

SUBCADMIUM IN-PILE ETAS AND ABSORPTION CROSS SECTIONS OF U^{233}, Pu^{239}, AND Pu^{241}. E. R. Gaertner, M. E. Jones, D. E. McMillan, J. B. Sampson, and T. M. Snyder (General Electric Co., Schenectady, N. Y.). Nuclear Sci. and Eng. 3, 758-71(1958) June.

Measurements of two independent types were made of the reactivity effect in a thermal test reactor of samples of U^{235}, U^{233}, Pu^{239}, and Pu^{241}. From these measurements average subcadmium values of eta ($\bar{\eta}$) relative to $\bar{\eta}$ of U^{235} are obtained independently of other knowledge of the average absorption cross sections. Average absorption cross sections are also obtained from the measurements. Values of ($\bar{\eta}$) for U^{233}, Pu^{239}, and Pu^{241} are respectively 2.231 ± 0.034, 1.927 ± 0.024, and 2.213 ± 0.07. The corresponding value of η (0.0253 ev) of Pu^{239} is found to be 2.025. A presentation of the method and results are given together with a comparison with previous work. (auth)

16.
9382

EFFECTIVE REACTOR CROSS SECTIONS IN MTR FUEL ASSEMBLIES. J. Halperin and J. O. Blomeke (Oak Ridge National Lab., Tenn.) and D. A. Mrkvicka (Argonne National Lab., Chicago). Nuclear Sci. and Eng. 3, 395-402(1958) Apr.

Analyses of uranium, neptunium, and plutonium isotopes have been carried out on dissolved MTR fuel elements. Effective reactor cross sections computed from these measurements are in good agreement with differential cross-section measurements. In particular, effective cross-section values were found for U^{236} and U^{238} of 34 barns and 23 barns, respectively. (auth)

17.
6362

TEN-GROUP CALCULATED EQUILIBRIUM NEUTRON SPECTRUM AND DIFFUSION LENGTH IN NATURAL URANIUM. D. Meneghetti, H. H. Hummel, and W. B. Loewenstein (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 3, 151-60(1958) Feb.

The degradation of neutron energies in a fast reactor is largely due to inelastic scattering. In a dilute fast system (large U^{238} to U^{235} atomic ratio) the neutron spectrum is then primarily determined by a fission spectrum distribution modified by inelastic scattering in U^{238}. In this investigation a set of ten-group fast cross sections for U^{238} have been prepared with the inelastic cross sections below about 1.35 Mev based upon levels at 45, 150, and 700 kev. The inelastic transfer contributions from unknown higher levels were chosen to be consistent with the gross measurements of Bethe, Beyster, and Carter, having the three-group energy division consisting of above 1.4 Mev between 0.4 and 1.4 Mev, and below 0.4 Mev. The ten-group fast cross sections were tested by comparing the calculated equilibrium spectrum, diffusion

length, and detector responses in natural uranium with reported experimental values found in the blanket of the Zephyr reactor and in the Snell experiments. (auth)

1959

18.

9427 BNL-463(p.53)

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

NEUTRON YIELDS FOR U^{233}. R. J. Beeley. p.53 [of] THORIUM–U^{233} SYMPOSIUM, SPONSORED BY THE UNITED STATES ATOMIC ENERGY COMMISSION AT BROOKHAVEN NATIONAL LABORATORY, JANUARY 9-10, 1958. 1p.

Investigation of cross section curves to obtain average values of η over eight energy groups for U^{233} are reported. (W.L.H.)

19.

11377 CF-59-1-70

Oak Ridge National Lab., Tenn.

ABSOLUTE MEASUREMENT OF ETA BY THE MANGANESE BATH TECHNIQUE. G. deSaussure and R. L. Macklin. Jan. 20, 1959. 11p. Contract [W-7405-eng-26]. $3.30(ph), $2.40(mf) OTS.

An experiment is described for measuring η of U^{233} and estimating the various errors involved. A value for η can be obtained with a precision of 1% or better. (auth)

20.

17438

COMPARISON OF CRITICAL EXPERIMENTS FOR THE DETERMINATION OF ETA OF U^{233}. D. W. Magnuson and Reginald Gwin (Oak Ridge National Lab., Tenn.). Nuclear Sci. and Eng. 2, No. 1, Suppl., 146-8(1959) June.

The series of experiments to determine η of U^{233} with five concentrations of $U^{233}O_2(NO_3)_2$ solutions and four concentrations of $U^{235}O_2(NO_3)_2$ solutions in a 69.2-cm-diam. aluminum sphere is described. Boric acid was used as a poison in the experiments in order to vary the uranium concentration. Unpoisoned solutions in a 122-cm-diam. sphere were also made critical. Experiments and calculations are summarized. The best value of η determined was 2.295 ± 0.036. (W.D.M.)

21.

1464 ORNL-2609

Oak Ridge National Lab., Tenn.

NEUTRON PHYSICS DIVISION ANNUAL PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 1, 1958. Oct. 28, 1958. 161p. Contract W-7405-eng-26. $3.00 (OTS).

A relatively low-power reactor research facility, the Pool Critical Assembly (PCA), was installed in one end of the pool at the Bulk Shielding Facility. A description of the facility and the initial critical experiments is given. Experiments to determine the effect on reactivity of large voids in the reflector are described. A new attempt was made to calculate the energy spectrum of the gamma rays leaking from the BSF reactor by integrating the sources over the entire reactor and applying an attenuation kernel. Some recent calculations pertaining to the UO_2-SS Bulk Shielding Reactor II are presented, as well as a status summary on the project. The latest design of the Tower Shielding Reactor II (TSR-II), with its associated controls and 5 Mw water cooling system, and several studies supporting the design are presented. The values of η of U^{233} and Pu^{239} relative to the value of η of U^{235} were determined. Experiments were performed to determine the critical parameters of aqueous solutions of 93.2% U^{235} enriched uranyl fluoride contained in cylindrical annuli formed by various combinations of aluminum cylinders varying in diameter from 2 to 30 in. Critical experiments were performed with ORR and BSR fuel elements to determine safe arrays in which the elements could be handled and stored. Experiments were performed with 2.09% U^{235}-enriched uranium metal plates in a light-water-moderated and -reflected assembly. The reflecting properties of water, Styrofoam, and Plexiglas were studied in critical experiments with 6 in. thick slabs of aqueous solutions of UO_2F_2 enriched to 93% U^{235}. A series of experiments with blocks of 2% U^{235}-enriched mixtures of UF_4 and paraffin are described. Some data are presented describing the critical conditions of spherical volumes of aqueous solutions of U^{233} and U^{235}. A study of gamma emission during thermal fission of U^{235} was carried out. The slowing down of fission neutrons in water was investigated. Results were obtained for neutron diffusion measurements for beryllium at room and liquid nitrogen temperatures. A 300-kv particle accelerator was constructed for the BSR. A detailed theoretical study was made of resonance absorption of neutrons in nuclear reactions. The connection between neutron flux and slowing down density was investigated. The empirical kernel method for treating bare critical systems is discussed and utilized to predict the material buckling of aqueous U^{235} bare reactors in an effort to clarify the age discrepancy. The infinite medium multiplication factor was computed for $U^{235}/O_2F_2-H_2O$ solutions as a function of neutron energy.

22.

9259 LAMS-2255

Los Alamos Scientific Lab., N. Mex.

NEUTRON CROSS SECTIONS FOR FAST AND INTERMEDIATE NUCLEAR REACTORS. C. B. Mills. Oct. 1958. 171p. Contract W-7405-eng-36. $3.00(OTS).

A neutron transport equation which was solved numerically has been connected with multigroup neutron cross sections and critical assemblies to display the area of coverage and range of errors. The cross sections, critical assemblies, and results are tabulated. (auth)

23.

13686 AERE-R/R-2457

United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.

RECENT HARWELL OSCILLATOR DATA FOR HE THE FISSILE NUCLEI U^{235}, U^{233}, AND Pu^{239}. H. Rose, W. A. Cooper, and R. B. Tattersall. Feb. 1958. 9p.

The pile oscillator technique was employed for a careful investigation of the reactivity effects produced by each of the three major fissile nuclei U^{235}, U^{233}, and Pu^{239} in two of Harwell's thermal reactors, Dimple and Gleep. The results are compared with predictions based upon the Westcott recommendations for effective thermal cross sections. In addition, the oscillator data are used to obtain fairly precise ratios of η for comparison with present world average data. The results yield for thermal neutrons at 20°C the relative values: $\eta(U^{235})$: $\eta(U^{233})$: $\eta(Pu^{239}) = 1.000$: 1.074 ± 0.015: 0.974 ± 0.015. (auth)

24.

5000 BNL-402

Brookhaven National Lab., Upton, N. Y.

THE VALUE OF η_{nat}. R. Sher. June 25, 1956. 6p. $1.80(ph OTS); $1.80(mf OTS).

The resonance escape probability was measured in the BNL reactor and in about 18 uranium–water lattices. From these measurements and previous measurements of f, ϵ, M^2, and B^2, values of η_{nat} are inferred. The results are discussed. (W.D.M.)

25.

9253 APEX-467

Nuclear Development Corp. of America, White Plains, N. Y.

SOME NEUTRON CROSS SECTIONS FOR MULTIGROUP CALCULATIONS. N. Tralli. June 30, 1958. 212p. For General Electric Co. Aircraft Nuclear Propulsion Dept. Contracts AF33(600)-38062 and AT(11-1)-171. Subcontract AT-70. $3.50(OTS).

Chapters 1 through 4 deal, respectively, with neutron diffusion data, neutron sensor data, inelastic scattering data, and photon activation data. Chapter 5 consists of a list of the references to the literature cited in the first four chapters. The neutron diffusion data consist of the absorption, scattering, transport, and slowing-down cross sections. The neutron sensor data are concerned with the activation cross section σ_{n,β^-} for the production of negative beta particles. The inelastic scattering data consist of the neutron inelastic scattering cross section, the spectrum of the inelastically scattered neutrons and the inelastic slowing-down cross section. The photon activation data consist of the neutron cross section for photon production $\sigma_{n,\gamma}$, the total photon energy release and the photon energy distribution spectrum. All cross sections are reported as obtained and as group-averaged and digitalized cross sections. The group-averaged and digitalized cross sections are defined. The groups over which the cross sections were to be averaged, the averaging procedure, and the digitalization procedure were specified so that the results obtained would be readily usable in the GE reactor calculation procedure. In addition, thermal averages of the neutron diffusion cross sections are reported. (auth)

26.

11364 TNCC(US)-39

[Tripartite Nuclear Cross-Sections Committee.]

REPORTS BEARING ON U^{235} THERMAL FISSION CROSS-SECTION DISCREPANCY. Mar. 1958. 27p. $4.80(ph), $2.70(mf) OTS.

Originally issued as LA-511 and LA-512.

The gamma-ray activity of two samples irradiated in the Clinton and Hanford piles and of the barium and cesium extracted from them was compared with the activities extracted from a sample in which the number of fissions was determined by monitoring during the neutron exposure. The results obtained show good internal consistency and indicate that $(8.9 \pm 0.3) \times 10^{17}$ and $(4.8 \pm 0.25) \times 10^{18}$ fissions, respectively, had occurred in the two samples. These results are compared with mass spectrometer data obtained by Williams and Yuster on the same samples to find the ratio of the capture and fission cross sections of U^{235}. (auth)

27.

934 AERE-R/R-2620

United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.

NEUTRON FLUX MEASUREMENTS IN THE CORE ON LIDO. J. W. Weale, H. R. Mck. Hyder, A. Green, E. D. Jones, C. J. Kenward, and P. J. Oram. July 1958. 47p. $1.26(BIS).

The fast, epithermal, and thermal neutron flux distributions through the core of LIDO have been measured using absolutely calibrated thin foils. By comparing the reaction rates of sodium, manganese, indium, and gold, absolute values have been inferred for the epithermal and thermal fluxes, and some revised cross sections are suggested for manganese and indium. (auth)

1960

28.

13217 WAPD-BT-17(p.51-6)

Westinghouse Electric Corp. Bettis Atomic Power Lab., Pittsburgh.

MULTIGROUP CROSS SECTIONS FROM DETAILED NUCLEAR DATA. H. J. Amster. 6p.

The cross sections of U^{238} have recently been tabulated with more than usual detail. They are used here for making up a fifty-four group experimental library for the MUFT

Code. The procedure is presented as an example of how one could, in principle, account for fine details in all nuclear data. (auth)

29.

13462

NEUTRON ENERGY SPECTRUM MEASUREMENTS IN UNMODERATED ASSEMBLIES. G. J. Fischer (Argonne National Lab., Lemont, Ill.). Nuclear Sci. and Eng. 7, 355-62(1960) Apr.

The neutron energy spectrum at the center of the dilute fast core of the coupled fast-thermal reactor ZPR-V was studied by use of fission chambers having electrodes quantitatively electrodeposited with U^{234}, U^{235}, U^{236} and U^{238}. Atomic fission ratios found with these four isotopes determine a four-group neutron energy spectrum which can readily be measured as a function of position in the core by use of suitable drive units. The same fission chamber procedure was used to study the equilibrium neutron energy spectrum in a natural uranium exponential column at Los Alamos. The results of measurements in these two spectra are shown and compared with theoretical predictions. The ZPR-V results are also compared to an analysis of this spectrum made by use of nuclear emulsions for the range 0.2 to 2.2 Mev. (auth)

30.

26168 CF-60-4-12

Oak Ridge National Lab., Tenn.

CRITICAL EXPERIMENTS FOR REACTOR PHYSICS STUDIES. R. Gwin and D. W. Magnuson. Sept. 16, 1960. 64p. OTS.

The thermal value of $\bar{\eta}$ for U^{233} and U^{235} was determined in a series of experiments involving direct comparison of the critical parameters of unreflected homogeneous aqueous solutions of the two isotopes. Auxiliary experiments establishing limits of error, testing certain aspects of the theoretical model employed, and experimentally determining the parameters in the critical equation were performed. Experiments performed with 27-in.-diameter and 48-in.-diameter spheres and 5-ft-diameter and 9-ft-diameter cylinders yielded consistent values of η. Measurements of the nonleakage probability in cylindrical geometry gave values consistent with those predicted by a two-group model in which the theoretical value of the age was used. Within the experimental error no differences were found in the ages of fission neutrons for U^{233} and U^{235}. The average thermal values of eta determined are $\bar{\eta}$ for U^{233}, 2.284 ± 0.016 and $\bar{\eta}$ for U^{235}, 2.074 ± 0.016. The 2200 m/sec values are the same since the g-factors for eta are unity. (auth)

31.

18428 BNL-607

Brookhaven National Lab., Upton, N. Y.

NEUTRON CROSS SECTION EVALUATION GROUP NEWSLETTER NO. 1, JUNE 1960. Rudolph Sher and Sophie Moore. 8p. OTS.

The discrepancy in the $Be^9(n,2n)$ cross section as measured by Fischer and Levin & Cranberg was partially resolved in the 2.6- to 4.1-Mev region by new measurements of the nonelastic neutron scattering. The cross section for differential elastic neutron scattering for Be^9 was determined at 2.6 to 6.0 Mev, together with that for $Be^9(n,\alpha)$ at 3.9 to 8.6 Mev. B^{10} was determined to be present in natural B to the extent of 19.8 ± 0.1%, and the cross sections for $B^{10}(n,\alpha)$ and natural B (absorption) were found to be 3840 ± 10 and 762 ± 3 barns, respectively. The cross sections for $B^{10}(n,\alpha)$ and $B^{10}(n,2\alpha)$ were determined in the Mev range. Cross sections are reported for $O^{16}(r.,\alpha)$ at 5 to 7.34 Mev, O^{16} differential elastic neutron scattering at 3.0 and 6.0 Mev, $Ar^{38}(n,\alpha)$ at 1.3 to 5.5 Mev, and $Ar^{40}(n,\alpha)$ at 5.75 to 8.94 Mev. Neutron capture cross sections in the kev energy range are reported for Nb, Mo, Rb, Pd, Ag, Cd, In, Sn, W, Pt, and Au. Thermal neutron total cross sections at 0.02 to 0.20 ev are reported for U^{233}, U^{235}, Pu^{240}, U^{234}, and I^{129}, with the results for 0.02 to 0.04 ev being given at 2200 m/sec. Neutron multiplication data are given for U^{233} and Pu^{239}. (D.L.C.)

1961

32.

22882 THE INTERACTION OF 14 MeV NEUTRONS WITH URANIUM. K. W. Allen, P. Bomyer and J. L. Perkin (Atomic Weapons Research Establishment, Aldermaston, Berks, Eng.). Reactor Sci. and Technol., J. Nuclear Energy, Pts. A and B, 14: 100-6(May 1961).

The multiplication of 14-Mev neutrons in uranium shells was studied. The measurements lead to a value of η, the average number of neutrons produced per inelastic collision of a 14 Mev neutron, of 3.30 ± 0.15. Neutron interactions in a thick uranium shell have also been investigated and effective cross sections for the reactions $U^{238}(n,f)$, $U^{235}(n,f)$, and $U^{238}(n,\gamma)$ have been obtained for the inelastic neutron spectrum in the shell. The value of η, combined with other nuclear data, leads to the following cross section data for U^{238} for neutrons in the energy range 13.4 to 14.8 Mev: $1.1 \geq \sigma_{n,3n} \geq 0.8$ barn and $0 \leq \sigma_{n,n'} \leq 0.3$ barn. (auth)

33.

24295 (TID-13165) DETERMINATION OF α FOR U^{235} IN THE INTERMEDIATE ENERGY SPECTRUM BY ANALYSIS OF URANIUM FUEL FROM A Be MODERATED INTERMEDIATE SPECTRUM LATTICE. E. B. Fehr, E. C. Hansen, D. G. Miller, and F. M. Rourke (Knolls Atomic Power Lab., Schenectady, N. Y.). [1961?]. Contract W-31-109-Eng-52. 9p.

The value for the capture-to-fission ratio α for U^{235} for intermediate-energy neutrons was determined by analysis of depleted uranium fuel from a Be-moderated and -reflected lattice. The radiochemical method of analysis is outlined. The results are compared with the 1947 KAPL results and with the work of S. Oleska. (D.L.C.)

34.

6766 (ORNL-3016(p.81)) DETERMINATION OF THE THERMAL VALUE OF η OF U^{233} AND U^{235} BY DIRECT COMPARISON OF CRITICAL PARAMETERS OF AQUEOUS SOLUTIONS. R. Gwin and D. W. Magnuson (Oak Ridge National Lab., Tenn.).

An experimental determination of the thermal value of η of U^{233} and U^{235}, based on the direct comparison of the critical parameters of dilute aqueous solutions of the uranyl nitrates of U^{233} and U^{235}, was completed. The average thermal values of η obtained are 2.284 ± 0.015 for U^{233} and 2.074 ± 0.015 for U^{235}. The 2200-m/sec values are the same, since the g factor for η is unity in each case. (auth)

35.

2195 MANGANESE BATH MEASUREMENTS OF η OF U^{233} AND U^{235}. R. L. Macklin, G. deSaussure, J. D. Kington, and W. S. Lyon (Oak Ridge National Lab., Tenn.). Nuclear Sci. and Eng. 8, 210-20(1960) Sept.

The absolute thermal value of η for U^{233} and U^{235} was measured directly by a method of total absorption which involves relative counting of manganese bath activations and some minor corrections. A thermal neutron beam (defined by cadmium difference) is introduced in the center of a one-meter-diameter sphere filled with a dilute solution of manganous sulfate in water. The beam is first made to activate the bath directly, then it is totally absorbed in the fissionable sample whose fission neutrons then activate the bath. The ratio of the two activities is equal to η except for small corrections. The results obtained for η corrected to 2200 m/sec were, for U^{233}, 2.296 ± 0.010; and for U^{235}, 2.077 ± 0.010. (auth)

36.

13654 (BNL-653) 'NEUTRON CROSS SECTION EVALUATION GROUP NEWSLETTER NO. 3, FEBRUARY 1961. Rudolph Sher and Sophie Moore (Brookhaven National Lab., Upton, N. Y.). 8p.

Capture-to-fission ratios in U^{233}, U^{235}, and Pu^{239} are given as a function of neutron energies from 30 kev to 1 Mev. Excitation functions of fast-neutron-induced reactions are given for $Th^{232}(n,2n)Th^{231}$ from 8.4 to 15.1 Mev. Reduced neutron widths, Γ_n^0, are tabulated for calcium from 254 to 595 kev and lead-208 from 352 to 718 kev, as determined from the neutron total cross section resonance parameters. Measurements of neutron radiative cross sections for 49 nuclides are tabulated relative to indium at 30, 65, and 167 kev. (For preceding newsletter see BNL-634.) (B.O.G.)

1962

37.

2772 (AAEC/E-59(Suppl.1)) FAST NEUTRON FLUX AND SPECTRUM MEASUREMENTS IN HIFAR. FIRST SUPPLEMENT—AVERAGE FISSION SPECTRUM CROSS SECTION FOR $Ni^{58}(n,p)Co^{58}$. J. W. Boldeman and K. P. Nicholson (Australia. Atomic Energy Commission Research Establishment, Lucas Heights, New South Wales). Oct. 1961. 3p.

The $Ni^{58}(n,p)Co^{58}$ reaction cross section in a fission spectrum was re-measured and found to be 102 ± 3 millibarns. (auth)

38.

26532 THE CORRELATION OF INTEGRAL EXPERIMENTS AND HIGH-ENERGY CROSS-SECTIONS. J. Chernick, H. C. Honeck, P. Michael, S. O. Moore, and G. Srikantiah (Brookhaven National Lab., Upton, N. Y.). p.435-43 of "Physics of Fast and Intermediate Reactors. Vol. I." Vienna, International Atomic Energy Agency, 1962. (In English) (BNL-5604; BNL-5892)

The correlation of integral experiments and high-energy cross sections is discussed. The importance of integral data where cross-section measurements are inadequate is pointed out. The sensitivity of estimates of fast fission of U^{238} to inelastic cross sections and energy degradation in the Mev energy range is shown by comparison of integral data with Monte Carlo calculations. It is shown that the Snell experiment is a sensitive index to the absolute values of inelastic cross sections above 1.4 Mev. The results of attempts by the Brookhaven Cross-Section Evaluation Group to reconcile measurements of inelastic cross sections of U^{238} are given. Other areas where integral data and critical experiments can be used to reduce computational uncertainties are the fast effect in beryllium, and η of U^{233} at intermediate energies. Critical experiments can reduce the present uncertainty in Be (n,2n) cross-sections and in intermediate energy values of η_{23}. (auth)

39.

29630 (ANL-6570) k CALCULATIONS FOR 22 ZPR-III FAST REACTOR ASSEMBLIES USING ANL CROSS-SECTION SET 635. W. G. Davey (Argonne National Lab., Idaho Falls, Idaho). May 1962. Contract W-31-109-eng-38. 21p.

The Yiftah, Okrent, and Moldauer cross-section set (ANL Set 135) was modified to include new measurements of ν and α for U^{235}, and to include, in an approximate manner, Hummel, Rago, and Meneghetti's corrections for flux depression at resonance energies in Al and stainless steel. This modified set, ANL Set 635, was used to compute values of k for 22 ZPR-III assemblies of widely varying composition. The DSN neutron transport code was used in spherical geometry and the S4 approximation; shape factors were used to convert from cylindrical to spherical geometry. Seventeen of the calculated values of k lie within ±1% of a mean value of 1.003, and the remaining 5 lie within ±2%. In terms of prediction of critical

mass, it appears that the procedure used here can achieve an accuracy of 5% to 10% for a wide range of U^{235}-fueled assemblies. (auth)

40.

3862 (ORNL-3193(p.177-87)) THE MEASUREMENT OF α AS A FUNCTION OF ENERGY. G. deSaussure, L. W. Weston, et al. (Oak Ridge National Lab., Tenn.).

Experiments leading toward a precise measurement of α, the capture cross section to fission cross section ratio, as a function of neutron energy are completed. Determinations of α are discussed at 30 and 65 kev, and over the range from 1 ev to 150 kev for the principal fissionable isotopes. A fission chamber is designed and tested, and various problems involving background radiation and counting techniques are investigated. Electronic circuitry is also designed, built, and tested. (auth)

41.

14080 CAPTURE TO FISSION RATIOS FOR FAST NEUTRONS IN U^{233}, U^{235}, AND Pu^{239}. B. C. Diven and J. C. Hopkins (Los Alamos Scientific Lab., New Mex.). p.407-12 of "Neutron Time-of-Flight Methods." Brussels, European Atomic Energy Community, 1961.

The ratio of neutron capture to fission cross sections was measured for U^{233}, U^{235}, and Pu^{239} at 9 incident neutron energies from 30 to 1000 kev. A pulsed and collimated neutron beam is passed through the target at the center of a large cadmium-loaded liquid scintillator. Captures and fissions are detected from their prompt γ; scattering is biased out since corresponding prompt pulses are small. Fission neutrons are thermalized in the scintillating solution and provide delayed pulses that identify a fission event. Capture events are not followed by delayed pulses from the scintillator. Corrections are applied for the fissions not followed by delayed neutron pulses and for the effect of backgrounds. This experiment yields values of α to an accuracy of 1 or 2%. (auth)

42.

12326 THE MEASUREMENT OF ETA AND OTHER NUCLEAR PROPERTIES OF U^{233} AND U^{235} IN CRITICAL AQUEOUS SOLUTIONS. R. Gwin and D. W. Magnuson (Oak Ridge National Lab., Tenn.). Nuclear Sci. and Eng., 12: 364-80(Mar. 1962).

The thermal value of eta for U^{233} and U^{235} is determined in experiments on unreflected homogeneous aqueous solutions of the two isotopes. These experiments also yield a value for the neutron age and the limiting concentrations of the fissile isotope in the aqueous solutions for infinite volumes. Auxiliary experiments, establishing limits of error, testing certain aspects of the theoretical model employed, and experimentally determining the parameters in the critical equation, are also performed. Experiments performed with 27-in.- and 48-in.-diam spheres, and 5-ft- and 9-ft-diam cylinders yield consistent values of eta. Measurements of the nonleakage probability in cylindrical geometry give values consistent with those predicted by a two-group model in which the theoretical value of the age is used. Within the experimental error no differences are found in the ages of fission neutrons for U^{233} and U^{235}. The average thermal values of eta determined are 2.292 ± 0.015 for U^{233} and 2.076 ± 0.015 for U^{235}. The 2200 meters/sec values are the same since the g-factors for eta are unity. The value of the neutron age to the indium resonance energy for U^{235} fission neutrons in water is found to be 25.6 ± 1.3 cm^2. The minimum U^{233} and U^{235} critical densities for these nitrate solutions are found to be 11.25 ± 0.10 and 12.30 ± 0.10 gm/liter for U^{233} and U^{235}, respectively. (auth)

43.

12501 DETERMINATION OF ETA BY COMPARISON OF $\overline{\eta\sigma}$ FOR U^{233} AND Pu^{239} WITH $\overline{\eta\sigma a}$ FOR U^{235} IN A FLUX TRAP CRITICAL ASSEMBLY. R. Gwin and D. W. Magnuson (Oak Ridge National Lab., Tenn.). Nuclear Sci. and Eng., 12: 359-63(Mar. 1962).

The values of eta for U^{233} and Pu^{239} were determined by a reactivity coefficient measurement. An aqueous solution of each isotope was introduced axially into a critical cylindrical annular flux trap reactor, and the resulting reactivity change was measured by period determinations. From these data the ratios $[\overline{\eta\sigma}(U^{233})/\overline{\eta\sigma}(U^{235})]$ and $[\overline{\eta\sigma}(Pu^{239})/\overline{\eta\sigma}(U^{235})]$ were obtained. Using measured values of eta for U^{235} and the absorption cross sections in this ratio, the thermal values of 2.317 ± 0.040 for eta of U^{233} and 2.032 ± 0.053 for eta of Pu^{239} were obtained. Correction to a neutron velocity of 2200 meters/sec by using the appropriate g-factor gives a value of 2.317 ± 0.040 for eta of U^{233} and 2.082 ± 0.054 for eta of Pu^{239}. (auth)

44.

4680 (LAMS-2543) SIX AND SIXTEEN GROUP CROSS SECTIONS FOR FAST AND INTERMEDIATE CRITICAL ASSEMBLIES. Gordon E. Hansen and William H. Roach (Los Alamos Scientific Lab., N. Mex.). Nov. 1961. Contract W-7405-Eng-36. 72p.

Six-group neutron cross sections are listed for the more common fissionable isotopes for study of fast neutron critical assemblies and sixteen group cross sections of the more common reactor material for study of intermediate neutron critical assemblies. Data sources and averaging scheme used for the development of these multigroup parameters are also given. (auth)

45.

12498 NEUTRON CAPTURE TO FISSION RATIOS IN U^{233}, U^{235}, Pu^{239}. J. C. Hopkins and B. C. Diven (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng., 12: 169-77(Feb. 1962).

The ratio of neutron capture to fission cross sections, α, was measured for U^{233}, U^{235}, and Pu^{239} at 9 incident neutron energies from 30 to 1000 kev. A pulsed and collimated neutron beam is passed through a target placed at the center of a large, cadmium-loaded, liquid scintil-

lator. Capture and fission events are detected by means of their prompt gamma rays; elastic and inelastic scattering events are discarded because of their smaller pulse height. Fission is identified by the delayed pulses produced by capture in the scintillator of the fission neutrons. Corrections are applied for the fission events not followed by delayed neutron pulses and for the effect of background counts. This procedure yields values of $1 + \alpha$ to an accuracy of 1 or 2%. (auth)

46.

26510 MEASUREMENTS OF α FOR U^{233}, U^{235} AND Pu^{239} FOR MONOENERGETIC NEUTRONS. J. C. Hopkins and B. C. Diven (Los Alamos Scientific Lab., N. Mex.). p.111-17 of "Physics of Fast and Intermediate Reactors. Vol. I." Vienna, International Atomic Energy Agency, 1962. (TID-12896). (In English)

The ratio of neutron capture to fission cross sections, α, was measured for U^{233}, U^{235}, and Pu^{239} at 30 to 1000 kev. A pulsed and collimated neutron beam is passed through the target at the center of a large cadmium-loaded liquid scintillator. Captures and fissions are detected by their prompt gamma rays; elastic and inelastic scattering events are ignored because of their smaller pulse heights. Fission events are identified by the neutrons that accompany the fission process. These neutrons are thermalized in the scintillating solution and provide delayed pulses which characterize a fission event. Capture events are not followed by delayed pulses from the scintillator. Corrections are applied for the fission events not followed by delayed-neutron pulses and for the effect of background counts. The capture, capture-plus-fission, and background events are recorded simultaneously. This procedure yields values of $1 + \alpha$ to an accuracy of 1 or 2%. (auth)

47.

4937 (ANL-6466) EFFECT OF RESONANCE SCATTERING ON CRITICALITY CALCULATIONS OF FAST ASSEMBLIES. D. Meneghetti (Argonne National Lab., Ill.). Dec. 1961. Contract W-31-109-Eng-38. 13p.

Critical masses of representative ZPR-III fast assemblies containing resonance scatterers are calculated using the SNG transport code. THE IBM-704 ELMOE code of Hummel and Rago was used to evaluate group transport and group elastic-transfer cross sections in the cores in conjunction with the 16-group cross section set of Yiftah, Okrent, and Moldauer. By using hundreds of neutron energy groups and the detailed elastic scattering matrices for the resonance scatterers, ELMOE carries out a fundamental mode analysis. It thereby obtains material buckling and the detailed fine structure flux dependence upon energy. It then re-evaluates the gross group cross sections for transport and elastic transfer. Consideration of the resonance scattering effects caused by aluminum and stainless steel increase the calculated critical masses of ZPR-III assemblies 23, 31, and 32 by about 15 kg, 21 kg, and 21 kg, respectively. Corresponding reactivity decreases are about −1%, −0.7%, and −2% $k_{effective}$, respectively. (auth)

48.

26541 THE SENSITIVITY OF CALCULATED CRITICAL MASSES OF SMALL FAST SYSTEMS TO CHANGES IN THE U^{235} AND U^{238} NEUTRON-SCATTERING DATA. E. D. Pendlebury (Atomic Weapons Research Establishment, Aldermaston, Berks, Eng.). p.97-110 of "Physics of Fast and Intermediate Reactors. Vol. II." Vienna, International Atomic Energy Agency, 1962. (In English)

The sensitivity of the calculated critical masses of simple systems to changes in the basic neutron scattering data were investigated. The systems considered are spheres of 29% U^{235} and 93½% U^{235}, both bare, and reflected by thick natural uranium. The calculations were carried out using the Carlson S_n method with 4 energy groups, and the percentage changes in the calculated critical masses of the different systems, due to specified changes in the aspects of the neutron scattering data, were obtained. The results are presented and discussed with particular reference to the adjustment of the basic data to agree with experimental critical sizes. (auth)

49.

33788 (BAW-1244) RESONANCE ABSORPTION IN U^{238} METAL AND OXIDE RODS. Final Report. W. G. Pettus, M. N. Baldwin, and C. Samuel (Babcock and Wilcox Co. Atomic Energy Div., Lynchburg, Va.). Apr. 1962. Work performed under United States–Euratom Joint Research and Development Program. Contract AT(30-1)-2578. 103p. (EURAEC-396)

The effective resonance integral was measured for U^{238} metal and oxide rods of various diameters. Three different experimental methods—activation, static reactivity, and oscillation—were used, and the following consistent results were obtained: U^{238}: $RI = 2.4 + 26.2\sqrt{S/M}$ and $U^{238}O_2$: $RI = 0.8 + 28.2\sqrt{S/M}$. The Doppler broadening effect was measured up to 600°C for U^{238} and up to 1000°C for $U^{238}O_2$. The results are in good agreement with the previous measurements of Hellstrand. Measurements of the Dancoff effect for both H_2O- and D_2O-moderated lattices are consistent with the Bell approximation. (auth)

50.

12768 THE POTENTIAL OF PLUTONIUM AS A FUEL IN NEAR-THERMAL CONVERTER REACTORS. Karl H. Puechl (Nuclear Materials and Equipment Corp., Apollo, Penna.). Nuclear Sci. and Eng., 12: 135-50(Feb. 1962). (NUMEC-P-26)

The potential of plutonium as a fuel in near-thermal converter reactors is investigated. Over certain ranges of fuel loading and/or moderation, it is shown that the effective absorption cross section (averaged over the entire neutron spectrum) of Pu^{240} decreases with fuel burnup; i.e., decreases with the associated softening of the neutron spectrum. The plutonium, therefore, behaves as a self-stabilizing or self-compensating fuel with the decrease in Pu^{240} cross section balancing fissionable material burnup and fission product build-up. Thereby long core lives are attainable with nominal shim control requirements. The

strong neutron temperature dependence of the effective Pu^{240} absorption cross section also results in a highly negative temperature coefficient of reactivity and thereby in the feasibility of spectral shift shim control. Economics evaluation indicates that fuel cycle costs of between 1.5 and 2.5 mills/kw-hr may be attainable with these plutonium fueled systems. (auth)

51.

26519 GROUP CROSS-SECTIONS FOR FAST REACTORS. P. F. Zweifel and G. L. Ball (Atomic Power Development Associates, Inc., Detroit). p.189-205 of "Physics of Fast and Intermediate Reactors. Vol. I." Vienna, International Atomic Energy Agency, 1962. (In English)

Multi-group-diffusion equations are given, and the correct form of the group cross sections is discussed. The average transport cross section may be written to a certain approximation in terms of an average mean free path. The calculation is lengthy because it is not amenable to expression in terms of elemental averages; inequalities are proved which simplify the averaging procedure required. The use of the same set of group-averaged cross sections for all fast reactors is invalid if the spectra in different reactors are dissimilar and if the cross-sections vary rapidly over the group, conditions which frequently hold. An iteration procedure is described by which the correct averages are found; it is used to determine the sensitivity of reactor calculations to spectral effects. In transport calculations such as S_n, averages must be made over angle and energy. Since the flux is non-separable in angle and energy, care is necessary to avoid erroneous results. The S_n equation is studied in terms of a simple model, and a criterion is derived which may prove useful in determining the importance of angular nonseparability in reactor calculations. A consistency relation among group-diffusion coefficients, slowing-down power and absorption cross-sections is derived from neutron-conservation arguments. A definition of group absorption cross section in terms of effective resonance integrals is not correct; it must be modified according to the type of multi-group scheme used. (auth)

1963

52.

19264 THE RATIO OF NEUTRON CAPTURE TO FISSION FOR URANIUM-235. M. J. Cabell and L. J. Slee (Atomic Energy Research Establishment, Harwell, Berks, Eng.). J. Inorg. Nucl. Chem., 24: 1493-1500(Dec. 1962). (AERE-3986)

Alpha (the ratio of neutron capture to fission) for U^{235} was measured at nine different points along a flux scanning tube of the reactor PLUTO. Experimental values for α varied from 0.178 to 0.189. Correcting all the results to 2200 m/sec neutrons gave a mean value for α_0 of 0.1725 ± 0.0034 (standard error), a result in excellent agreement with present recommended values. (auth)

53.

28277 THE RATIO OF THERMAL NEUTRON CAPTURE TO FISSION FOR ^{239}Pu. M. J. Cabell and L. J. Slee (Atomic Energy Research Establishment, Harwell, Berks, Eng.). J. Inorg. Nucl. Chem., 25: 607-14(June 1963). (AERE-R-4173).

Alpha, the ratio of neutron capture to fission, for Pu^{239} was measured at seven different points along a flux scanning tube of the PLUTO reactor. Highly-enriched Pu^{240} samples were also irradiated to make allowance for the destruction of this nuclide. Experimental values of α for Pu^{239} varied from 0.458 to 0.470. Correcting all results to 2200 m/sec neutrons gave a mean value for α_0 of 0.370 ± 0.008, a result that is in excellent agreement with present recommended values, but does not include uncertainties in the g and s values for neutron capture and fission. (auth)

54.

702 (ANL-6617) A CRITICAL COMPARISON OF MEASURED AND CALCULATED FISSION RATIOS FOR ZPR-III ASSEMBLIES. W. G. Davey (Argonne National Lab., Idaho Falls, Idaho). Sept. 1962. Contract W-31-109-eng-38. 35p.

A critical comparison of measured and calculated central fission ratios for 18 ZPR-III fast reactor assemblies was made with the object of examining the accuracy of computation of spectra and of the fission cross sections used. This comparison uses fission ratios measured with Kirn absolute fission chambers and computed with ANL cross section Set 635. The Kirn chambers and experimental technique are described. It is shown that ratios measured with threshold detectors must be corrected for the effects of inelastic scattering in the chamber walls. Possible sources of error in the experimental technique are discussed, and experimental evidence for the validity of the method is presented. The derivation of ANL Set 635 is described. It is shown that Set 635 is a modification of the Yiftah, Okrent, and Moldauer ANL Set 135 and that the central spectra and fission ratios calculated with the two sets are generally similar. The U^{234} and U^{236} fission cross sections are not given in Sets 135 and 635; these were taken from ANL Set 179. The measured and calculated fission ratios obtained with U^{233}, U^{234}, U^{235}, U^{236}, U^{238}, Pu^{239}, and Pu^{240} were compared, both to search for trends which might occur with progressive changes in spectra, and also to determine the accuracy of prediction of ratios. It was found that the calculated relative fission rates of Pu^{239} and U^{233} are within about ±1½% of the measured values, and the calculated rates for Pu^{240}, U^{236}, and U^{238} relative to either Pu^{239} or U^{233} are within ±3½ to ±5% of the measured values. However, calculated fission rates for U^{235} and U^{234} are about 6% low and 8% high, respectively, relative to those of the other five isotopes. (auth)

55.

12773 (ORNL-3360(p.51-63)) THE MEASUREMENT OF α AS A FUNCTION OF ENERGY. G. deSaussure, L. W. Weston, et al. (Oak Ridge National Lab., Tenn.).

The ratio, α, of the capture cross section to the fission

cross section of U^{235} was measured at neutron energies of 30 and 65 kev. The results, $\alpha = 0.372 \pm 0.026$ at 30 ± 8 kev and $\alpha = 0.315 \pm 0.06$ at 64 ± 20 kev, are in agreement with previously measured values. The capture cross section of U^{238} was also determined—0.531 barn at 30 kev and 0.340 barn at 64 kev, values which are in agreement with published data. (auth)

56.

24637 (WAPD-BT-28(p.21-6)) USE OF GADOLINIUM SEPTA IN SLAB CORES. J. J. Kepes and J. A. Mitchell (Westinghouse Electric Corp. Bettis Atomic Power Lab., Pittsburgh). (WAPD-T-1451)

The nuclear characteristics of gadolinium in septa form were predicted and compared to experiment in three critical assemblies. These cores were designed to provide different spectral environments for sheets of 4.3 wt % gadolinium in zirconium. Two diffusion theory models were examined. A variational method predicted the k_{eff} to within +0.3% and −1.4%, whereas an iterative treatment was 1% to 2% low. (auth)

57.

19569 (HW-75007(Paper 14)) **THE PHYSICS OF PLUTONIUM IN FAST REACTORS.** D. Okrent (Argonne National Lab., Ill.) and F. W. Thalgott (Argonne National Lab., Idaho Falls, Idaho). 33p.

A review of the Pu behavior physics associated with medium and large fast reactors is presented. The relative reactivity worth and breeding potential of the various Pu isotopes are deduced from examination of pertinent cross sections. This information is used as a basis for examination of a broad range of fast reactors. Other discussion is concerned with Doppler effect, Na reactivity coefficient, selected kinetics, and reactor safety. (J.R.D.)

58.

22709 (GEMP-173) NINE AND SIXTEEN GROUP CROSS SECTIONS FOR REACTOR ANALYSIS. C. S. Robertson and E. M. Benson (General Electric Co. Flight Propulsion Lab. Dept., Cincinnati). Jan. 7, 1963. Contract AT(40-1)-2847. 35p.

Nine and sixteen group cross sections for use in transport theory codes are listed. Elements included are H, Be, B, C, O, Al, Ti, Cr, Mn, Fe, Ni, Co, Y, Zr, Nb, Mo, Eu, Gd, Ta, W, Re, Th, U, and Pu. A discussion of the format and the method used to process these is included. (auth)

1964

59.

17287 (NP-13666) **GRUPPOVYE KONSTANTY BYSTRYKH I PROMEZHUTOCHNYKH NEITRONOV DLYA RASCHETA YADERNYKH REAKTOROV. (Group Constants of Fast and Intermediate Neutrons for the Designing of Nuclear Reactors).** L. P. Abagyan, N. O. Bozazyants, I. I. Bondarenko, and M. N. Nikolaev (U.S.S.R. Sovet Ministrov. Gosudarstvennyi Komitet po Ispol'zovaniyu Atomnoi Energii). 1962. 243p.

Multigroup constants for fast and intermediate neutrons are compiled, and new data on neutron interactions with nuclei are included. The principle of utilizing multigroup constants, selection of energy groups, averaging of macroscopic cross sections, averaging effective cross sections, considerations of resonance effects, and determination of moderator cross section are discussed. A review is given on fission cross sections mean numbers of secondary neutrons, spectra of fission neutrons, capture cross sections and elastic and inelastic scattering. Tables of group constants and fission spectra for reactor materials are included. 578 references. (R.V.J.)

60.

33011 (A/CONF.28/P/259) PHYSICS OF FAST REACTORS. R. Avery (Argonne National Lab., Ill.), H. H. Hummel, R. N. Hwang, D. Meneghetti, P. A. Moldauer, A. B. Smith, P. Greebler, and J. B. Nims. 17p.

Prepared for the United Nations Third International Conference on the Peaceful Uses of Atomic Energy, 1964.

Developments in methods used to study the physics of fast reactors are reviewed. The status of experimentally obtained cross section data and of cross section theory is considered. Some representative values and the methods used to obtain multigroup constants based on the latest microscopic data are presented and described. Experimental results for a representative set of ZPR-III assemblies were compared with calculations based on these latest multigroup constants. The sodium void coefficient and the Doppler effect are discussed. (M.C.G.)

61.

24852 MEASUREMENTS OF BUCKLING AND RELATIVE REACTION RATES IN SOME PLUTONIUM–GRAPHITE ASSEMBLIES. D. H. Carter, W. G. Clarke, C. Hunt, J. Marshall, D. B. McCulloch, J. E. Sanders, and C. R. Symons (Atomic Energy Establishment, Winfrith, Dorset, Eng.). J. Nucl. Energy, Pt. A & B, 18: 105-24 (Mar. 1964). (AEEW-R-201)

The materials bucklings of a series of plutonium–aluminum alloy-fueled, graphite-moderated assemblies were measured, the carbon/plutonium ratio ranging from 14,520 to 2420. Fission ratios of ^{239}Pu, ^{235}U, and ^{233}U in the assemblies were obtained using small fission chambers. On one of the systems, experiments were extended up to a temperature of 370°C. The experimental data were compared with the predictions of a 43-group diffusion-theory code. Agreement between predicted and measured

bucklings is good over the whole range of composition, within the limits of experimental error, which are equivalent to an uncertainty of about ±2 per cent in reactivity. The spectrum-sensitive $^{239}Pu/^{235}U$ fission ratios were 6 to 9 per cent higher than those calculated from the multigroup spectra. Comparison with further results from an assembly loaded with ^{235}U in similar configuration, leads to a ratio of the 2200 m/sec η-values of ^{239}Pu and ^{235}U of 1.02 ± 0.02. (auth)

62.

1024 (TID-19594) MEASUREMENT OF α, THE RATIO OF THE NEUTRON CAPTURE CROSS SECTION TO THE NEUTRON FISSION CROSS SECTION, FOR U^{235} IN THE ENERGY REGION FROM 4 ev TO 2 kev. G. de Saussure, L. W. Weston, R. Gwin (Oak Ridge National Lab., Tenn.); J. E. Russell, and R. W. Hockenbury (Rensselaer Polytechnic Inst., Troy, N. Y.). [1962]. Contract AT(30-3)-328. 56p.

The ratio (α) of the neutron capture to the neutron fission cross section for U^{235} was measured for neutron energies from 4 ev to 2 kev using a pulsed neutron source. A multiplate fission chamber was placed in the center of a large liquid scintillator. The gamma rays emitted by the U of this chamber, following a neutron absorption, were detected with high efficiency by the scintillator; fission fragments were detected by the fission chamber. After various background and efficiency corrections, α was obtained as the ratio of the count rate of the scintillator in anticoincidence with the fission chamber to the count rate of the scintillator in coincidence with the fission chamber. The α's obtained were used to compute η from 4 to 24 ev, in good agreement with direct measurements. (auth)

63.

36786 (UCRL-7827) A NEW DERIVATION OF MULTIGROUP CROSS SECTIONS FOR BeO- AND GRAPHITE-MODERATED SYSTEMS. R. J. Doyás and E. H. Canfield (California. Univ., Livermore. Lawrence Radiation Lab.). Apr. 28, 1964. Contract W-7405-eng-48. 75p.

Multigroup (18-group) diffusion equation cross sections are rederived using updated numerical techniques and recent basic cross-section measurements. The basic series of Spade (BeO-moderated) and Snoopy (graphite-moderated) critical assemblies are calculated using these group constants, and the results are compared with those from earlier compilations. (auth)

64.

28564 (CONF-446-5) A MEASUREMENT OF α FOR U-235 IN THE EPITHERMAL REGION. L. J. Esch, F. Feiner, J. L. Mewherter, A. F. Heitcamp, and H. M. Eiland (Knolls Atomic Power Lab., Schenectady, N. Y.). [1963]. Contract [W-31-109-eng-52]. 12p.

From the American Nuclear Society 10th Annual Meeting, Philadelphia, June 1964.

The accurate integral measurement of α, the capture-to-fission ratio, for ^{235}U was experimentally determined for samples containing only 0.095 ppm ^{236}U. (R.E.U.)

65.

36582 DISTRIBUTION OF THE RATIO OF CAPTURE TO FISSION CROSS SECTION FOR ^{239}Pu ACCORDING TO HEIGHT IN THE BR-5 REACTOR. V. I. Ivanov, N. N. Krot, and G. N. Smirenkin. At. Energ. (USSR), 16: 497-500(June 1964). (In Russian)

A study was made of the distribution of the capture to fission ratio σ_γ/σ_f, for ^{239}Pu according to height in the BR-5. The distribution of capture reactions was measured by determining ^{240}Pu as a function of the spontaneous fission rate in plutonium samples irradiated by an integral flux of 10^{21} to 10^{22} n/cm^2. Almost isotopically pure ^{239}Pu (^{240}Pu content ≈5 × 10^{-3}%) was used as the initial material. Values of β as a function of increasing distance from the reactor center increased from 0.1 to 0.8. Data corresponding to an equilibrium neutron spectrum in the active zone and in an external region of the reflector agree with results of measurements of σ_γ and σ_f for monoenergetic neutrons. (tr-auth)

66.

4525 (BNL-7349) TOTAL NEUTRON AVERAGE CROSS SECTIONS IN THE kev REGION AND THE OPTICAL MODEL. A. P. Jain, R. E. Chrien, J. A. Moore, and H. Palevsky (Brookhaven National Lab., Upton, N. Y.). [Sept. 10, 1963]. Contract [AT(30-2)-Gen-16]. 4p. (CONF-327-3)

From Topical Conference on Compounds Nuclear States, Gatlinburg, Tenn., Oct. 1963.

The average total neutron cross sections for Nb, Mo, Rh, Ag, Cd, and In are measured at 10 to 100 kev, and the values obtained are used to calculate the optical-model parameters of the nuclei. The neutron energies used fall in the region of the P-wave giant resonances in these nuclei. (T.F.H.)

67.

25249 METHODS FOR THE DETERMINATION OF $_{28}\rho$ AND δ^{28} BASED ON CHEMICAL SEPARATION OF Np^{239} AND Mo^{99} FROM URANIUM AND FISSION PRODUCTS. S.O. Larvin (Nuclear Research Center "Democritus," Athens). p.161-79 of "Exponential and Critical Experiments. Vol. II." Vienna, International Atomic Energy Agency, 1964.

Methods are presented for the measurement of ρ^{28} and δ^{28}. The methods for ρ^{28}, the ratio of epicadmium to subcadmium capture of neutrons in ^{238}U, is based on the chemical separation of neptunium from uranium and fission products in two fuel samples, one irradiated in the lattice under study and one in a pure thermal neutron flux. The irradiations are normalized by means of auxiliary dysprosium foils. A description of the separation procedure and measuring technique is presented together with some experimental values found for D_2O-moderated lattices studied in the NORA zero-power reactor. The method for δ^{28}, the ratio of the fission rate in ^{238}U to that in ^{235}U, is based on the chemical separation of ^{99}Mo from uranium foils of different enrichment, irradiated in the lattice under investigation. A description of the irradiation and separation procedure is presented together with experimental results obtained in D_2O-moderated lattices studied in JEEPNIK, a miniature exponential facility. (auth)

68.

38013 (A/CONF.28/P/368) EXPERIMENTAL STUDIES ON FAST-NEUTRON REACTOR PHYSICS. A. I. Leipunskii, I. I. Bondarenko, O. D. Kazachkovskii, et al. (U.S.S.R.). May 1964. 24p.

An experimental study was made of some problems of fast-neutron reactor physics. The studies continue a series of experiments on fast reactor physics that has been carried out at the Physical-Energy Institute since 1950. The following problems are considered: refinement and development of multigroup cross-section sets for nuclear reactors calculation; studies of neutron propagation in large blocks of material performed on BR-I reactor are described; the BFS reactor for large fast-neutron critical assembly investigations; and radiative-capture cross-section measurements on fissionable isotopes for neutron spectra of the BR-5 core and reflector. (auth)

69.

30957 (AEEW-R-318) MEASUREMENTS OF BUCKLING AND Pu239/U235 RELATIVE REACTION RATES IN AN UNDERMODERATED GRAPHITE ASSEMBLY FUELLED WITH HIGHLY ENRICHED U235 AND Pu239. E. J. Maunders (United Kingdom Atomic Energy Authority. Reactor Group. Atomic Energy Establishment, Winfrith, Dorset, England). Apr. 1964. 39p.

As part of a general program of work on intermediate reactor assemblies, a series of measurements were made in a subcritical plutonium-fueled, graphite moderated system in which S, the ratio of moderator to fissile atoms, was 920, and in a ^{235}U fueled system in which S was 440, to determine the flux fine-structure, the material buckling, and the relative $^{239}Pu/^{235}U$ reaction rates. Predictions of material buckling and the shape of the neutron energy spectrum were made using a Parks scattering kernel, a Schofield scattering kernel, and a Free Gas Model together with what is basically OCUSOL (See ANL 5800) group-averaged nuclear data, and also with the point tabulated data from the Winfrith Nuclear Data File. Unfortunately, the latter does not give a good representation of the cross section between 4 and 60 ev. The $^{239}Pu/^{235}U$ relative reaction rates predicted by the Parks model using OCUSOL group-averaged data are in good agreement with the measured values in both assemblies although the values predicted by the other models using this data are not very different. With the Winfrith Nuclear Data File, discrepancies of as much as 20% arise between predicted and observed values of this reaction rate ratio. With OCUSOL data, satisfactory agreement is obtained between predicted and observed values of B_m^2 for the ^{235}U fueled assembly, but for the ^{239}Pu fueled assembly better agreement was found when the η value for the ^{239}Pu was increased by 25% at energies above 10 ev. For both assemblies using the Winfrith Nuclear Data File leads to values of B_m^2 up to 6% lower than those predicted using OCUSOL data. (auth)

70.

30707 (AECL-1965) A DETERMINATION OF THE RATIO OF CAPTURE-TO-FISSION CROSS SECTIONS OF U^{235}. A. Okazaki, M. Lounsbury, R. W. Durham, and I. H. Crocker (Atomic Energy of Canada Ltd., Chalk River, Ont.). Apr. 1964. 14p.

The capture-to-fission cross section ratio, α, for ^{235}U was determined for two samples of ^{235}U–Al alloy irradiated to 2.03 and 2.26 × 10^{21} neutrons/cm^2 in the NRU reactor. The changes in isotopic and total uranium content with irradiation were determined by chemical and mass spectrometric analysis. The value of α_0 obtained from these measurements is 0.1718 ± 0.0006, which is 2.5% lower than the weighted mean of other recent measurements. (auth)

71.

38652 GRAPHITE EXPONENTIAL ASSEMBLY. T. Rzeszot, E. Warda, H. Markovic, and S. Takac. Rev. Phys., Acad. Rep. Populaire Roumaine, 6: Suppl., 415-22 (1961).

An exponential assembly is described, and buckling and diffusion length measurements with it are reported. The device consists of a graphite cube 240 cm on a side with 100 fuel channels 60 × 60 mm, lattice constant 240 mm, and 26 measurement channels 30 × 30 mm. The cube is covered with cadmium foil. There is 2088 kg of natural U in rods 27.2 × 300 mm in a can 1-mm thick. The value of L obtained is 48.4 cm. From the buckling experiment the dimension of a cubic critical reactor is 734 ± 14 cm. (T.R.H.)

72.

33030 APPLICATION OF THE PULSED-NEUTRON TECHNIQUE TO CRITICAL AND SUBCRITICAL ASSEMBLIES. M. Sagot, H. Tellier, R. Jacquemart, and J. Kremser (CEN, Saclay, France). p.163-89 of "Exponential and Critical Experiments. Vol. III." Vienna, International Atomic Energy Agency, 1964. (In French)

The pulsed-neutron technique was used in different critical experiments. This technique is particularly interesting since it can be used without altering the reactor structures. Its application requires only a time selector and pulsed sources which, because of their low density, are introduced on a small scale into the reactor or its reflector. The technique can be used to measure the neutron lifetime and calibrate control rods quickly and in perfectly safe conditions. Two different Rubeole cores were studied in a critical experiment with a beryllium–oxide moderator and slightly enriched uranium–oxide fuel, and the negative importance function was measured in different arrays: (a) reduction in the volume of the nonreflected core; (b) introduction of cadmium control rods; (c) unloading of peripheral fuel; and (d) fuel replacement. Other experiments reported (Alize: light water and uranium enriched to 90%; Aquilon: heavy water and natural uranium; Marius: graphite and natural uranium) show that the technique is applicable to natural-

uranium reactors. Using a nonportable neutron generator, the same technique was employed to measure graphite diffusion parameters. The buckling range was between 7 m^{-2} and 155 m^{-2} and the following results were obtained: diffusion coefficient Do = $(2.19 \pm 0.03)10^5$ cm^2 sec^{-1}; cooling coefficient C = $(37.9 \pm 4)10^5$ cm^4 sec^{-1}; and graphite anisotropy $D_{\perp\perp}/D_{\perp} = 1.017 \pm 0.008$. These results are compared with those obtained by others. (auth)

73.

15476 SOME PROBLEMS IN THE CALCULATION OF HYDROGENOUS SYSTEMS. L. K. Shishkov and A. F. Piskunkov. Nekotorye Vopr. Inzh. Fiz., No. 4, 14-33(1963). (In Russian)

Starting with the Boltzmann equation in the diffusion approximation, the scattering by hydrogen was treated separately from the scattering by all other nuclei, which were considered as "heavy" for this purpose. The resulting equations were then put into the multigroup (in lethargy) form including inelastic scattering and using the fission spectrum as the source. The final results treated hydrogenous media in either the P_1 approximation or in a "transport" approximation. The connection with the "moment method" was developed. The criticality problem is discussed formally using the matrix method. A discussion of the extrapolation length as a lethargy group quantity is presented. Results of fifteen-group calculations are presented in both approximations for the critical radius of an assembly based on a core consisting of a UO_2F_2–water solution with enriched uranium and compared with the experimentally determined dependence of the radius on hydrogen concentration. Similarly in a nine-group calculation the results from the two approximations for the age to 1.4 ev from fission in water–iron mixtures of various compositions are compared with experimental results. Twelve tables giving the group cross sections for several elements, especially hydrogen, are included in the appendix. (TTT)

74.

21598 (IA-899) INELASTIC SCATTERING MATRICES FOR FAST REACTOR CALCULATIONS. G. Szwarcbaum, M. Sieger, and S. Yiftah (Israel. Atomic Energy Commission. Soreq Research Establishment, Rehovoth). Dec. 1963. 31p.

By minimizing the χ^2 function, improved values are obtained for the parameters of three formulas for nuclear temperatures (the formulas of Weisskopf, Bethe and Lang-Le Couteur). A comparison of the improved formulas shows that all fit the experimental data equally well and yield lower vaues for the nuclear temperature. On the basis of the improved Weisskopf formula, new multigroup inelastic cross sections are calculated and used in the calculation of several fast assemblies. The calculated critical masses are generally in slightly better agreement with experiment. An explanation of this effect is given. (auth)

75.

38307 (A/CONF.28/P/511) INELASTIC SCATTERING OF HIGH-ENERGY NEUTRONS IN FAST REACTORS. G. Szwarcbaum, M. Sieger, and S. Yiftah (Israel. Atomic Energy Commission. Soreq Research Establishment, Rehovoth). May 1964. 20p.

For neutrons above 1 or 2 Mev, inelastic scattering is described to a good approximation by an evaporation formula. It is necessary to find fitting values for the nuclear temperature at various energies and for a wide range of nuclides. The formulas for the nuclear temperature, θ, are obtained as derivatives of the formulas for the average level spacing, D. Usually, first parameters appearing in the formula for D are fitted to the experimental data, and then the formula for the nuclear temperature is derived. It is advantageous first to derive the formula for θ, in which only one parameter is involved. The fit with experiment is done for parameters in the formulas resulting from the three most common formulas for D—those of Weisskopf, Bethe and Lang-Le-Couteur. This is done by minimizing the χ^2 function. Comparison of the improved formulas shows that all describe equally well the experimental data and yield lower values for the nuclear temperature. On the basis of the improved Weisskopf formula, new multigroup inelastic cross sections are derived and used in the calculation of several fast assemblies. The calculated critical masses are generally in slightly better agreement with experiment than previous calculations. (auth)

76.

4532 (EANDC-33U(p.64-86)) NEUTRON CAPTURE IN U^{238} AND THE RATIO OF CAPTURE TO FISSION IN U^{235}. L. W. Weston, G. de Saussure, and R. Gwin (Oak Ridge National Lab., Tenn.).

Large liquid scintillator tanks have been applied to the measurement of the capture cross section of U^{238} at 30 and 64 kev, and the measurement of the ratio of capture to fission in U^{235}. The ratio of capture to fission in U^{235} is studied over a wide neutron energy range (4 ev to 700 kev) by two different methods. (auth)

77.

34886 RATIO OF CAPTURE-TO-FISSION IN U^{235} AT kev NEUTRON ENERGIES. L. W. Weston, G. deSaussure, and R. Gwin (Oak Ridge National Lab., Tenn.). Nucl. Sci. Eng., 20: 80-7(Sept. 1964).

The ratio of the neutron-capture cross section to the fission cross section (α) for ^{235}U was measured for incident neutron energies from 12 to 690 kev by a large gadolinium-loaded liquid-scintillator technique. Additional measurements at 30 and 64 kev were made by a method using a liquid scintillator and a fission chamber. The experimental values of α can be approximately described by a linear decrease from 0.374 at 10 kev to 0.177 at 210 kev, followed by a less rapid linear decrease to 0.095 at 700 kev. The results of these experiments are consistent and in reasonable agreement with other reported values of α in this energy range. (auth)

78.

38249 (A/CONF.23/P/510) NUCLEAR CROSS SECTIONS FOR FAST REACTORS. S. Yiftah and M. Sieger (Israel. Atomic Energy Commission, Tel-Aviv). May 1964. 16p.

The 16-group YOM cross section set for fast reactor analysis is revised. A revised 16-group cross section set takes account of: new information on neutron angular distributions, modified inelastic scattering matrices necessitated by different nuclear temperatures and statistical model calculations, modified capture cross sections of the fissile nuclides, modified ν values of uranium-235, and modified cross section values. An extension of the 16-group set to a 20-group set by adding 4 groups in the 2–14 Mev range is presented. Thus, taking into account the (n,p), (n,α) and (n,2n) reactions. A detailed study to 14 Mev will facilitate analysis of experimental results obtained with pulsed neutron sources. A finer group structure above the uranium-238 fission threshold enables study of the fine structure of the flux in this region. Another extension of the YOM set, below the present boundary of 0.5 kev, is in progress. This will enable spectrum analysis of neutrons in blankets and in Doppler calculations. (auth)

79.

21168 (GEMP-223) NINE- AND SIXTEEN-GROUP CROSS SECTIONS FOR TUNGSTEN. J. W. Zwick and A. Prince (General Electric Co. Advanced Technology Services, Cincinnati). June 3, 1963. Contract AT(40-1)-2847. 31p.

Revised neutron cross sections of natural tungsten are presented in the nine- and sixteen-group formats for use in transport theory codes S-VIII, S-X, S-XI, and TDC reactor nuclear calculations. The method employed in processing of the cross sections is outlined. The source of data are summarized, and a tabulation of cross section point-values for use in other calculations is included. (auth)

80.

21170 (GEMP-279) NEUTRON CROSS SECTIONS FOR TANTALUM IN THE NINE-ENERGY-GROUP FORMAT. J. W. Zwick (General Electric Co. Advanced Technology Services, Cincinnati). Apr. 6, 1964. Contract AT(40-1)-2847. 5p.

Neutron microscopic cross sections for tantalum are presented in the nine-group format. (auth)

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81.

35440 TIME-OF-FLIGHT MEASUREMENTS OF NEUTRON SPECTRA FROM THE FISSION OF U^{235}, U^{238}, AND Pu^{239}. Barnard, E.; Ferguson, A. T. G.; McMurray, W. R.; Van Heerden, I. J. (Atomic Energy Research Establishment, Harwell, Berks, Eng.). Nucl. Phys., 71: 228-40(1965).

The energy spectra in the range 100 kev to 9 Mev of the neutrons emitted in the fission of ^{238}U by 2.086 and 4.908 Mev incident neutrons were measured with a time-of-flight technique in conjunction with a multi-plate fission chamber. The experimental arrangement is described and possible errors discussed. The fission neutron spectra obtained from the slow neutron fission of ^{235}U and ^{239}Pu were also measured. These results and other published data are summarized and compared with the predictions of statistical theory. (auth)

82.

29329 MEASUREMENT OF FAST-FISSION RATIOS IN NATURAL URANIUM. C. B. Bigham (Atomic Energy of Canada Ltd., Chalk River, Ont.). Trans. Am. Nucl. Soc., 8: 268-9(May 1965).

83.

18734 (CEA-R-2455) MESURES D'INDICES DE SPECTRE DANS LES MILIEUX MULTIPLICATEURS HOMOGENES. (Measurements of Spectral Indices in Homogeneous Multiplying Media). Jean-G. Bruna, Jean-Paul Brunet, Christian Clouet d'Orval, Jacques Kremser, Jean Moret-Bailly, Henry Tellier, and Philippe Verriere (Commissariat à l'Energie Atomique, Saclay (France). Centre d'Études Nucléaires). Sept. 1964. 57p. Dep.(mn).

Methods for computation of spectra in light water are available, and it is interesting to carry out at the same time experimental studies of simple media (such as solutions of fissionable salts) which allow quite direct comparisons with computed values. The spectral indices measurements were made with two small fission chambers, one containing deposited plutonium, the other deposited uranium-235. Their response, when neutron spectrum is modified, allows the epithermal part of the flux to be studied. The media studied with these chambers are fissionable solutions (of plutonium or 90 percent enriched uranium) which were made critical in bare cylindrical geometry in the ALECTO reactor. If the ratio of the chambers is normalized to unity in a Maxwell spectrum, then the noted variation of the ratio of the counts Pu chamber/U-235 chamber reaches 1.4 in the range of the studied concentrations. (auth)

84.

10300 CAPTURE-TO-FISSION RATIOS OF U^{233}, U^{235}, U^{238} AND Pu^{239} IN EBR-I, MARK III. Carl E. Crouthamel (Argonne National Lab., Ill.), Donald C. Stupegia, Peter Kafalas, and Charles M. Stevens. Nucl. Sci. Eng., 21: 179-85(Feb. 1965).

In order to compare the breeding capabilities of the major nuclear fuels in the spectrum of a fast-breeder reactor, integral measurements were made for the ratio of their capture and fission cross sections in the third loading of the First Experimental Breeder Reactor (EBR-I, Mark III). The capture-to-fission ratio was determined as a function of position in the reactor for ^{233}U, ^{235}U, and ^{239}Pu. In addition, for ^{233}U the ratio of (n,2n) and fission cross sections was determined. Further, for ^{238}U the following cross-section ratios were determined: $\sigma_{ny}(^{238}U)/\sigma_f(X)$ and $\sigma_f(^{238}U)/\sigma_f(X)$, where $\sigma_f(X)$ refers to the fission cross sections of ^{233}U, ^{235}U, and ^{239}Pu. The capture-to-fission ratio results for the three primary fissile species were compared with calculations based upon 16-group neutron diffusion theory using two different sets of monoenergetic neutron cross sections, and the agreement is good. The

present data show that of the three major fissile species, ^{239}Pu has the highest value of $\eta - 1$, the maximum number of neutrons available for breeding, for each fissile nucleus consumed. (auth)

85.

47944 (ANL-7034) THE RATIO OF ^{238}U CAPTURE AND ^{235}U FISSION CROSS SECTIONS IN FAST REACTORS. Davey, William G. (Argonne National Lab., Idaho Falls, Idaho). May 1965. Contract W-31-109-eng-38. 31p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

An activation technique was developed for the measurement of the ratio of the capture cross section of ^{238}U and the fission cross section of ^{235}U in zero-energy fast reactors. This work was initiated because of the long-standing discrepancy between calculated values of this ratio and radiochemically measured values. The new technique is a direct counting method which does not involve chemical separation in any way. Measurements were made in four ZPR-3 fast reactor assemblies, two with hard spectra and two with soft spectra. In all four cases the measured ratio was slightly (4%) higher, on the average, than the calculated value. This is in strong contrast with the past radiochemical measurements in ZPR-3 assemblies, which gave values 16% less than calculation. The present measurements, therefore, support the general correctness of the calculated ratio, and hence indicate that there are no gross errors in the assumed average microscopic values of the ^{238}U capture cross section and the ^{235}U fission cross section. (auth)

86.

41749 MEASUREMENT OF THE ^{235}U NEUTRON CAPTURE-TO-FISSION RATION, α, FOR INCIDENT NEUTRON ENERGIES FROM 3.25 ev TO 1.8 kev. deSaussure, G. (Oak Ridge National Lab., Tenn.); Weston, L. W.; Gwin, R.; Russell, J. E.; Hockenbury, R. W. Nucl. Sci. Eng., 23: 45-57(Sept. 1965). (ORNL-TM-1041).

The ratio of the neutron capture cross section to the fission cross section, α, for ^{235}U was measured for incident neutron energies from 3.25 ev to 1.8 kev. A pulsed and collimated neutron beam was passed through a ^{235}U fission chamber placed at the center of a large liquid scintillator, and both capture and fission events in the chamber were detected in the scintillator by means of their prompt gamma rays. A fission event was distinguished from a capture event by a coincidence of the scintillator signal with a signal with a signal from the fission chamber. The values of α obtained, after various efficiency and background corrections were applied, are in good agreement with data derived from other experiments. (auth)

87.

39775 (AEEW-M-513) NEUTRON CROSS-SECTIONS OF Be9 IN THE ENERGY RANGE 1 Mev TO 15 Mev. Doherty, G. (Atomic Energy Establishment, Winfrith (England)). Mar. 1965. 13p. Dep.; BIS, $0.40; HMSO, 2s.

Self consistent neutron cross sections for the energy range 1 to 15 Mev are tabulated. The cross section for the (n,2n) reaction in the important energy region 2-4 Mev is still not well determined but the shape was adjusted to give agreement with a recent integral measurement. The angular distribution of elastically scattered neutrons, and the energy distribution of neutrons from the (n,2n) reaction are not reviewed. (auth)

88.

3387 A MEASUREMENT OF ALPHA FOR URANIUM-233 AND URANIUM-235 IN THE EPITHERMAL REGION. L. J. Esch and F. Feiner (Knolls Atomic Power Lab., Schenectady, N. Y.). Trans. Am. Nucl. Soc., 7: 272(Nov. 1964).

89.

29321 CROSS SECTIONS FOR REACTOR DESIGN. N. C. Francis (Knolls Atomic Power Lab., Schenectady, N. Y.). Trans. Am. Nucl. Soc., 8: 214(May 1965).

90.

17233 MULTILEVEL CROSS SECTIONS AND REACTOR CRITICALITY. M. Goldsmith and N. R. Candelore (Westinghouse Electric Corp., Pittsburgh). Nucl. Sci. Eng., 21: 576-8(Apr. 1965). (WAPD-T-1719)

The effect of using the approximate Breit-Wigner single-level formula for representing resonances on criticality calculations was studied by considering the fission/absorption ratio in a pure ^{235}U fuel element. Absorption cross sections and fission/absorption ratios obtained with the single-level formula were compared with results obtained with a multilevel formula. (D.C.W.)

91.

27417 (GA-5944) NEUTRON CROSS SECTIONS FOR U^{235}. G. D. Joanou and M. K. Drake (General Atomic, San Diego, Calif.). Dec. 10, 1964. Contract SNPC-27. 141p. (NASA-CR-54263; N65-16203). $4.00(CFSTI).

Experimental and analytical neutron cross sections for ^{235}U are tabulated. The data are incorporated into the GAM-II slowing-down program and the GATHER-II thermalization program. The neutron energy range studied is that from 1 Mev to 15 Mev. The results are also presented in graphical form. (T.F.H.)

92.

35609 (BNWL-84) CRUNCH — A GROUP-COLLAPSING CODE FOR FAST REACTOR ANALYSIS. Little, W. W., Jr.; Hardie, R. W. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Apr. 29, 1965. Contract AT(45-1)-1830. 52p. Dep.(mn); $3.00(cy), 1(mn) CFSTI.

A group collapsing code for condensing fast reactor cross section data is described. The code uses the fundamental mode spectrum as a weighting function in computing few-group cross sections. The input cross sections can have up to 26 energy groups with 10 downscattering elements. The collapsed set can contain any number of groups ($1 \leq n \leq$ number in input set). A FORTRAN IV source deck listing, input information, and a sample case are given. (auth)

93.

29540 EFFECTIVE FISSION-RATIO MEASUREMENTS IN A SERIES OF FAST REACTORS. G. W. Main, F. H. Helm, and H. H. Meister (Argonne National Lab., Ill.). Trans. Am. Nucl. Soc., 8: 242-3(May 1965).

94.

3540 CALCULATIONS OF ZPR-III FAST ASSEMBLIES USING A TWENTY-SIX-GROUP CROSS-SECTION SET. D. Meneghetti and J. R. White (Argonne National Lab., Ill.). Trans. Am. Nucl. Soc., 7: 237-8(Nov. 1964).

95.

29200 EVALUATION OF GROUP CROSS SECTIONS FOR LIGHT REFLECTORS OF FAST REACTORS. David Meneghetti (Argonne National Lab., Ill.). Trans. Am. Nucl. Soc., 8: 247-8(May 1965).

96.

14484 (AECL-2148) A DETERMINATION OF THE RATIO OF CAPTURE-TO-FISSION CROSS SECTION OF U^{233}. A. Okazaki, M. Lousbury, and R. W. Durham (Atomic Energy of Canada Ltd., Chalk River, Ont.). Dec. 1964. 14p. Dep.; $0.50(AECL).

The neutron capture to fission cross section ratio (α) for ^{233}U has been determined for two samples of $^{233}U-Al$ alloy irradiated to 2.29 and 2.90 × 10^{21} neutrons/cm^2 in the NRU reactor. The changes in isotopic composition and total uranium content with irradiation were determined by mass spectrometric and isotopic dilution analyses. The average value of α obtained was 0.0940 ± 0.0004 for the irradiation in a neutron spectrum having r = 0.022 ± 0.004 and T = 40°C. The 2200 m/sec value of α_0 = 0.00939 ± 0.0009 was derived from the measured α. (auth)

97.

24043 TWENTY-SIX GROUP CROSS SECTIONS. D. M. O'Shea, H. H. Hummel, W. B. Loewenstein, and D. Okrent. 10p. (CONF-654-74)

From American Nuclear Society Meeting, San Francisco, Nov.-Dec. 1964.

The generation of a set of 26-group neutron cross sections for analysis of fast reactors is discussed. The evaluation of cross section data for individual inelastic level excitation, total inelastic cross sections, fission, nuclear temperature, and capture is outlined. Cross sections for both reactor fuel and structural materials are included. (D.C.W.)

3374 TWENTY-SIX-GROUP CROSS SECTIONS. D. M. O'Shea, H. H. Hummel, W. B. Loewenstein, and D. Okrent (Argonne National Lab., Ill.). Trans. Am. Nucl. Soc., 7: 242-3(Nov. 1964).

98.

3391 INVESTIGATION OF THE RADIATIVE CAPTURE OF NEUTRONS WITH ENERGIES UP TO 50 kev. Yu. P. Popov. Tr. Fiz. Inst., Akad. Nauk SSSR, 24: 111-68 (1964).

The energy dependence of the radiative capture cross sections for Cu, Fe, Pb, and Ni was determined, and the measured resonance integral of the capture cross section was compared with calculated and previous experimental values. The average cross sections for Br, Rb, Nb, Mo, ^{98}Mo, ^{100}Mo, Rh, Ag, In, I, Cs, W, Ir, and Au were obtained; and the energy dependence of the radiative capture cross sections for these nuclei were plotted. Resonance integrals were tabulated. The dependence of the average cross section for 30-kev neutrons on the atomic weight of odd-z nuclei was determined, and the average cross section was analyzed on the basis of the interaction of p-neutrons with nuclei. The capture cross section was calculated; the experimental and theoretical values were compared. The average cross sections were also analyzed. (D.C.W.)

99.

47595 (TID-22141, pp 50-66) NEUTRON CROSS SECTIONS. (Rensselaer Polytechnic Inst., Troy, N. Y.).

The capture-to-fission cross section ratio of ^{235}U was measured, and preliminary values of the neutron resonance parameters for Hf isotopes were obtained by analysis of total cross section measurements on Hf. Neutron scattering and total cross sections of ^{238}Pu, ^{233}U, and ^{235}U were measured from 10 to 100 ev. The analysis of the cross section data for enriched samples of ^{182}W, ^{183}W, ^{184}W, and ^{186}W was almost completed; resonances were identified and assigned by isotopes from 100 ev to 4 kev using a computer program. (D.C.W.)

100.

45244 (AEEW-M-472) A NINETEEN GROUP EXTENSION TO THE YIFTAH, OKRENT AND MOLDAUER CROSS-SECTION SET. Rowlands, J. L.; Wardleworth, D. (Atomic Energy Establishment, Winfrith (England)). Mar. 1965. 30p. Dep. mn.

A nineteen group extension to the first fourteen groups of the Yiftah, Okrent and Moldauer cross-section set was produced. The thirty-three groups cover the energy range 0.414 ev to 10 Mev, the highest energy group being of unit lethargy width and the remainder of half-lethargy width. Resonance self-shielding factors at 300°K for ^{235}U, ^{238}U, ^{239}Pu and ^{240}Pu are included. They are cast as a simple formula relating the shielded cross section in each group g to the cross section at infinite dilution via σ_s, the total cross section per atom, and two parameters, α_g and β_g. (auth)

101.

45250 (BNL-325(2nd Ed.)(Suppl.2)(Vol.III)) NEUTRON CROSS SECTIONS. VOLUME III. Z = 88 to 98. Stehn, John R.; Goldberg, Murrey D.; Wiener-Chasman Renate; Mughabghab, Said F.; Magurno, Benjamin A.; May, Victoria M. (Brookhaven National Lab., Upton, N. Y.). Feb. 1965. Contract AT(30-2)-Gen-16. 282p. Dep. mn; CFSTI $3.00 cy, $1.50 mn.

Neutron cross sections are tabulated and shown graphically for elements having Z = 88 to 98. (T.F.H.)

102.

10480 (CRRP-1185) THE EFFECT OF NEW DATA ON REACTOR POISONING BY NON-SATURATING FISSION PRODUCTS. W. H. Walker (Atomic Energy of Canada Ltd., Chalk River, Ont.). Nov. 1964. 32p. (AECL-2111). Dep.; $1.00(AECL).

An Addendum to CRRP-913 and CRRP-1090.

New pseudo-fission product yields have been calculated for thermal-neutron fission of U-233, U-235, Pu-239, and Pu-241, and fast-neutron fission of U-238, based on new input cross section and fission yield data. The input values were obtained from a review of published yields and cross sections, including data published since 1960. The effect on reactor poisoning of three β-active nuclides with rela-

tively short half-lives are demonstrated. In the case where these are secondary to a non-saturating primary fission product (Pm-148 and Pm-148m) a simplified decay scheme has been used to include their contribution in the pseudo-fission product calculations. (auth)

103.

29247 (JINR-P-2024) VZAIMODEISTVIE NEITRONOV S YADRAMI URANA-235 V OBLASTI ENERGII 0.002–30 Kev. (Reactions of Neutrons with Uranium-235 Nuclei at 0.002 to 30 kev). Thih-ti Wang, Yung-chang Wang, E. Dermendzhiev, and Yu. V. Ryabov (Joint Inst. for Nuclear Research, Dubna (USSR). Lab. of Neutron Physics). 1965. 25p. (CONF-650301-35). Dep.(mn).

From Symposium on the Physics and Chemistry of Fission, Salzburg, Austria.

Level parameters, mean fission cross sections, and radiative capture in neutron reactions with ^{235}U nuclei at 0.002 to 3 kev were measured by a time-of-flight method using pulsed fast reactor as a resonance neutron source. A 2048-channel analyzer with channel width 32, 16, and 8 μsec was used for recording energy intervals at 0.002 to 0.005, 0.005 to 1.5, and 1.5 to 30 kev, respectively, resulting in a resolution of $\Delta t/L = 0.04$ μsec/m. (R.V.J.)

104.

3385 MEASUREMENTS OF THE VARIATION OF THE RATIO OF THE CAPTURE AND FISSION CROSS SECTIONS OF URANIUM-235. L. W. Weston (Oak Ridge National Lab., Tenn.), G. deSaussure, R. Gwin, J. E. Russell, and R. W. Hockenbury. Trans. Am. Nucl. Soc., 7: 270(Nov. 1964).

105.

3476 (IA-980) NUCLEAR CROSS SECTIONS FOR FAST REACTORS. S. Yiftah and M. Sieger (Israel. Atomic Energy Commission. Soreq Research Establishment, Rehovoth). July 1964. 111p. Dep.(mn).

See also A/CONF.28/P/510.

Some revisions to the previously published 16-group YOM cross section set are presented. An extension is made from 16 to 20 groups by additions in the 2 to 14 Mev range. Both sets take account of new information on neutron angular distributions, modified inelastic scattering matrices, modified capture cross sections for fissile nuclides, modified $\bar{\nu}$, and modifications for errors and omissions. The materials studied are the same as in YOM, namely: ^{242}Pu, ^{241}Pu, ^{240}Pu, ^{239}Pu, ^{238}U, ^{235}U, ^{233}U, ^{232}Th, Bi, Pb, Ta, Mo, Nb, Zr, Ni, Fe, Cr, V, Ti, K, Al, Na, O, C, and B. (R.E.U.)

1966

106.

3398 ANALYSIS AND MEASUREMENT OF PuAl–POLYETHYLENE SYSTEMS IN THE PCTR. Albertson, D. G.; Busselman, G. J. (Battelle-Northwest, Richland, Wash.). Trans. Amer. Nucl. Soc., 8: 448-9(Nov. 1965).

107.

4829 (GEMP-173(Suppl.1)) NINE AND SIXTEEN GROUP CROSS SECTIONS FOR REACTOR ANALYSIS. Supplement No. 1. Bermanis, H. L.; Henderson, W. B.; Robertson, C. S. Jr. (General Electric Co., Cincinnati, Ohio. Nuclear Materials and Propulsion Operation). Sept. 29, 1965. Contract AT(40-1)-2847. 11p. Dep. mn; CFSTI $1.00 cy, $0.50 mn.

Additions and modifications to the sixteen group cross sections listed in the first edition of GEMP-173 dated January 18, 1963, are reported. (auth)

108.

30342 (AERE-M-1670) ETA AND NEUTRON CROSS SECTIONS OF ^{235}U FROM 0.03 TO 200 ev. Brooks, F. D.; Jolly, J. E.; Schomberg, M. G.; Sowerby, M. G. (Atomic Energy Research Establishment, Harwell (England)). Feb. 1966. 17p. Dep. mn.

Measured values of eta, σ_T and σ_F and derived values of σ_C are given for ^{235}U from 0.03 to 200 ev. The averages of the cross sections and of eta and alpha have been calculated for a number of energy groups together with various ratios and integrals of these quantities which are of interest for reactor design studies. Comparisons are made between the results obtained in the present experiment and those from other laboratories. (auth)

109.

15774 MASS SPECTROMETRIC MEASUREMENTS OF THE RATIO OF THERMAL NEUTRON CAPTURE TO FISSION FOR ^{241}Pu. Cabell, M. J.; Wilkins, M. (Atomic Energy Research Establishment, Harwell, Eng.). J. Inorg. Nucl. Chem., 27: 2481-91(Dec. 1965). (AERE-R-4865).

Highly enriched samples of ^{241}Pu were irradiated in five different positions in a flux scanning tube of the PLUTO reactor so that α (the ratio of neutron capture to fission) could be measured for this nuclide. Highly enriched samples of ^{239}Pu and ^{240}Pu were also irradiated at the same time so that allowance could be made for the effects of their destruction. Experimental values of $\hat{\alpha}$ for ^{241}Pu varied from 0.352 to 0.395, with a mean value of 0.371 ± 0.016. Correcting all results to 2200 m/sec neutrons gave a final value for α_0 of 0.390 ± 0.023. Where a comparison was possible with previously published data, good agreement was found. (auth)

110.

10045 (AI-66-16) TUNGSTEN RESONANCE INTEGRALS AND DOPPLER COEFFICIENTS. Second Quarterly Report, October–December 1965. Carpenter, S. G.; Otter, J. M.; Paschall, R. K.; Royden, H. N. (Atomics International, Canoga Park, Calif.). Feb. 11, 1966. Contract NAS3-7982. 29p. (NASA-CR-54888).

The spectra of real and adjoint fluxes at the sample position at the center of the Sodium Graphite Reactor Critical Assembly (SGR-CA) were calculated with a one-dimensional diffusion code using 15 energy groups. Fifteen-group sample cross sections were calculated for a number of sample materials and sizes. These cross sections, together with the calculated spectra, were used in a perturbation theory calculation of sample reactivities and Doppler coefficients. Some sample cross sections were recalculated by using the cross section definition appropriate to unperturbed fluxes. Perturbation theory calculations made by using these cross sections were in good agreement with most experimental results. Work on Monte Carlo calcula-

tion of resonance integrals and Doppler coefficients has started. Self-shielding of the 1/v contribution to the resonance integral and reactivity effects due to sample thermal expansion were estimated with the aid of Wigner's rational approximation to the escape probability for a lump. The indicated thermal expansion effect is quite large for samples not under cadmium. The first set of radial flux maps using tungsten and gold foils was completed. The reactivity of a natural-tungsten slug was measured as a function of temperature up to 1250°K, both bare and under cadmium. The measured points lie on a smooth curve with little scatter. Each reactivity is determined with a precision of about 0.002 to 0.004 cents; the maximum reactivity change from room temperature is about 0.12 cents. Auxiliary measurements were carried out that demonstrate the insignificance of the effects of scattering by heavy elements in the sample. Preliminary calibration of the epithermal sensitivity of the oscillator was carried out with gold and uranium samples. (auth)

111.

28265 (AI-66-85) TUNGSTEN RESONANCE INTEGRALS AND DOPPLER COEFFICIENTS. Quarterly Progress Report No. 3, January-March 1966. Carpenter, S. G.; Levitt, L. B.; Otter, J. M.; Paschall, R. K.; Royden, H. N. (Atomics International, Canoga Park, Calif.). May 13, 1966. Contract NAS3-7982. 17p. (NASA-CR-54954).

The final 15-group calculation of parameters of the 10.6-in. SGR-CA lattice was completed. Multigroup effective cross sections were calculated for the actual isotopic compositions of the ^{182}W and ^{186}W samples. From these (and previous) cross sections and the multigroup real and adjoint fluxes, expected reactivities were computed for samples of natural tungsten, ^{182}W, ^{186}W, and gold. Doppler coefficients were calculated for the three tungsten samples. Agreement of these quantities with experiment is as good as could be expected for absolute reactivity calculations; however, measured reactivity ratios among the three samples are significantly different from calculated ratios. An existing Monte Carlo resonance-escape code was modified so that only those neutron histories resulting in passage through the sample are followed. This relatively minor modification has greatly increased the efficiency of the code. Activation measurements of the ^{186}W resonance integral were completed at two temperatures, 293 and 473°K. Reactivity worths and Doppler coefficients of samples enriched in ^{182}W and in ^{186}W were measured in the cadmium sleeve at the center of SGR-CA. Doppler measurements were made at several temperatures from room temperature to above 1000°K. Reactivities of gold samples with widely varying surface-to-mass ratio were measured to allow absolute calibration of the reactivity measurements. (auth)

112.

34618 MEASUREMENTS OF THE EPITHERMAL-NEUTRON CAPTURE-TO-FISSION CROSS-SECTION RATIOS IN ^{233}U AND ^{235}U. Conway, D. E.; Gunst, S. B. (Bettis Atomic Power Lab., West Mifflin, Pa.). Trans. Amer. Nucl. Soc., 9: 245-6(June 1966).

113.

47074 CRITICAL EXPERIMENTS AND THE MAIN CHARACTERISTICS OF THE ANNA ZERO POWER REACTOR. Dabek, Waclaw; Kubowski, Jerzy (Inst. of Nuclear Research, Swierk, Poland). Nukleonika, 11: 213-26(1966).

A short description is presented of the assembly start-up, critical experiments in different core configurations and measurement of various operational characteristics. The data obtained during operation period are summarized. Some safety aspects are considered. Present and future programs of experiments are included. (auth)

114.

34949 EXPERIMENTS ON INELASTIC SCATTERING IN A FAST-REACTOR CORE USING THE SPHERICAL-SHELL TECHNIQUE. Davey, William G.; Amundson, Paul I. (Argonne National Lab., Idaho Falls, Idaho). Trans. Amer. Nucl. Soc., 9: 187(June 1966).

115.

44622 (CONF-660303, pp 796-808) **A CRITICAL EVALUATION OF FAST FISSION CROSS SECTIONS. Davey, William G. (Argonne National Lab., Idaho Falls, Idaho).**

The published cross sections of ^{235}U, ^{238}U, ^{239}Pu, and ^{240}Pu from 1 kev to 10 Mev were studied to select best cross sections for fast reactor analysis. Emphasis was placed on determining the reference data used and, where necessary, revising the measurements to accord with currently accepted cross sections. Some checks were made by comparing calculations based on the selected cross sections with integral measurements in the fission spectrum and in 19 fast reactor critical assemblies. The study shows a clear need for more work on precise fission cross sections, particularly that of ^{235}U since it is widely used as a reference. (auth)

116.

17851 MEASUREMENT AND ANALYSIS OF THE FISSION CROSS-SECTION OF ^{239}Pu FROM 0 TO 5 kev. de Saussure, G.; Blons, J.; Jousseaume, C.; Michaudon, A.; Pranal, Y. (Centre d'Etudes Nucleaires, Saclay, France). pp 205-18 of STI/PUB/101(Vol. 1). (In French).

The cross section of ^{239}Pu for low-energy neutrons was measured between 0.16 ev and 5 kev by the time-of-flight method, with a gas scintillator containing 310 mg of ^{239}Pu. Under the best conditions and at high energy, the resolving power of the measurement was 3 ns/m. The Saclay linear accelerator was used as a pulsed neutron source. The measurement consisted in the successive determination of the fission rate (by means of a gas scintillator) and the incident neutron energy spectrum (by means of a BF_3 counter whose efficiency was proportional to $E^{-\frac{1}{2}}$). The background was determined by interposing black resonance screens in the neutron beam. The quantity $\sigma_F\sqrt{E}$ is proportional to the ratio of the fission rate to the counting-rate of the BF_3 detector. It was calibrated by reference to the resonance at 7.8 ev, the parameters of which are known. At the discrimination threshold, adjusted to an alpha-ray build-up rate of 1 counts/s, an efficiency of 65% for the detection of fissions was obtained. The fission cross section was measured in several sequences to observe the various energy ranges with sufficient resolving power. A detailed description is given of the experimental apparatus, the tests with the

scintillator and the experimental conditions in which the various measurement sequences were carried out. A curve showing the fission cross section as a function of energy is included. The results of an analysis of the resonance parameters are also reported. (auth)

117.

13770 (GA-6576) A COMPILATION AND EVALUATION OF THE NUCLEAR DATA AVAILABLE FOR THE MAJOR PLUTONIUM ISOTOPES. Drake, M. K.; Dyos, M. W. (General Atomic, San Diego, Calif. John Jay Hopkins Lab. for Pure and Applied Science). July 30, 1965. Contracts AT(04-3)-167; AT(04-3)-187. 173p. Dep. mn. CFSTI $5.00 cy, $1.00 mn.

The available data for ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu were compiled and evaluated for neutron energies between thermal and 15 Mev. The data obtained from this evaluation are presented in the form used in the General Atomic data library. A best set of single level resonance parameters was obtained, over the resolved resonance region for each of the above nuclides, in order that some estimate of the Doppler coefficient of reactivity can be made. (auth)

118.

17697 (WANL-TME-1028) A NEW CROSS SECTION LIBRARY FOR THE n + ^{235}U REACTIONS. Drawbaugh, D. W.; Gibson, G.; Melnick, M. M. (Westinghouse Electric Corp., Pittsburgh, Pa. Astronuclear Lab.). Nov. 11, 1964. Contract SNP-1. 60p. Dep. mn. CFSTI $3.00 cy, $0.75 mn.

In an enriched undermoderated reactor resonance absorption of epithermal neutrons must be considered. A new set of neutron cross sections of ^{235}U are presented. In particular resonances are included in the experimentally unresolved energy range from ~50 ev to 30,000 ev. Resonance self-shielding and Doppler broadening in this range is handled by a picket fence model for the resonance structure. Account is taken of the fact that within this statistical range there are in general two different distributions for each partial width of a resonance corresponding to different spin states of the compound nucleus. In the unresolved region use is made of the cross section measurements reported by Uttley and those in BNL-325. At low energies, $E < 1.86$ ev, the data of Shore and Sailor is used and at high energies, $E > 3.18 \times 10^4$ ev, data in the BNL-325 report is used. For the experimentally resolved energy region resonance parameters obtained recently by the Saclay group and also parameters given in the BNL-325 1964 supplement are used; smooth background cross sections due to the negative energy resonance are included. Values for the average number of prompt neutrons per fission are obtained from the empirical formulas based on measurements by Moat, Mather and Fieldhouse, and the fission energy spectrum is obtained from the experimental results of Cranberg and Nereson. The variation with energy of the resulting cross sections are shown to compare favorably with various experimental cross sections. Also results of calculations with the GAM, QUERY, and TNS codes are compared with results of integral experiments, viz. the fission integral, the absorption integral, and measured values of $\langle\alpha\rangle$. A comparison is also made of the calculated hydrogen-worth with the measured values for a particular experiment. (auth)

119.

24419 INTEGRAL $^9Be(n,2n)$ CROSS SECTION. Felber, F. F. Jr.; Farmelo, D. R.; Van Sickle, V. C. Nucl. Sci. Eng., 25: 1-7(1966).

The integral $^9Be(n,2n)$ cross section was determined by using Be and BeO samples irradiated in the core of the Battelle Research Reactor. The value obtained was 460 ± 60 mb for neutrons having energies above 2.7 Mev and a fission spectrum. **Fast-neutron dosimetry was accomplished using iron, nickel, and titanium threshold detectors. The results are based on measurement of the total amount of helium produced during irradiation. Corrections were made for the small contribution from (n,α) reactions on ^9Be and ^{16}O.** (auth)

120.

30723 (NAA-SR-11831) THE DTF-II SHIELDING DATA LIBRARY. Green, W. B. (Atomics International, Canoga Park, Calif.). May 25, 1966. Contract AT(11-1)-Gen-8. 80p. Dep. mn. CFSTI $3.00 cy, $0.75 mn.

A new data library is available for use with the DTF-II S_N theory transport code in computing neutron fluxes in shield systems and reactor criticality. Two group structures are available: a 21-group structure with nine energy groups in the 0.41- to 14.9-Mev range (primarily for use in shielding problems) and a 16-group structure (suitable for use in criticality calculations). The epithermal data for both group structures were obtained by group-averaging pointwise data over a typical SNAP spectrum using the GAM-II code (these calculations were performed by personnel at Oak Ridge National Laboratory). Data for the thermal range were taken from previously generated data libraries at Atomics International. A complete listing of the library data is also included. (auth)

121.

34782 (NAA-SR-Memo-11718) THE DTF-II SHIELDING DATA LIBRARY. Green, W. B. (Atomics International, Canoga Park, Calif.). Nov. 2, 1965. Contract AT (11-1)-Gen-8. 105p. Dep. mn. CFSTI $4.00 cy, $0.75 mn.

A new data library is available for use with the DTF-II S_N theory transport code in computing neutron fluxes in shield systems and reactor criticality. Two group structures are available: a 21 group structure with nine energy groups in the .41 to 14.9 Mev range (primarily for use in shielding problems) and a 16 group structure (suitable for use in criticality calculations). The epithermal data for both group structures were obtained by group averaging pointwise data over a typical SNAP spectrum using the GAM-II code. Data for the thermal range were taken from previously generated data libraries at Atomics International. The available materials, the generation of the library data, and information for those wishing to use this library in DTF calculations are described. A complete listing of the library data is also included. (auth)

122.

26422 (BNL-10058) THE SENSITIVITY OF REACTOR CHARACTERISTICS TO CROSS SECTION UNCERTAINTIES BELOW 100 ev. Hellens, R. L. (Brookhaven National Lab., Upton, N. Y.). Mar. 1966. Contract AT(30-2)-Gen-16. 44p. (CONF-660303-1). Dep. mn. CFSTI $2.00 cy, $0.50 mn.

From American Physical Society, Conference of Neutron

Cross Section Technology, Washington, D. C.

The development of accurate computational methods for thermal reactor lattices has reached the point at which neutron cross section uncertainties are a major limitation to the confidence with which theory can be applied to practical situations. This sensitivity to nuclear data is readily seen in the analysis of extremely simple lattice experiments, although it is still present, of course, in calculations for power reactors of greater geometrical complexity where errors in describing spatial effects may effectively hide basic discrepancies. The basic lattice data and computational methods are briefly reviewed and the connection between uncertainties in cross sections and in predicted reactor characteristics is traced for some typical reactor lattices. The effect during reactor life of changes in the cross sections assigned to both fertile and fissile isotopes in the uranium and plutonium chains is described for some light and heavy water moderated reactors. (auth)

123.

30349 (JAERI-1102) PROCEEDINGS OF THE SEMINAR ON FAST-NEUTRON CROSS SECTIONS. (Japan Atomic Energy Research Inst., Tokyo). Aug. 1965. 110p. (In Japanese). Dep. mn.

A survey of data was made in order to discuss in detail the problem of neutron cross section data from the experimental and theoretical points of view. The data and their phases involved were discussed for the following: reactions leaving the discrete states of final nucleus, reactions leaving the continuous state of final nucleus, (n,γ) reactions and resonance phenomena, and nuclear fusion. (M.O.W.)

124.

14062 (JUL-214-RG) MULTIGRUPPEN-WIRKUNGSQUERSCHNITTE FUER DIE STRUKTUR-, LOESUNGS- UND BRUTMATERIALIEN DES MOSEL-REAKTORS. (Multigroup Effective Cross Sections for the Structural, Solution, and Breeding Materials of the Mosel Reactor). Jung, Hans H. (Kernforschungsanlage, Juelich (West Germany). Institut fuer Reaktorentwicklung). Dec. 1964. 40p. Dep. mn.

Group cross sections for 68-group structure of the GAM-1 code for absorption, fission, and (n,2n) processes for the structural solution, and breeding materials of the Mosel Reactor concept (Ni, Mo, Cr, Fe, Na, Be, F, and Th) are given. The sources of the basic data are compiled. (tr-auth)

125.

38451 MULTIGROUP CROSS SECTIONS OF RESONANCE ABSORBERS. Keane, A. (Wollongong Univ. Coll., N. S. W.); Pollard, J. P. Nucl. Sci. Eng., 25: 439-40(1966).

The preparation of multigroup data, typically 18 groups, for input to codes such as CRAM and DSN for systems in which the low-energy resonances dominate is considered. (J.F.P.)

126.

34950 MEASUREMENT OF THE RATIO OF THE CAPTURE CROSS SECTION OF MANGANESE TO THE FISSION CROSS SECTION OF ^{235}U IN ZPR-3 ASSEMBLY 47, (SEFOR). Keeney, William P. (Argonne National Lab., Idaho Falls, Idaho). Trans. Amer. Nucl. Soc., 9: 187-8(June 1966).

127.

2972 ON GROUP CROSS SECTIONS OF FISSILE NUCLIDES IN THE UNRESOLVED ENERGY RANGE. Kelber, C. N.; Kier, P. H. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 8: 469(Nov. 1965).

128.

38420 (ANL-7120, pp 423-33) INFLUENCE OF SOME IMPORTANT GROUP CONSTANTS ON INTEGRAL FAST REACTOR QUANTITIES. Kuesters, H.; Metzenroth, M. (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Neutronenphysik und Reaktortechnik).

Uncertainties in the group constants used in multigroup calculations for large fast reactors are considered. **The effect of resonance self-shielding on multiplication factors for some reactors is shown. The influence of different weighting spectra used in various sets of group constants** on the Doppler coefficient, sodium coefficient, and breeding ratios was investigated. The characteristics and preparation of the first Karlsruhe group-constant set KFK 26-10 are briefly described. Effects of uncertainties in microscopic data on reactor calculations are demonstrated. Results of calculations using different group-constant sets are compared with measured data for ZPR III/39. (A.G.W.)

129.

26333 (CONF-660303-2) PREPARATION OF EVALUATED CROSS SECTION LIBRARIES. Lubitz, Cecil R. (Knolls Atomic Power Lab., Schenectady, N. Y.). [nd]. Contract W-31-109-eng-52. 14p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From American Physical Society, Conference of Neutron Cross Section Technology, Washington, D. C.

The process of putting together a library of neutron cross sections for use in reactor and shielding calculations is discussed. A modular, user-oriented, automated data handling and computational system designed for a high-speed computer is briefly described. The theoretical methods used for interpolation and extrapolation of experimental results are discussed, as well as the problems associated with feeding back experimental information into our libraries. A number of problems concerning the status of cross section measurements and the methods of reporting them are enumerated, emphasizing those experimental points of particular importance in the evaluation process. (auth)

130.

3010 (IDO-14663) BURNUP DETERMINATION OF NUCLEAR FUELS. Project Report for the Quarter, April 1-June 30, 1965. Maeck, William J.; Rein, James E. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). Sept. 1965. Contract AT(10-1)-205. 23p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

In the development of methods for the accurate determination of the burnup of nuclear fuels, the main emphasis was the measurement of ^{233}U thermal fission yields for many stable and long-lived fission products. A capsule containing a weighed quantity of highly enriched ^{233}U, irradiated to 11.4 atom percent fission, was dissolved and analyzed for the number of atoms of each uranium isotope and for the number of atoms of various stable and long-lived fission product nuclides. **From these data the capture-to-fission ratio (α) and absolute fission yields were determined. The value for α_0 is 0.0986 ± 0.0015. (auth)**

131.
34512 (ANL-7170) EFFECTS OF LEAKAGE UPON EVALUATION OF FAST-REACTOR MULTIGROUP CROSS SECTIONS FOR REFLECTORS. Meneghetti, David (Argonne National Lab., Ill.). Feb. 1966. Contract W-31-109-eng-38. 16p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

Group cross sections for aluminum and nickel reflectors, which include effects of energy-dependent leakages into and out of reflector regions, are considered. Leakages are introduced as absorptions in fine-group ELMOE calculations. Iterations are made between coarse-group regional fluxes from criticality calculations and the leakage-modified ELMOE analyses. (auth)

132.
46997 CRITICALITY STUDIES OF HYDROGEN-MODERATED URANIUM SYSTEMS. Newlon, C. E. (Oak Ridge Gaseous Diffusion Plant, Tenn.). Nucl. Safety, 7: 418-23(Summer 1966).

The general applicability of computational surveys in establishing criticality limits and identifying optimum criticality conditions for nuclear safety control is briefly reviewed. Recent studies made with 1-group theory, 16-group transport theory, and a first-order solid-angle method for interacting arrays include: the infinite-medium multiplication factor, k_∞, of homogeneous hydrogen-moderated U of all ^{235}U enrichments, critical spacings of arrays of individually subcritical units in air, and the effects of composite metal-water reflectors on the minimum critical cylinder dimensions of U systems enriched to 4.98 wt. % ^{235}U. Good correlation with applicable experimental data is noted. Specific items discussed are the significance of k_∞ minima found for high-enrichment systems of undermoderated U, the ^{235}U-enrichment and -concentration limits derived from k_∞ determinations, and the inverse ^{235}U-enrichment phenomena found for both individual units and interacting arrays. (auth)

133.
2960 EFFECT OF REVISIONS TO THE MULTIGROUP REPRESENTATIONS OF THE BERYLLIUM (n,2n) AND BERYLLIUM (n,α) REACTIONS. Ottewitte, E. H. (Atomics International, Canoga Park, Calif.). Trans. Amer. Nucl. Soc., 8: 456-7(Nov. 1965).

134.
24334 (BNL-10083) DIFFERENTIAL AND INTEGRAL CROSS SECTIONS FOR THE TRANSURANIUM ELEMENTS. Pearlstein, S. (Brookhaven National Lab., Upton, N. Y.). [1964]. Contract AT-30-2-GEN-16. 16p. (CONF-660303-8). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From American Physical Society, Conference of Neutron Cross Section Technology, Washington, D. C.

Cross sections useful for determining the production of transuranium elements are evaluated using the Breit-Wigner single level formula. Thermal cross sections and resonance integrals for neutron capture and fission are presented for the ^{234}U, ^{236}U, ^{237}Np, ^{238}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, and ^{244}Cm nuclides. (auth)

135.
3430 DIRECT MEASUREMENT OF EFFECTIVE CAPTURE-TO-FISSION RATIO AT LOW FLUX. Redman, W. C.; Bretscher, M. M. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 8: 531-2(Nov. 1965).

136.
45071 (CONF-660303, pp 1092-7) LOW FLUX DETERMINATION OF CAPTURE-TO-FISSION RATIO. Redman, W. C.; Bretscher, M. M. (Argonne National Lab., Ill.).

A novel method which makes possible a direct measurement in low-power reactors of the effective capture-to-fission ratio, $\bar{\alpha}$, has been developed. It involves a comparison of reactor response to oscillated samples of a fissile material, an absorber, and a spontaneous fission source, augmented by an experimental determination of the respective fission rate, capture rate, and neutron-source strength. These experimental results, combined with the number of neutrons per fission of the fissile material, yield a value for the quantity $1 + \bar{\alpha}$. Applications for this technique are illustrated by the results for measurements with ^{235}U in both an undermoderated critical assembly of moderately enriched ^{235}U in light water and an epithermal, 1/E neutron spectrum. (auth)

34617 LOW FLUX MEASUREMENT OF ^{235}U EPICADMIUM CAPTURE-TO-FISSION RATIO. Redman, W. C. (Argonne National Lab., Ill.); Bretscher, M. M. Trans. Amer. Nucl. Soc., 9: 245(June 1966).

137.
34905 (GA-7059) ADEQUACY OF FAST AND INTERMEDIATE CROSS SECTION DATA FROM MEASUREMENT OF NEUTRON SPECTRA IN BULK MEDIA. Russell, J. L. Jr.; Profio, A. E. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Apr. 27, 1966. Contract AT(04-3)-167. 17p. (CONF-660303-45). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From American Physical Society, Conference of Neutron Cross Section Technology, Washington, D. C.

Methods used to calculate intermediate neutron spectra in fast reactors are suspect because of inadequacies in cross section data and as well as approximations used in preparing (group averaging) cross section data for transport codes. Direct measurement of neutron spectra in bulk media provides a means of evaluating discrepancies. The status of spectrum measurements is reviewed, preliminary results of spectrum measurements are discussed, and some new measurement concepts are described. (auth)

138.
17856 INTERACTION OF NEUTRONS WITH ^{235}U NUCLEI IN THE ENERGY RANGE 2 ev-30 kev. Wang, Shih-ti; Wang, Yung-ch'ang, Dermendzhiev, E.; Ryabov, Yu. V. (Joint Inst. for Nuclear Physics, Dubna, USSR). pp 287-305 of STI/PUB/101(Vol. 1). (In Russian).

The fast pulsed reactor was used to measure the total cross section, the fission cross section and the radiative capture cross section of ^{235}U by the time-of-flight method with a resolution of ~0.04 μs/m. The flight length was 1000 m. The time spectra were recorded by 2048-channel time analyzers. To measure the total cross sections by the transmission method, a resonance-neutron scintillation

counter with lithium glass was used. The fission, radiative-capture, the total cross sections were measured by the auto-indication method using a large liquid scintillation detector with cadmium added to the solution. The sample to be investigated was placed in the axis of a cylindrical opening in the vessel, in a geometry close to 4π. The volume of the detector was scanned by 32 FEU-24 photomultipliers. The fissions were identified by the delayed coincidences between the scintillations corresponding to the recording of the prompt gamma rays and the moderated prompt fission neutrons. Radiative capture is not accompanied by a delayed pulse. The efficiency of recording fissions and radiative capture was ~50 and ~25% respectively, while the background (expressed as a percentage of the strong resonances) was ~1 and ~15% respectively. The area method was used to obtain the parameters $g\Gamma_n$, Γ, Γ_f, and Γ_γ for a number of low-lying levels. The average widths of $\bar{\Gamma}_\gamma$ and $\bar{\Gamma}_f$ over all the levels were (42 ± 3) mv and (51 ± 6) mv respectively. The levels were divided into two systems with $\alpha_1 < 1 (\bar{\alpha}_1 \sim 0.5)$ and $\alpha_2 > 1 (\bar{\alpha}_2 \sim 1.5)$ and $\bar{\Gamma}_f^1 \approx 72$ mv and $\bar{\Gamma}_f^2 \approx 25$ mv respectively, which may be connected with the two spin values 3^- and 4^- of the compound nucleus. The energy dependence of the ratio of radiative-capture to fission cross sections was obtained in the neutron-energy range from 2 ev to 30 kev. Measurements using a sample of ^{238}U instead of ^{235}U made it possible to take into account experimentally the contribution of the radiative capture of ^{238}U to α. (auth)

139.

21967 (AEET-234) REVIEW OF NUCLEAR DATA RELATED TO THORIUM FUEL CYCLE. Singh, R. Shankar (Atomic Energy Establishment, Trombay (India)). 1965. 10p. Dep. mn.

A knowledge of accurate and reliable nuclear data for the materials in any fuel cycle is essential in assessing its technical feasibility and economic potential. The nuclear parameters, in which a reactor physicist is usually interested, are discussed for the heavy elements associated with the thorium cycle, such as ^{232}Th, ^{233}U, ^{233}Pa, and ^{234}U and the fission products. The existing data for these materials and the gaps that have to be filled up are described. Temperature dependent effective group capture cross sections for thorium, which were evaluated with the available resonance parameters, are also tabulated. (auth)

140.

15718 (IDO-17083) MEASUREMENT OF THE ABSOLUTE VALUE OF ETA FOR ^{233}U, ^{235}U AND ^{239}Pu USING MONOCHROMATIC NEUTRONS. Smith, J. R.; Reeder, S. D.; Fluharty, R. G. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). Feb. 1966. Contract AT(10-1)-205. 51p. Dep. mn. CFSTI $3.00 cy, $0.50 mn.

The absolute value of eta, the number of fission neutrons per absorption, was measured for ^{233}U, ^{235}U, and ^{239}Pu, using monochromatic neutrons from the crystal spectrometer at the Materials Testing Reactor. Measurements were made on all three isotopes at 0.025-ev neutron energy, and on ^{233}U and ^{239}Pu at 0.057 ev. The Bragg beam from Be (0002) was passed through a mechanical monochromator to remove higher-order neutrons and yield a truly monochromatic beam. The neutron detector was a manganous sulphate bath, which absorbed in turn the Bragg beam and then the fission neutrons produced when the beam was completely absorbed in a fissionable sample. The ratio of the **levels of ^{56}Mn activities produced in the two types of irradiation yielded the value of eta for the fissionable material of the sample, after the application of a few small corrections. The method of least squares was used to extract the values of eta from the experimental data. (auth)**

141.

26080 (BAW-393-5) NEUTRON CROSS SECTIONS FOR ^{233}U. Snidow, N. L. (Babcock and Wilcox Co., Lynchburg, Va. Atomic Energy Div.). Apr. 1966. Contract AT (38-1)-393. 67p. Dep. mn. CFSTI $3.00 cy, $0.75 mn.

^{233}U neutron cross sections are tabulated for 1198 energy points from 0.0001 ev to 10 Mev, and the sources of the values are given. The file is intended for use with a Fortran program in obtaining group average cross sections for dilute ^{233}U systems. In addition, 1/E weighted averages for a 68-group set are given. (auth)

142.

36436 NEUTRON-PRODUCTION CROSS SECTION OF ^{238}Pu IN A FAST SPECTRUM. Stubbins, Warren Fenton; Barton, David M.; Lonadier, Frank D. Nucl. Sci. Eng., 25: 377-82(1966). (TID-22431).

The production cross sections $\overline{\sigma_p} = (\nu - 1 - \alpha)\sigma_f$, where $\alpha = \sigma_c/\sigma_f$ for ^{238}Pu and ^{239}Pu, were compared in the fast neutron flux at the center of a bare spherical critical assembly of ^{239}Pu. These quantities averaged by the fast-neutron spectrum indicate nearly the same properties for the even-even ^{238}Pu nucleus as for the odd-even ^{239}Pu nucleus. The ratio measured in a neutron flux peaked at 0.25 Mev with an average neutron energy $\bar{E} = 1.67$ Mev is $\overline{\sigma_p}(^{238}Pu)/\overline{\sigma_p}(^{239}Pu) = 1.01 \pm 0.06$, and $\overline{\sigma_p}(^{238}Pu) = 3.76 \pm 0.23$ b. The results of this study indicate that ^{238}Pu metal probably has a critical mass of the same order of magnitude as ^{239}Pu metal. (auth)

143.

24707 (CONF-660303-6) CROSS SECTION SENSITIVITY CALCULATIONS FOR SMALL HYDROGEN MODERATED SYSTEMS. Weinstein, S.; Feiner, F.; Cooper, K. V.; Armstrong, S. I. (Knolls Atomic Power Lab., Schenectady, N. Y.). Mar. 22, 1966. Contract W-31-109-eng-52. 17p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From American Physical Society, Conference of Neutron Cross Section Technology, Washington, D. C.

Experiments were performed with the KAPL Solid Homogeneous Assembly. The cores built on the assembly are leakage dominated and were designed to have an enhanced sensitivity to high energy cross sections. Sensitivity calculations were made using evaluated microscopic cross sections for oxygen, zirconium, carbon, and uranium-235. (M.O.W.)

144.

17852 MEASUREMENT OF FISSION CROSS-SECTIONS FOR NEUTRONS OF ENERGIES IN THE RANGE 40-500 kev. White, P. H.; Hodgkinson, J. G.; Wall, G. J. (Atomic Weapons Research Establishment, Aldermaston, Eng.). pp 219-33 of STI/PUB/101(Vol. 1).

Measurements were made of the fission cross section of ^{233}U, ^{234}U, ^{236}U, ^{237}Np, ^{239}Pu, and ^{241}Pu at several neutron energies between 40 and 500 kev. Measurements in this energy range are of importance in reactor calculations especially in fast dilute systems where the neutron flux is

high in the 10–100-kev energy range. Recent measurements of the ^{235}U fission cross section gave absolute values slightly lower than previous data. The present series of measurements are made relative to the new values of the ^{235}U fission cross section using back-to-back ionization chambers. The fissile foils were assayed by α-assay, direct weighing and coulometry. Good agreement was obtained between these assays. The fission measurements have an estimated accuracy of between 1 and 2% and, combined with the error on the ^{235}U fission cross section, give a final error of about 3% in the fission cross sections. The results together with those of previous measurements are given, and the corrections for fission fragment absorption, backgrounds, and scattering are discussed. (auth)

145.

32391 GAMMA SPECTROMETER MEASUREMENTS OF ^{238}U CAPTURE-TO-FISSION RATIO. Yurova, L. N.; Bushuev, A. V. At. Energ. (USSR), 20: 60-1(Jan. 1966). (In Russian).

The capture-to-fission cross section ratio of ^{238}U may be measured by determining the relative intensities of two lines of the γ spectrum of a neutron-irradiated specimen; one of these lines belongs to an isotope formed by capture, such as ^{239}U or ^{239}Np, the other to the fission product ^{140}La. Evaluation of these data requires knowledge of the absolute yields of these lines, of the probability of formation of ^{140}La, and of the efficiency of the γ spectroscope which depends largely on the properties of the detector. Even though all these factors are not known with the desired precision, valuable data were obtained. The ratio may be determined more conveniently by calibrating the specimen before the actual measurement in a thermal flux, assuming that the ^{140}La is formed only by the fission of ^{235}U. This method was used in experiments carried out on the BR-1 reactor; an accuracy of 0.5% was reached. The method is considered suitable for determining other reactor parameters, such as the initial breeding ratio, fast-neutron multiplication factor, etc; it may be used for systems ranging from fast assemblies to thermal reactors. (TTT)

1967

146.

1208 (LA-DC-7962) COMPARISON OF MULTIGROUP CROSS SECTION SETS USED IN REACTOR CALCULATIONS. Battat, M. E.; LaBauve, R. J. (Los Alamos Scientific Lab., Univ. of California, N. Mex.). [1965]. Contract W-7405-eng-36. 12p. (CONF-661019–1). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From International Conference on Fast Critical Experiments and Their Analysis, Argonne, Ill.

In order to provide a common basis for comparing various neutron multigroup cross-section sets, calculational models of two well defined critical experiments were specified. The critical assemblies chosen for this study were the Jezebel (Pu) assembly and the 610-liter ZPR VI (Assembly No. 2) UC core. The Hansen-Roach 16-group and Russian 26-group cross-section sets were used for the initial comparisons. Calculations were made to determine the critical core volumes and central reactivity worths using the DTF-IV transport code in the S_8 approximation and assuming spherical geometry. In addition, two types of perturbation theory calculations were made to arrive at central reactivity worths for small samples. The first of these is used to compute worths in terms of simple atomic perturbation cross sections. The second is based on transport perturbation theory, using real and adjoint currents and fluxes, with results expressed in terms of $\Delta k/g$-atom. Wherever available, experimental data are presented for comparison with calculations. Central reactivity worths have been calculated for the following materials: ^{239}Pu, ^{240}Pu, ^{238}U, ^{235}U, depleted U, Fe, Ni, Cr, Na, Ta, Al, Mo, and B. In the case of ZPR VI, the reactivity effect of voiding the core Na was also calculated. (auth)

147.

4321 HARWELL MASS SPECTROMETRIC MEASUREMENTS OF THE RATIO OF NEUTRON CAPTURE TO FISSION FOR ^{233}U, ^{235}U, ^{239}Pu, AND ^{241}Pu, IN REACTOR AND MAXWELLIAN NEUTRON SPECTRA. Cabell, M. J. (Atomic Energy Research Establishment, Harwell, Eng.). Vienna, International Atomic Energy Agency, 1966, Preprint No. CN-23/21, 24p. (CONF-661014–14). DTIE.

From IAEA Conference on Nuclear Data, Paris.

Use of the mass-spectrometric method for the measurement of α, the ratio of neutron capture to fission, for the four nuclides ^{233}U, ^{235}U, ^{239}Pu, and ^{241}Pu is discussed. In order to obtain useful results as rapidly as practicable, the high neutron fluxes present in the lattice of a nuclear reactor are usually employed for the irradiations. Under these conditions the large epithermal contents of the neutron spectra made the computation of the 2200 m/sec values, from the measured values, less certain. More recently, similar measurements have been made in the thermal column of a reactor for ^{239}Pu and ^{241}Pu in an attempt to reduce uncertainty from this source. This technique and its limitations are discussed. (S.F.L.)

148.

5538 MASS SPECTROMÉTRIC MEASUREMENTS OF THE RATIO OF NEUTRON CAPTURE TO FISSION FOR ^{239}Pu AND ^{241}Pu, AND OF THE ABSORPTION AND CAPTURE CROSS SECTIONS OF ^{239}Pu, ^{240}Pu, AND ^{241}Pu, FOR MAXWELLIAN NEUTRONS. Cabell, M. J.; Wilkins, M. (Atomic Energy Research Establishment, Harwell, Eng.). J. Inorg. Nucl. Chem., 28: 2467-75(Nov. 1966). (AERE-R-5166).

Mixtures of highly enriched samples of ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu were irradiated for 295.6 days in the DIDO Reactor near the core. Neutron doses were monitored by cobalt metal wire. Neutron temperature was (116 ± 4°) with an unperturbed Maxwell–Boltzmann distribution. Results are given for mass analyses of the samples, both before and after the irradiation. (S.F.L.)

149.

2732 ANALYSIS OF THE FISSION CROSS SECTIONS OF ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{237}Np, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, AND ^{242}Pu FROM 1 keV TO 10 MeV. Davey, William G. Nucl. Sci. Eng., 26: 149-69(1966).

The published cross sections of ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{237}Np, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu from 1 keV to 10 MeV were carefully studied to select the best cross sections for fast reactor analysis. Emphasis was placed on determining the reference data used; and, where necessary, the published data were revised to accord with more accurate, currently accepted cross sections. Thus, it is believed that a consistent set of cross-section data was derived. Some cross checks were made by comparing calculations based on the selected cross sections with integral measurements in broad fast-neutron spectra. The study shows the great importance of the ^{235}U fission cross section in deriving other cross sections and emphasizes the necessity of re-evaluating nearly all fission cross sections, if it proves necessary to revise the ^{235}U data. (auth)

150.

4323 DIRECT AND ABSOLUTE MEASUREMENTS OF AVERAGE FISSION NEUTRON YIELD FROM ^{235}U AND ^{252}Cf. De Volpi, A.; Porges, K. G. (Argonne National Lab., Ill.). Vienna, International Atomic Energy Agency, 1966, Preprint No. CN-23/40, 12p. (CONF-661014-18). DTIE.

From IAEA Conference on Nuclear Data, Paris.

A recent survey by the IAEA has shown that significant discrepancies exist among absolute values of $\bar{\nu}$ and σ_f. For $\bar{\nu}(^{235}U)$ only one measurement, made in 1958 with a quoted precision of 1.5%, is considered independent, and the Westcott group [Atomic Energy Rev. 3: 2(1965)] reports that the original value has been lowered so that it is now 2.5% below the recommended least-squares average. There are four accepted values for $\bar{\nu}$ (^{252}Cf), one of which is also about 2.5% lower than the least squares fit; another is 1.6% low, while the average of all measurements is 1% below the recommended values derived from the multiparameter fit. A similar situation exists with regard to 2200 m/s fission cross-sections, wherein the discrepancies far exceed the precision quoted for each experiment. Since these measurements represent the cornerstone of a strongly interrelated structure of nuclear data utilized in reactor physics, it is important that they be independently and accurately evaluated, despite the fact that there exists strong confidence in σ_a, η, and α values which provide an overdetermined set of parameters. An experiment is reported which has been designed to circumvent certain plausible systematic errors which may be responsible for the discrepancies. The total neutron yield was measured for thermal neutron fission of ^{235}U and also independently for ^{252}Cf with an improved manganese bath apparatus. Evaluation of possible error sources has led to the adoption of a sequence of precision techniques subjected to extensive verification. The neutron yield of a fission counter was determined with the manganese bath; the accuracy of the bath system was independently corroborated with 0.7% precision against the U. S. National Bureau of Standards secondary neutron source. Absolute beta-gamma and relative gamma-gamma coincidence techniques are important facets of this calibration. The fission rate of the fission counter was found in a separate prompt fission-neutron coincidence experiment, borrowing well-established methods from beta-gamma coincidence work. In the course of this calibration it was discovered that angular anisotropy in fission neutron emission is much more a problem than universally realized; it is possible that some of the discrepancies in reported $\bar{\nu}$ and σ_f measurements result from discounting this correction too readily. This conclusion is supported with a series of angular traverses and by the contemporary $\bar{\nu}$ results reported. (auth)

151.

2706 (GA-7076) NEUTRON CROSS SECTIONS FOR ^{233}U. Drake, M. K. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Sept. 15, 1966. Contract AT(04-3)-167. 123p. Dep. mn. CFSTI $4.00 cy, $0.75 mn.

The available nuclear data for ^{233}U were critically analyzed, and recommended sets of neutron cross sections were obtained. Comparisons were made between resonance integrals obtained from the recommended data and those obtained from experimental measurements. The complete sets of neutron cross sections and associated data are presented in graphical and tabular form. The survey is believed to be complete for information available to March 1966. (S.F.L.)

152.

5527 (AECL-2618) THE RATIO OF CAPTURE TO FISSION IN ^{235}U AND ^{239}Pu. Durham, R. W.; Hanna, G. C.; Lounsbury, M.; Bigham, C. B.; Hart, R. G.; Jones, R. W. (Atomic Energy of Canada Ltd., Chalk River (Ontario). Chalk River Nuclear Labs.). Oct. 1966. 16p. (CN-23/2). Dep. mn. AECL $0.50.

The neutron capture-to-fission ratio and the resonance integral of ^{235}U were measured together with the capture-to-fission ratio, absorption cross section, and fission cross section of ^{239}Pu. The measurements were performed in a thermal and epithermal flux. (D.C.W.)

153.

5460 (KAPL-P-3176) EVALUATION OF NEUTRON CROSS SECTIONS: CALCULATIONAL METHODS AND EVALUATED LIBRARIES. Francis, N. C.; Lubitz, C. R; Reynolds, J. T.; Slaggie, E. L. (Knolls Atomic Power Lab., Schenectady, N. Y.). [1966]. Contract W-31-109-eng-52. 17p. (CONF-661014-6). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From IAEA Conference on Nuclear Data, Paris, France.

Two theoretical models are outlined which are useful, in conjunction with experimental data, for providing neutron cross sections for reactor and shielding calculations. Several recommended sets of derived data are also given. (S.F.L.)

154.

4276 (LA-DC-7851) MEASUREMENT OF AVERAGE CROSS-SECTION RATIOS IN FUNDAMENTAL FAST-NEUTRON SPECTRA. Grundl; J. A.; Hansen, G. E. (Los Alamos Scientific Lab., Univ. of California, N. Mex.). [1966]. Contract W-7405-eng-36. 20p. (CONF-661014-4). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From IAEA Conference on Nuclear Data, Paris, France.

Spectral indices (energy-sensitive average cross-section ratios) are presented for the fast neutron spectra at the center of bare and natural-uranium-reflected spheres of ^{239}Pu, ^{233}U, and ^{235}U. By means of energy-sensitive nuclear reactions, the six fast critical assembly spectra are compared (1) among themselves, (2) to corresponding fission-neutron spectra, and (3) to neutrons of a single energy. Experimental errors depend upon reproducibility of detecting systems, spectral distortions due to the presence of detecting systems, and purity of the monoenergetic and fission neutrons. Recommended excitation functions and estimated uncertainties are presented for each detector reaction in coarse group structures for computational checks. The observed spectral indices and detector excitations, together with their uncertainties, provide a set of values that must be met by the microscopic cross sections ^{239}Pu, ^{233}U, and ^{235}U via transport computation. (S.F.L.)

155.

5558 EVALUATION OF CROSS SECTIONS AND RESONANCE PARAMETERS FOR ^{235}U, ^{233}U, ^{239}Pu, AND ^{241}Pu BETWEEN Cd-CUTOFF AND 10 keV. Hennies, H. H. (INTERATOM, Bensberg, Ger.). Vienna, International Atomic Energy Agency, 1966, Preprint No. CN-23/5, 46p. (CONF-661014-21). DTIE.

From IAEA Conference on Nuclear Data, Paris.

Fission and capture cross sections between Cd-cutoff and 10 keV have been evaluated for the fissile isotopes, based on the available experimental differential cross sections, capture-to-fission ratios, and total fission and capture resonance integrals. In particular, the recommended cross-sections contained in the 1965 edition of BNL-325 were compared with some other evaluations and with more recent results of cross-section measurements using nuclear explosives in space and underground. As a result of this evaluation, it was found that even for the most intensively studied nucleus, ^{235}U, the uncertainties in the fission cross-section σ_f are about ±5% and in the capture cross section σ_c, ±20%. For ^{239}Pu, σ_f is known to ±10% and σ_c to ±25%. For ^{233}U and ^{241}Pu, the uncertainty in σ_f is ±20% and in σ_c ±30%. Different sets of resonance parameters for the resolved resonances, which are recommended in BNL-325 or by other research laboratories, were compared by calculating infinitely dilute total and partial resonance integrals. Calculations were performed with the resonance integral code TRIX-1 which was developed at Atomics International and is based on the single-level Breit Wigner model. The well-known limitations of this model for nuclei, such as the

fissile nuclei, with narrowly spaced interfering levels, results in calculated fission resonance integrals which are 10 to 20% too small when compared with integrated measured differential cross sections. This result was found for energy regions where the cross sections are sufficiently well-known to make such a comparison possible. Average parameters for the unresolved resonances have been extrapolated from resolved resonance parameters. Calculated partial resonance integrals, using these parameters, were compared with integrated measured cross sections where these are available. As no published capture cross-section measurements exist in the resonance region for ^{233}U, ^{239}Pu, and ^{241}Pu these cross-sections were calculated with the determined average parameters and appropriate statistical distributions. (auth)

156.

5535 (ORNL-P-2599) RATIO OF CAPTURE TO FISSION IN ^{239}Pu AT keV NEUTRON ENERGIES. Lottin, A.; Weston, L. W.; de Saussure, G.; Todd, J. H. (Oak Ridge National Lab., Tenn.). [1966]. Contract W-7405-eng-26. 17p. (CONF-661019-5). Dep. mn. CFSTI $3.00 cy, $0.65 mn.

From International Conference on Fast Critical Experiments and Their Analysis, Argonne, Ill.

The neutron capture to fission ratio, α, was measured for ^{239}Pu for neutron incident energies from 20 to 600 keV. A pulsed beam of neutrons was collimated on a sample of plutonium placed in the center of a large hydrogeneous gamma-ray scintillator poisoned with gadolinium. A capture event in the sample was characterized by a single pulse of the scintillator due to the cascade of capture gamma rays, while a fission event was characterized by a pulse due to the prompt fission gamma rays followed, a few microseconds later, by additional pulses due to the gamma rays produced when the thermalized fission neutrons were captured in the gadolinium of the scintillator. Below 100 keV the neutron energies were measured by the time-of-flight technique with a resolution of 7 nsec/m. Above 100 keV approximately monoenergetic neutrons were used. Similar measurements were also performed on ^{235}U and the results are included for completeness. The uncertainties in the values of α obtained are approximately 10%. The data are consistent with measurements performed by other laboratories. (auth)

157.

2584 (NAA-SR-Memo-11907) AN UPDATED SIXTEEN-GROUP HANSEN-ROACH STRUCTURE LIBRARY. Ottewitte, E. H. (Atomics International, Canoga Park, Calif.). May 12, 1966. Contract AT(11-1)-Gen-8. 14p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

The contents of a library of revised Hansen-Roach 16-group cross sections are summarized and are discussed. Modified Be and Zr cross sections are included together with ^{239}Pu cross sections and cross sections of other elements used in SNAP reactors. (D.C.W.)

158.

4322 USE OF INTEGRAL MEASUREMENTS AS SUPPLEMENTARY DATA IN NEUTRON CROSS-SECTION EVALUATION. Pazy, A.; Rakavy, G.; Reiss, Y.; Yeivin, Y. (Hebrew Univ., Jerusalem). Vienna, International Atomic Energy Agency, 1966, Preprint No. CN-23/15, 16p. (CONF-661014-28). DTIE.

From IAEA Conference on Nuclear Data, Paris.

The exact formulation of a method is given to improve the microscopic cross-section evaluation by means of integral data. Considering a function $\sigma(E)$ and a set of functionals of this function $\gamma_i[\sigma]$, it is proposed that the function $\bar{\sigma}(E)$, which minimizes the quadratic form $\int [(\sigma - \hat{\sigma})/(\Delta\hat{\sigma})]^2 w dE + \Sigma_{i=1}^{N} \{[\gamma_i(\sigma) - \hat{\gamma}_i]/[\Delta\hat{\gamma}_i]\}^2$ is the best estimate of $\sigma(E)$, where $\hat{\sigma}$ is the estimate obtained from the analysis of direct measurements of $\sigma(E)$, $\Delta\hat{\sigma}$ is the standard deviation of a single measurement, and $w(E)dE$ is the significance attributed to the estimate in the interval dE. The $\hat{\gamma}_i$ are the experimental values of $\gamma_i[\sigma]$, and $\Delta\hat{\gamma}_i$ are their standard deviations. Explicit expressions for $\bar{\sigma}(E)$ and $\Delta\bar{\sigma}(E)$ are derived, and various special cases and applications are discussed. To illustrate the method, data from a few simple systems are used to correct the appropriate cross sections. (auth)

159.

1010 (GA-7427) TIME-OF-FLIGHT MEASUREMENTS OF NEUTRON SPECTRA IN ^{235}U AND TUNGSTEN. Profio, A. E.; Young, J. C.; Crosbie, K. L.; Hackney, R.; Antunez, H. M.; Russell, J. L. Jr. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Sept. 22, 1966. Contract AT(04-3)-167. 31p. (CONF-661019-2). Dep. mn. CFSTI $2.00 cy, $0.50 mn.

From International Conference on Fast Critical Experiments and Their Analysis, Argonne, Ill.

The pulsed-source, time-of-flight method was used to measure neutron angular flux spectra to 15 MeV from a spherical ^{235}U metal critical assembly and spheres of tungsten. The object of the ^{235}U experiment was to test calculational methods for fast reactor spectra and the model for neutron inelastic scattering. The tungsten spectra were measured at 300°K and 800°K, as part of a current program to measure the effect of Doppler broadening on spectra. The principal features of the experimental technique are discussed, particularly time resolution, beam extraction, and detectors. The ^{235}U sphere measurements are presented and are compared with GAPLSN calculations. The spectrum is very close to that in a critical reactor, but there is a discrepancy between theory and experiment. The tungsten results show a clear difference between hot and cold, even to high energies. The difference can not be explained by the density change, and it is apparently the result of Doppler broadening scattering resonances. Calculations do not take this into account. (auth)

160.

5528 (AEET-257) THERMAL AND SIXTEEN GROUP CROSS SECTION DATA FOR ^{233}Pa, ^{234}U, AND ^{236}U. Singh, R. Shankar (Atomic Energy Establishment, Trombay (India)). 1966. 14p. Dep. mn.

The latest available nuclear data were collected for ^{233}Pa, ^{234}U, and ^{236}U, and recommended values of thermal cross sections and resonance integrals are given for these isotopes for use in thermal-reactor calculations in the analysis of thorium fuel cycles. Sixteen-group cross sections for studying fast systems based on the same fuel cycle were prepared. Accuracies in the data and the need for further measurements and improvements were specified. (auth)

161.

5456 (AEEW-R-491) THE F.D.2 GROUP AVERAGED CROSS SECTION SET FOR FAST REACTOR CALCULATIONS. Smith, R. W.; Rowlands, J. L.; Wardleworth, D. (Atomic Energy Establishment, Winfrith (England)). Aug. 1966. 67p. Dep. BIS $1.80. HMSO 9s.

The F.D.2 is a 33-group cross-section library produced for sodium-cooled fast reactor calculations. The data were derived mainly from the UKAEA Nuclear Data Library using the GALAXY program and a smooth weighting spectrum which approximates that of a large sodium-cooled fast reactor. Other sources were used to provide: resonance shielding factors for ^{235}U, ^{238}U, ^{239}Pu, and ^{240}Pu; sodium elastic moderation and transport cross sections and the capture cross section for the 3-keV resonance; elastic

moderation cross sections for oxygen, chromium, iron, and nickel; and data for substances that are not included in the Nuclear Data Library, or for which more recent evaluations are available. With the exception of the highest energy group, the group boundaries are at half-lethargy intervals based on an origin at 10 MeV. The top group covers energies above 3.68 MeV and has a ^{235}U fission-spectrum weighting. The remaining 32 half-lethargy groups cover the energy range down to 0.414 eV. The scatter matrix was limited to ten down-scatter groups with scattering to lower energies included in the scattering to the tenth down-scatter group. (S.F.L.)

4. REACTIVITIES

1956

1.

10534

AN UNREFLECTED U-235 CRITICAL ASSEMBLY. R. E. Peterson and G. A Newby (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 1, 112-25(1956) May.

An unreflected, spherical U^{235} critical assembly has been in operation at the Los Alamos Scientific Laboratory since August, 1951. A remotely controlled mechanical system is used to assemble subcritical components of the sphere, and reactivity is adjusted with U^{235} control rods. The maximum power level during sustained operation is about 1 kw. Investigations with the assembly include studies of the neutron spectrum, observation of the changes of reactivity produced by inserting foreign materials into the assembly, and determination of parameters such as the temperature coefficient of reactivity. In addition, experiments at reactivities above prompt critical have been carried out. The assembly has also been used as a source of short, high-intensity bursts of neutrons in the study of delayed neutrons following fission. (auth)

1957

2.

13161 KAPL-M-PLH-16

Knolls Atomic Power Lab., Schenectady, N. Y.

CALCULATION OF REACTIVITY COEFFICIENTS FOR HAFNIUM, CADMIUM, AND BORON FOR VARIOUS SAMPLE THICKNESSES. P. L. Hofmann. May 25, 1954. Decl. Mar. 9, 1957. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Reactivity coefficients of samples of Hf, Cd, and B of various thicknesses have been calculated by means of perturbation theory at the center and core edge of the clean PPA-18. Calculated and experimental results are compared. (auth)

3.

13955 LA-1708

Los Alamos Scientific Lab., N. Mex.

MATERIAL REPLACEMENT MEASUREMENTS IN TOPSY AND GODIVA ASSEMBLIES. L. B. Engle, G. E. Hansen, H. C. Paxton, J. C. Hoogterp, and D. S. Young. July 1954. Decl. May 9, 1957. 98p. Contract W-7405-eng-36. $15.30 (ph OTS); $5.40(mf OTS).

This report brings together an extensive accumulation of material replacement (danger coefficient) data for the various oralloy critical assemblies at Pajarito. Corresponding values of effective absorption and transport cross sections are derived. In certain favorable cases, inelastic scattering contributions to the effective absorption cross sections are estimated. Special applications of material replacement data include computation of the relationship between oralloy critical mass and concentration of a diluent, and hydrogen isotopic analysis of heavy water and heavy polythene. (auth)

4.

648 DP-48

Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.

REACTIVITY MEASUREMENTS IN A SUBCRITICAL PILE. R. C. Axtmann, G. Dessauer, and T. F. Parkinson. Nov. 1955. 20p. Contract AT(07-2)-1.

Experiments showed that there exists in a subcritical pile of low critical mass a range of multiplications that are high enough for precision testing of pile components and yet low enough for the rapid establishment of equilibrium in each test. Advantages of subcritical testing over the conventional "danger coefficient" method are discussed. (auth)

1958

5.

1073 LA-1525
Los Alamos Scientific Lab., N. Mex.
MATERIAL REPLACEMENT EXPERIMENTS: THEORY AND MEASUREMENTS FOR THE LADY GODIVA ASSEMBLY. Gordon E. Hansen and Clifford Maier. Apr. 23, 1953. Decl. Apr. 1, 1957. 58p. Contract W-7405-Eng-36. $9.30(ph OTS); $3.60(mf OTS).

The perturbation theory for material replacement experiments in a bare assembly is given through second order, thus permitting corrections for sample size. Computed flux and adjoint distribution functions are tabulated for the Lady Godiva assembly, enabling the observed danger coefficients for U^{238} and U^{235} to be compared with corresponding predicted values. Consistency of this data is checked by its use in three independent combinations, each yielding the same value for the effective fraction of delayed neutrons from fast fission. Reactivity contributions associated with inelastic scattering were calculated and comparison made between central danger-coefficient ratios and ratios for the Topsy assembly. Evaluation of transport cross sections by means of replacement measurements in Godiva is illustrated for the several elements C, Cu, and Au. (auth)

6.

2050 HW-27851
Hanford Atomic Products Operation, Richland, Wash.
EXPERIMENTAL RESULTS OBTAINED FROM TEST PILE REACTIVITY MEASUREMENTS ON PLUTONIUM. Final Report on Production Test No. 305-2-N. H. W. Lefevre and J. R. Triplett. Apr. 28, 1953. Decl. Feb. 19, 1957. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Reactivity measurements to allow determination of the amount of poisoning produced by the Cd and Ta sleeves and to determine the relative loss of Pu^{239} when removing Pu^{240} are reported. (F.S.)

1959

7.

21727 NP-7911
Iowa State Coll., Ames.
CHANGES IN REACTIVITY IN A SUBCRITICAL ASSEMBLY OWING TO LOCALIZED PERTURBATIONS. (thesis). Donald Miller Beck. 1959. 72p.

The reactivity changes in a subcritical assembly due to localized perturbations were investigated. The perturbations considered were the addition of fuel, poison, and the introduction of a void. Experimental and theoretical results were correlated to ascertain the feasibility of predicting reactivity changes in a subcritical assembly. (W.D.M.)

8.

9434 BNL-483(p.75-6)
Los Alamos Scientific Lab., N. Mex.
REACTIVITY COEFFICIENT DATA ON U^{233} IN FAST CRITICAL ASSEMBLIES. G. E. Hansen. p.75-6 [of] THORIUM–U^{233} SYMPOSIUM, SPONSORED BY THE UNITED STATES ATOMIC ENERGY COMMISSION AT BROOKHAVEN NATIONAL LABORATORY, JANUARY 9-10, 1958. 2p.

Data are presented on reactivity coefficient ratios obtained at the centers of Topsy (U reflected Oy), Godiva (Bare Oy), and Jezebel (bare Pu). (W.L.H.)

9.

17454
REACTIVITY EFFECTS OF LARGE VOIDS IN THE REFLECTOR OF A LIGHT-WATER-MODERATED AND REFLECTED REACTOR. A. B. Reynolds, K. M. Henry, and E. B. Johnson (Oak Ridge National Lab., Tenn.) and T. J. Thompson (Massachusetts Inst. of Tech., Cambridge). Nuclear Sci. and Eng. 2, No. 1, Suppl., 163-5(1959) June.

Reactivity effects of large voids in the reflector of the Pool Critical Assembly (PCA), an enriched-uranium, light-water-moderated and -reflected, pool-type reactor, were investigated experimentally and theoretically. The four principal effects measured were the variations of reactivity of voids at the center of a core face, void position on core face, and separation distance between void and core face. Superposability of the reactivity effects of multiple voids was studied. The effect of the largest void on the thermal neutron flux distribution was measured. (W.D.M.)

1960

10.

1093 AECU-4391
[Oak Ridge National Lab., Tenn.].
REACTIVITY EFFECTS OF LARGE VOIDS IN THE REFLECTOR OF A LIGHT-WATER-MODERATED AND -REFLECTED REACTOR (thesis). Albert Barnett Reynolds. June 1959. 309p. OTS.

Submitted to Massachusetts Inst. of Tech.

Reactivity effects of large voids in the reflector of the Pool Critical Assembly (PCA), an enriched-uranium, light-water-moderated and -reflected reactor, were investigated experimentally and theoretically. The three principal effects which were studied experimentally were: the variation of reactivity with the size of a void located at the center of one face of the core (including a void covering the entire face), the variation of reactivity with void position on the core face, and the superposability of the reactivity effects of voids. The effect

of the largest void on the thermal-neutron flux distribution was also measured. The experimental value for the reactivity divided by γ for a void covering entirely the largest face of a 23.6 × 20.5 × 14.4 in. core was (2.79 ± 0.12)%, where γ is the ratio of the effective delayed-neutron fraction to the actual delayed-neutron fraction. The calculated value of γ for this core was 1.22 ± 0.05. (auth)

11.

10876 KAPL-M-MLS-12
Knolls Atomic Power Lab., Schenectady, N. Y.
DISTRIBUTED REACTIVITY COEFFICIENTS IN A PMA SLAB CORE. MEASUREMENT AND ANALYSIS. N. E. French, R. F. Ruane, and M. L. Storm. Feb. 9, 1960. 50p. Contract W-31-109-Eng-52. OTS.

Measurements of the core distributed reactivity coefficients of moderator and structural materials that were made in a polyethylene moderated and reflected slab reactor (PMA-6) are described. In addition, the experiments are analyzed with the use of various cross section routines currently employed in nuclear design studies at this laboratory. The adequacy with which the WOX-type and 3-Group *MUFT-SOFOCATE* cross-section codes predict the experimental reactivity coefficient data is demonstrated. (auth)

1961

12.

32919 (APEX-705) REACTIVITY EFFECTS OF STRUCTURAL MATERIALS IN THE BEM-II B. K. V. Cooper and W. B. Henderson (General Electric Co. Flight Propulsion Lab. Dept., Cincinnati). Aug. 1961. Contracts AF33(600)-38062 and AT(11-1)-171. 27p.

Reactivity effects of 93.2% enriched oralloy, molybdenum, iron, niobium, 310 stainless steel, 304 stainless steel, Inconel X, FeCrAl, zirconium, nickel, nichrome V, and chromium clad with FeCrAl, in the form of foils in the center cell of the BEM-II B minimum void, beryllium moderated critical experiment are reported. Nineteen-energy-level diffusion calculations with cell corrections from both coarse and fine energy detail were correlated with the measurements. The fine energy detail improved the correlation, mainly because resonance self-shielding and flux depression are more adequately treated, and residual discrepancies for the most part can be attributed to inadequacies in the cross section data. (auth)

13.

7011 (ORNL-3016(p.3-5)) REACTIVITY WORTH OF THE CENTRAL FUEL ELEMENT IN THE BULK SHIELDING REACTOR I. G. deSaussure, K. M. Henry, Jr., and R. Perez-Belles (Oak Ridge National Lab., Tenn.).

The reactivity worth of a BSR-1 fuel element at the center of a critical lattice was experimentally measured by a pulsed-neutron-source technique. The value obtained for BSR-1 loading No. 78 was 6.1 ± 0.5 dollars. (auth)

14.

6599 REACTIVITY CONTRIBUTIONS OF VARIOUS MATERIALS IN TOPSY, GODIVA, AND JEZEBEL. L. B. Engle, G. E. Hansen, and H. C. Paxton (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 8, 543-69(1960) Dec.

An accumulation of reactivity contribution data is brought together for the various critical assemblies at Pajarito. Corresponding values of effective absorption and transport cross sections are derived, and relations between critical mass and volume fraction of diluents are obtained in terms of these cross sections. In some favorable cases, inelastic scattering contributions to the effective absorption cross sections are estimated. (auth)

15.

6598 PERTURBATION THEORY OF REACTIVITY COEFFICIENTS FOR FAST-NEUTRON CRITICAL SYSTEMS. Gordon E. Hansen and Clifford Maier (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng. 8, 532-42 (1960) Dec.

The first-order perturbation theory, or equivalently the interpretation of reactivity coefficients, is illustrated for fast neutron critical systems. A second-order perturbation theory for small size sample replacements is given and illustrated especially as it pertains to measurements of reactivity coefficients. (auth)

16.

21406 REACTIVITY AND ACTIVATION MEASUREMENTS AND CALCULATIONS IN HIGHLY ENRICHED U–Zr–H_2O CRITICAL ASSEMBLIES BETWEEN 65 AND 460°F. L. O. Herwig, W. F. Vogelsang, J. J. Kepes, E. L. Humez, W. A. Vogele, and T. M. Ryan (Westinghouse Electric Corp., Pittsburgh). Trans. Am. Nuclear Soc., 4: No. 1, 52-3(June 1961).

17.

8386 (CF-61-1-42) REVISED VERSION OF HFIR CRITICAL EXPERIMENT-2 (HFCE-2). P. R. Kasten and R. D. Cheverton (Oak Ridge National Lab., Tenn.). Jan. 16, 1961. 6p.

A listing and description is given of the experiments associated with the HFIR Critical Experiment-2. The primary experiments concern the reactivity of the bare core, reactivity worth of "gray" control plates, core-power

distribution, reactivity worth of "black" control plates, temperature coefficients of reactivity, and the island void coefficient of reactivity. The secondary experiments concern the reactivity of the fuel, and the reactivity worth of a "partial" gray plate. (auth)

18.

25595 (NAA-SR-Memo-2805) REACTIVITY IN SRE OF THE BETTIS ALLOYS. J. J. McClure (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). May 21, 1958. 4p.

The reactivities of several U–Nb–Zr alloys in the Sodium Reactor Experiment are compared with that of 2.8% enriched U fuel. The alloys considered have the following compositions: 15% Nb–70% U–15% Zr; 12% Nb–82% U–6% Zr; and 6% Nb–82% U–12% Zr. Each alloy is studied at U enrichments of 3, 7, and 11%, and the alloy enrichments necessary to give the same reactivity as the 2.8% enriched U fuel are found by interpolation. (T.F.H.)

19.

16754 (KAPL-M-JJS-2) REACTIVITY COEFFICIENTS IN THE UNPOISONED PMA-40 SLAB CORE. PART I. MEASUREMENTS. James J. Schultheis (Knolls Atomic Power Lab., Schenectady, N. Y.). Jan. 9, 1961. Contract W-31-109-eng-52. 50p.

Results of reactivity coefficient and flux distribution measurements in the unpoisoned PMA-40 slab core are described. The PMA-40 core is the first in a series of slab cores to be constructed in the KAPL Plastic Mockup Assembly (PMA) for evaluating analytical techniques and cross-section schemes. In addition to the distributed reactivity coefficients of U^{235}, aluminum, and polyethylene, the spatial variation in the reactivity coefficients along the short 12-in. dimension of the core is presented. The description of the core, measurements, and results presented is complete enough so that calculations can be performed without additional information. (auth)

20.

32666 (KAPL-M-JRT-1) REACTIVITY COEFFICIENTS IN PMA-47 A ZIRCONIUM–URANIUM, UNPOISONED SLAB CORE. PART I. MEASUREMENTS. J. R. Tomonto (Knolls Atomic Power Lab., Schenectady, N. Y.). Aug. 21, 1961. Contract W-31-109-Eng-52. 73p.

The results of measuring neutron flux distributions, distributed reactivity coefficients of U^{235}, polyethylene, and zircaloy as well as the spatially dependent reactivity coefficients of zircaloy, aluminum, and zirconium–boron in PMA-47 are reported. PMA-47 is the second of a series of cores constructed in the Plastic Mockup Assembly (PMA) to provide reactivity coefficients and criticality data in cores with simple geometry. It differs from the first core of the series (PMA-40) in that it contains a higher fuel density and utilizes zirconium as its structural material. The reactivity data will be used to test analytical techniques and to provide a comparison with related values for subsequent similar aluminum–uranium cores. (auth)

1962

21.

2668 (ANL-6392) IDAHO DIVISION SUMMARY REPORT, OCTOBER 1960 THROUGH MARCH 1961. (Argonne National Lab., Ill.). Contract W-31-109-eng-38. 75p.

Experimental Breeder Reactor-I. Three methods were used to determine the reactivity worth of the EBR-I lead cup reflector to be 5.4%. The feedback mechanism postulated for the resonant instabilities of Mark-II core is discussed in some detail. ZPR-III. Criticality and reactivity measurements for a wide variety of materials are reported for Assemblies 29 to 31 (uranium oxide) and 32 to 33 (stainless steel-uranium). BORAX-V. Current development and fabrication work carried out on the BORAX-V facility are described. Transient Reactor Test Facility. Results are presented for a 1010-Mw-sec temperature-limited transient in the facility and for the integrated and peak power as a function of reactivity addition. A temperature rise vs power graph was derived for calculating the energy release of transients. Gamma flux measurements are discussed. (D.L.C.)

22.

32260 (ANL-6582) TWO SPHERICAL FAST CRITICAL EXPERIMENTS (ZPR-III ASSEMBLIES 38 AND 39). J. C. Bates, W. G. Davey, P. I. Amundson, J. M. Gasidlo, J. K. Long, and W. P. Keeney (Argonne National Lab., Idaho Falls, Idaho). July 1962. Contract W-31-109-eng-38. 35p.

Two spherical versions of earlier cylindrical assemblies were used for a critical study of shape effects for fast reactor cores with volumes of 300 to 400 liters. Assemblies 24 (cylindrical) and 38 (spherical) had a high-density metallic U blanket, whereas the set of assemblies numbered 31 (cylindrical) and 39 (spherical) had a low-density U-fueled core (with steel and Al diluents) with a high-density blanket of depleted U. The main features of these assemblies are summarized. Reactivity coefficients of a small number of fissile and nonfissile materials were measured in both assemblies (38 and 39). In Assembly 38 the effects of environment, etc., upon fission rates measured with absolute fission chambers were investigated. Radial fission rate traverses in different directions were made in Assembly 39 to reveal any flux asymmetry due to heterogeneity of the core; no such effect was detected. (auth)

23.

2400 (GA-2313) REACTIVITY OSCILLATION MEASUREMENTS OF THE THORIUM DOPPLER COEFFICIENT IN THE HTGR CRITICAL FACILITY. J. R. Brown and J. B. Sampson (General Atomic Div., General Dynamics Corp., San Diego, Calif.). July 1, 1961. Contract AT(04-3)-314. 46p.

The Doppler coefficient, (1/I)(dI/dT), was measured for several HTGR fuel elements in the HTGR critical assembly at temperatures up to 650°K by a reactivity oscillation technique. The correction required because of a small effect upon neutron temperature caused by heating the central fuel element (the thermal base effect) was also determined experimentally. The coefficient, (1/I)(dI/dT), is about 3.6×10^{-4}/°C at 300 to 650°K. It increases slightly with thorium loading. (auth)

24.

15795 SOME MEASUREMENTS OF REACTIVITY IN A LIGHT-WATER MODERATED HIGHLY ENRICHED URANIUM ASSEMBLY. P. N. Cooper, K. Firth, M. Kerridge, R. F. Mathams, A. J. Salmon, and K. G. Stephens (Associated Electrical Industries Ltd., Aldermaston, Berks, Eng.). J. Nuclear Energy, Pts. A & B. Reactor Sci. and Technol., 16: 65-9(Feb. 1962).

A modified version of the inhour equation, which relates the reactivity to the rate of change of power of a reactor, was used. It takes into account the different probabilities of leakage from the reactor core during the slowing down of prompt and delayed neutrons. Using this equation the reactivity absorbed by a control element in a reactor was deduced. Good agreement was found with other methods of estimating the element effectiveness including critical mass measurements and sub-critical measurements with distributed absorbers. (auth)

25.

17234 (DP-641) PDP MEASUREMENTS OF COOLANT VOID REACTIVITY COEFFICIENTS. Albert E. Dunklee (Du Pont de Nemours (E. I.) and Co. Savannah River Lab., Aiken, S. C.). Mar. 1962. Contract AT(07-2)-1. 22p.

Measurements in the Process Development Pile (PDP) determined the effect on reactivity of removing all D_2O coolant from pressure tube fuel assemblies consisting of four concentric U metal tubes ranging in outer diameter from 1.06 to 3.08 in. and immersed in D_2O moderator. At a triangular lattice pitch of 9.33 in. between the centers of the fuel assemblies, the measured increase in the multiplication constant of an infinite lattice was 2.8%. This result confirmed earlier calculations and exponential experiments which indicated that this type lattice of nested U metal tubes would be unstable in a boiling-D_2O-cooled power reactor. (auth)

26.

30019 (NASA-TN-D-1322) CRITICALITY EFFECTS OF CENTRALLY LOCATED TUBES AND RODS OF ALUMINUM, IRON, AND TUNGSTEN IN A HOMOGENEOUS REACTOR. Daniel Fieno, Eugene Gunn, Clayton Barber, Thomas Fox, Donald Alger, and Robert Mueller (National Aeronautics and Space Administration. Lewis Research Center, Cleveland). Aug. 1962. 27p.

The NASA zero-power reactor, consisting of an unreflected cylinder containing a solution of uranyl fluoride salt in water, was used to study simple heterogeneous effects. This heterogeneity was introduced into the reactor as tubes or rods of aluminum, iron, and tungsten. For a given hydrogen- to uranium-235-atom ratio of the fuel solution, criticality was achieved by varying the height of the fuel solution contained in a 12-inch-inside-diameter tank. Criticality measurements of cylindrical void regions located on the axis of the reactor were also made. These measurements were analyzed by using 19 energy groups with a one-dimensional diffusion-theory code written for the IBM 704 computer. (auth)

27.

931 (ANL-6338) CRITICAL STUDIES OF A DILUTE FAST REACTOR CORE (ZPR-III ASSEMBLY 31). J. M. Gasidlo, J. K. Long, and R. L. McVean (Argonne National Lab., Ill.). Oct. 1961. Contract W-31-109-eng-38. 25p.

Critical studies were performed on a dilute, metallic, fast reactor core. The fuel was 38.6% enriched uranium and reduced-density aluminum was used to simulate sodium coolant. The measured critical mass was 463 kg of U^{235} in a 425-liter core. The main part of the experimental program consisted of measurements of the standard fission ratios, a number of central reactivity coefficients, and the prompt neutron lifetime. An additional series of experiments were performed to obtain the worths of aluminum, aluminum oxide, and sodium at various radial positions in the core. (auth)

28.

2397 (ANL-6336) CRITICAL STUDIES OF A 450-LITER URANIUM OXIDE FAST REACTOR CORE (ZPR-III ASSEMBLY 29). A. L. Hess, W. Gemmell, J. K. Long, and R. L. McVean (Argonne National Lab., Ill.). Nov. 1960. Contract W-31-109-eng-38. 46p.

Results of studies with ZPR-III Assembly 29, a mockup of a typical, dilute UO_2-fueled fast reactor, are reported. The assembly consisted of a 454-liter cylindrical core, blanketed with depleted uranium, with a critical mass of 421 kg U^{235}. Experiments included measurements of fission rates, material reactivity worths, and Rossi alpha. The results of multigroup calculations using the present Argonne cross section sets are presented, and discrepancies between experimental results and calculations are discussed. (auth)

Reactivities 29-34

29.

3837 (ORNL-3193(p.3-4)) THE REACTIVITY EFFECT OF AN AIR-FILLED CAVITY WITHIN A POOL CRITICAL ASSEMBLY CORE. E. B. Johnson (Oak Ridge National Lab., Tenn.).

It is shown experimentally that a positive void coefficient of reactivity exists in the core center-line of the Pool Critical Assembly. The assembly is fueled with BSR-I fuel elements, and the void is introduced in a Lucite container in order to keep the metal/water ratio constant. (T.F.H.)

30.

18612 (DC-59-7-700) REACTIVITY MEASUREMENTS DURING PHASE II TESTS WITH THE HOTCE REACTOR. R. M. Kinkaid, J. K. Kunze, and K. J. Skow (General Electric Co. Aircraft Nuclear Propulsion Dept., Idaho Falls, Idaho). July 1, 1959. Contract AT(11-1)-171. 14p.

The results of reactivity measurements made with the HOTCE reactor during phase II tests are given along with the significant test procedures that evolved during the experimentation. The reactivities corresponding to the asymptotic periods were determined from the in-hour relation. The worth of each test rod relative to the boron test rod was computed. (M.C.G.)

31.

4999 REACTIVITY MEASUREMENTS—PROBLEMS, METHODS, AND APPLICATION POSSIBILITIES (SUMMARY REPORT). P. Liewers (Zentralinstitut für Kernphysik, Rossendorf, Ger.). Kernenergie, 4: 593-618(Aug. 1961). (In German)

In this survey report on reactivity measurements, the kinetic behavior of a reactor is first considered, then the methods used for the measurement of the reactivity are reviewed. The application possibilities discussed are measurement of physical properties, static reactor parameters, kinetic reactor parameters, dynamic reactor parameters, and parameters which are necessary for the operation of the reactor. (J.S.R.)

32.

4999 REACTIVITY MEASUREMENTS—PROBLEMS, METHODS, AND APPLICATION POSSIBILITIES (SUMMARY REPORT). P. Liewers (Zentralinstitut für Kernphysik, Rossendorf, Ger.). Kernenergie, 4: 593-618(Aug. 1961). (In German)

In this survey report on reactivity measurements, the kinetic behavior of a reactor is first considered, then the methods used for the measurement of the reactivity are reviewed. The application possibilities discussed are measurement of physical properties, static reactor parameters, kinetic reactor parameters, dynamic reactor parameters, and parameters which are necessary for the operation of the reactor. (J.S.R.)

33.

28438 REACTIVITY EFFECTS OF MODERATOR EXPULSION IN AN ENRICHED, LIGHT WATER REACTOR. Jerome Lee Shapiro (Univ. of Michigan, Ann Arbor). Dissertation Abstr., 22: 1120-1(Oct. 1961).

An experimental and analytical investigation of the effect of removal of part of the water moderator on the reactivity of a small pool-type reactor is made. The experimental technique consists of bubbling air through one element of the core at a time. The air volume (which is considered to be a void) is determined by calibrating measurements made in an out-of-core gamma ray densitometer. Density perturbations made in individual (3" × 3") core elements produce void coefficients ($\equiv v\Delta k/k\Delta v$) varying from -0.316 at the center to -0.110 at the outer corner of a roughly square core. The average void coefficient produced by volumetrically averaging all the local coefficients is -0.190. Linearity of the reactivity effect with void concentration is observed up to the maximum void fraction introduced of 11 per cent. This result is compared with a diffusion-theory analysis that includes a comparison of two-group perturbation, using one-dimensional and two-dimensional eigenvalue-variation methods. This comparison shows that for uniformly distributed density changes, perturbation theory is simple and accurate within 5 per cent. It is also apparent that variation of the geometry (e.g., cylindricizing a parallelepiped) has very little effect on either uniformly distributed or localized reactivity effects. Effects of neutron streaming, neutron temperature changes, and changes in flux depression are shown to be negligible in the average water-density range ($\rho > 0.75$ gm/cc) examined. The most important feature not adequately accounted for by applying density-changed cross sections to one- or two-dimensional codes is the variation of transverse bucklings. This feature is examined with the aid of the Buckling Iteration subroutine of the MAGNUM code. The combination of buckling iteration with any of the two-group techniques considered adequately solves the distributed void coefficient problem. However, for localized voids this technique is not accurate since it assumes no radial variation of axial buckling (separability). The interior void coefficient is overestimated while the outer void coefficient is underestimated (both by about 15 to 30 per cent). On the basis of this evidence suggestions are made for modification of the buckling iteration procedure to allow radial variations of axial bucklings.

34.

19988 (KAPL-M-JRT-2) REACTIVITY COEFFICIENTS IN PMA-47. A ZIRCONIUM—URANIUM, UNPOISONED SLAB CORE. PART II. ANALYSIS. J. R. Tomonto (Knolls Atomic Power Lab., Schenectady, N. Y.). Jan. 8, 1962. Contract W-31-109-Eng-52. 46p.

The purpose of this report is to compare calculated criticality, reactivity coefficients, and flux distributions with related quantities measured in PMA-47. PMA-47 is

the second of a series of cores constructed in the Plastic Mockup Assembly (PMA) to provide criticality data and reactivity coefficients in cores with simple geometry. A comparison of results is made to PMA-40 which was the first in this series. The two cores differ in that PMA-47 contains a higher fuel density and utilizes Zr as its principle structural material. The calculations were performed using the D1G-WOXX and MUFT-KATE cross section routines with the KARE diffusion theory code. (auth)

1963

35.

4224 (ANL-6635(p.8-27)) LIQUID METAL-COOLED REACTORS. (Argonne National Lab., Ill.).

Experiments were conducted with ZPR-III assembly 41 which consists of 20% enriched U diluted in metal. Data on central fission ratios and sample-size effects in measurements of central- and radial edge-reactivity coefficients are presented. Other experiments are reported on blankets and reflectors and a rotary control rod mockup containing boron carbide in the radial blanket. In developmental work on the ZPR-VI and -IX, installation of the control wiring for the reactor-cell air conditioning and installation of the Ar storage system were completed. A summary of foil irradiation activities is included. Developmental programs concerning EBR-I, EBR-II, and FARET are summarized. (J.R.D.)

36.

26055 REACTIVITY AND KINETIC MEASUREMENTS WITH THE ADVANCED TEST REACTOR CRITICAL EXPERIMENT. Arland L. MacKinney and John W. Poston (Babcock and Wilcox Co., Lynchburg, Va.). Trans. Am. Nucl. Soc., 6: 47-8(June 1963).

37.

4342 (TID-17282) EXPERIMENTAL REACTIVITY EVALUATION OF THE ADVANCED TEST REACTOR CRITICAL EXPERIMENT. J. W. Poston and A. L. MacKinney (Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va.). Nov. 1962. For Ebasco Services, Inc., New York. Contract AT(10-1)-1075. 25p. (ATRCE-105)

The reactivity of the Advanced Test Reactor Critical Experiment (ATRCE) was evaluated with the core in the reference condition. Control cylinders were calibrated using determinations of positive reactor periods interpreted by the in-hour equation. The pulsed neutron technique was used to measure subcritical reactivities. Described are the reactivity evaluations for the hafnium neck shim rods, the outer shim control cylinders, and the borated Mylar tape used in the core region. The results of perturbation measurement of the reactivity coefficient of U^{235} are included. (H.D.R.)

38.

9882 SUBCRITICAL REACTIVITY MEASUREMENT BY A SOURCE-JERK METHOD. T. B. Ryves (National Physical Lab., Teddington, Middx., Eng.) and M. C. Scott. J. Nucl. Energy, Pts. A & B, 16: 455-63(Sept. 1962).

The theory of the source-jerk technique for the measurement of subcritical reactivity was refined to take account of harmonics present in the neutron flux distribution. The method was applied to the measurement of the subcritical activity in a large graphite-moderated uranium-fueled reactor. The subcritical reactivity was found to be [−2.63 ± 0.21] per cent ΔK/K. This value is about 10 per cent higher than estimates based on the subcritical approach to critical and on the measured interaction of shut-off rods. (auth)

39.

12765 (ORNL-3360(p.12-16)) EXPERIMENTS AND CALCULATIONS ON THE REACTIVITY EFFECTS OF SMALL FUEL DISPLACEMENTS. E. G. Silver, Z. M. Bartolome, et al. (Oak Ridge National Lab., Tenn.).

The reactivity effects associated with the motion of reactor fuel are being investigated in a calculational and experimental program at the BSF. Results indicate that negative reactivities result from all perturbations that decrease the homogeneity of the core, except when the perturbation adds fuel to high-importance regions, such as those adjacent to interior water volumes. The calculation methods appear to be inadequate to cope with the kind of small-dimension, large-magnitude perturbations represented by the experiments; however, experiments are being designed which can be more accurately represented in the calculations. (auth)

40.

29733 NEUTRON LIFETIME AND REACTIVITY MEASUREMENTS IN THERMAL REACTORS WITH THE HELP OF THE ROSSI-α-METHOD. Theodor Stribel (Siemens-Schuckertwerke AG, Garching, Ger.). Nukleonik, 5: 170-3(June 1963). (In German)

Rossi α-measurements on two thermal reactors, the Siemens Instruction Reactor (SUR) and the Siemens Argonaut Reactor (SAR), are reported. Methods and equipment are described and the results presented. For the SUR an average neutron lifetime of 48.75×10^{-6} sec (with $\beta_{eff} = 7.4 \times 10^{-3}$) was obtained, for SAR a lifetime of 173.6×10^{-6} sec (with $\beta_{eff} = 8.5 \times 10^{-3}$) was determined. Reactivity measurements on the basis of the α-method agreed satisfactorily with those obtained by the "rod-drop" method. The statement concerning the size of the time-dependent part of the frequency distribution, which is supplied by theory in the energy and place independent region, could not be confirmed. (tr-auth)

Reactivities 41-46

41

13337 REACTIVITY MEASUREMENT ON THE REFLECTED SEMI-HOMOGENEOUS CRITICAL ASSEMBLY BY PULSED NEUTRON TECHNIQUE. Kenji Sumita (Japan Atomic Energy Research Inst., Tokyo), Yoshihiko Kaneko, Riyosuke Kurokawa, and Tae Wan Park. Nippon Genshiryoku Gakkaishi, 4: 825-34(Dec. 1962). (In English)

Measurements are made of the prompt neutron lifetime and of the equivalent reactivity worths of control rods and the gap between two halves of the SHE-1 (Semi-Homogeneous Critical Assembly Core-1) by pulsed neutron technique. The decay constant α can be measured in the range from 7.19 to 147.7 sec^{-1} within 2.1%. The prompt neutron lifetime of this assembly is determined to be 1.21 ± 0.03 ms (assuming $\beta_{eff} = 0.0067$), and the equivalent reactivity worth of a Cd control rod to be 4.10 ± 0.13 $. However, the reactivity found from the value of α by the conventional formula, $\$ = (\alpha - \alpha_c)/\alpha_c$ is found to have a larger error than experimental one below about − 9 $ for SHE-1, because the change of the value of $k_{eff}\beta_{eff}/l$ with the degree of subcriticality is neglected. The theoretical validity of reactivity measurement in the reflected subcritical system by pulsed neutron technique is discussed. (auth)

42.

4254 (TID-17292) EXPERIMENTAL AND ANALYTICAL RESULTS FOR A ZIRCONIUM–URANIUM UNPOISONED CRITICAL SLAB ASSEMBLY. James R. Tomonto (Knolls Atomic Power Lab., Schenectady, N. Y.). [Sept. 20, 1962]. Contract [W-31-109-eng-52]. 10p.

Reactivity coefficients, criticality data, and flux distributions were measured in the Plastic Mockup Assembly for an unpoisoned, slab reactor. The core was constructed in two sections, one stationary and the other on an elevator platform. The fuel was 20 wt % uranium enriched to 93% U^{235} in a zirconium matrix plate. Reactivity coefficients were measured on both a position dependent and core distributed basis. The reactivity change associated with a particular material substitution was evaluated by means of doubling time measurements and were in the range from 10 to 20¢. The results were tested with two cross-section schemes. (M.C.G.)

1964

43.

3119 BUCKLINGS, DISADVANTAGE FACTORS AND δ^{28} MEASUREMENTS IN SOME UNDERMODERATED SLIGHTLY ENRICHED CORES. Q. L. Baird and A. R. Boynton (Argonne National Lab., Ill.). Trans. Am. Nucl. Soc., 6: 248(Nov. 1963).

44.

15471 (GA-4838) MEASUREMENTS OF SUBCRITICAL REACTIVITY USING A PULSED-NEUTRON SOURCE. J. R. Brown (General Atomic Div., General Dynamics Corp. San Diego, Calif.). Jan. 15, 1964. Contract AT(04-3)-167. 24p. (CONF-137-68)

From American Nuclear Society Meeting, New York, Nov. 1963.

Pulsed neutron source measurements were made for a series of subcritical to critical loadings on a bare U^{235} water assembly in slab geometry. For each loading, the assembly was repetitively pulsed, and the time distribution of thermal neutrons in the core was recorded on time analyzers. From these data, the reciprocal die-away time (α), the delayed neutron background, and the reactivity of the system were determined. Measurements at various positions in the assembly verified that spatial modes do not affect the reactivity determination. (auth)

3124 MEASUREMENTS OF SUBCRITICAL REACTIVITY USING A PULSED-NEUTRON SOURCE. J. R. Brown, J. C. Young, J. R. Beyster, D. H. Houston, and W. R. Mowry (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Trans. Am. Nucl. Soc., 6: 284-5(Nov. 1963).

45.

3353 THE DETERMINATION OF k_∞ FROM MEASUREMENTS ON A SMALL TEST SAMPLE IN A CRITICAL ASSEMBLY. D. D. Lanning, R. E. Heineman, I. Kaplan, and T. J. Thompson (Massachusetts Inst. of Tech., Cambridge). Trans. Am. Nucl. Soc., 6: 247-8(Nov. 1963).

46.

35083 THE VOID AND HOLE EFFECT IN D_2O MODERATED REACTOR. Yasushi Nakai (Mitsubishi Atomic Power Industry Inc., Saitama, Japan), Osamu Sugimoto, Kinichi Hagioya, Mitsuo Yokota, Yasuhiko Miyasaka, Isao Arase, Shizuo Kikkawa, and Denziro Nemoto. J. Nucl. Sci. Technol. (Tokyo), 1: 11-17(Apr. 1964).

The reactivity change due to an increase in the radius of an empty hole was measured in a heavy water-moderated reactor, and some results differing from experiments with ZEEP were obtained. It was concluded that the streaming in a hole is not so effective for reactivity. In measuring neutron flux in a void, a flat thermal neutron flux distribution was obtained; and it was concluded that the neutrons leaking through the empty hole or the void do not consist of thermal neutrons, but of fast neutrons for the most part. The experimental result of reactivity change due to the void location in the core indicates that the relation between the void location and the reactivity change is independent of the neutron flux distribution. (auth)

47.

21613 (HW-79960) REACTIVITY MEASUREMENTS OF MIXED OXIDE FUEL ELEMENTS. V. O. Uotinen and L. C. Schmid (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Dec. 11, 1963. Contract [AT(45-1)-1350]. 8p.

The relative worths of six PuO_2--UO_2 fuel elements and a Pu–Al fuel element in the center of the Plutonium Recycle Critical Facility were determined from positive period data. Five of the PuO_2–UO_2 elements will go into the PRTR; the sixth will be retained as a standard. The worths of the five elements relative to the standard, D_2O, and Pu–Al were obtained. The variation in worth relative to D_2O of the five elements was 1%. An inconsistency (within 1%) between the measured worths and the PuO_2 content of the elements was observed. The use of total reactivity worth as a criterion indicates that the fabrication of the PuO_2–UO_2 elements was very uniform. (D.C.W.)

48.

3094 (CONF-187-12) REACTIVITY WORTH OF TRANSVERSE GAPS. S. Weinstein and F. Feiner (Knolls Atomic Power Lab., Schenectady, N. Y.). Nov. 1963. Contract W-31-109-eng-52. 6p.

From American Nuclear Society Meeting, New York, Nov. 1963.

Fuel blocks of compacted ZrO_2 and UO_2 powder in paraffin were added axially to two unreflected assemblies (20 × 16 in. and 20 × 24 in. cross sections). The separation of the assembly halves when criticality occurred was determined, and the reactivity worth at a given critical gap size was obtained by bringing the halves slightly closer together and observing the reactor period. The core with the larger cross section had a maximum gap twice that of the smaller core. Diffusion theory treatment agreed with the data reasonably well for small gap thickness. (D.C.W.)

3113 REACTIVITY WORTH OF TRANSVERSE GAPS. S. Weinstein and F. Feiner (Knolls Atomic Power Lab., Schenectady, N. Y.). Trans. Am. Nucl. Soc., 6: 221(Nov. 1963).

1965

49.

3503 (AEEW-M-441) ANALYSIS OF THE RESULTS OF REACTIVITY CHANGE MEASUREMENTS ON PLUTONIUM–URANIUM FUEL ELEMENTS. R. C. Bannerman, H. H. W. Pitcher, and R. Richmond (United Kingdom Atomic Energy Authority. Reactor Group. Atomic Energy Establishment, Winfrith, Dorset, England). July 1964. 42p. Dep.; $1.20(BIS) 6s. (HMSO).

Reactivity change measurements were carried out in the graphite-moderated reactor HECTOR on plutonium–uranium fuel elements with a low ^{240}Pu content. The aim of the work was to develop a method in which measurements on small fuel samples (typically 18 in. long) could be used to check the validity of proposed methods of calculation for plutonium-bearing lattices. The experiments were carried out in a variety of lattices and demonstrated that at a lattice pitch of 8.5 in., which was used for the majority of the measurements, the results were not significantly affected by the lattice environment. In particular, the results showed that it is not necessary to use plutonium–uranium fuel in the lattice surrounding the sample but that uranium fuel with similar properties is adequate. The accuracy obtained corresponded to an error of 0.5% in k_∞ for a lattice containing the plutonium–uranium fuel. The results of the measurements were compared with the predictions of the TRACER method of lattice calculation. It was found that, at room temperature, there was no significant difference between the measured and calculated values of k_∞ but that, at temperatures in the range 400 to 450°C, the calculated value of k_∞ was about 1½% lower than the measured value. (auth)

50.

1937 (PAEC(D)RE-649) REACTIVITY EFFECT OF FUEL ADDITION IN THE PRR-1. Z. M. Bartolome and C. A. Tejada (Philippines. Atomic Energy Commission. Atomic Research Center, Manila). 1964. 4p. Dep.(mn).

Results are given of reactivity gains in PRR-1 due to fuel addition or removal at various core positions. (R.E.U.)

51.

1893 (TID-21251) MEASUREMENT OF HBWR-II CORE PARAMETERS IN THE NORA REACTOR. K. Bryhn-Ingebrigtsen, N. L. Snidow, A. Myking, and H. Smidt Olsen (Norway. Institutt for Atomenergi, Kjeller). Aug. 1964. Contract [AT(30-1)-2607]. 29p. (KR-70; NORA-8). Dep. (mn); $2.00(cy), 1(mn) OTS.

Some experimental results for NORA in-pile tests of enriched fuel clusters for the second loading of the Halden Boiling Water Reactor are presented. Measurements were made of thermal-neutron flux distributions, moderator level coefficient of reactivity, core importance function, flux depression by an SS thimble, and void effects. (R.E.U.)

52.

17311 REACTIVITY MEASUREMENTS IN THE ADVANCED EPITHERMAL THORIUM REACTOR (AETR) CRITICAL EXPERIMENTS. S. G. Carpenter (Atomics International, Canoga Park, Calif.). Nucl. Sci. Eng., 21: 429-40(Apr. 1965).

Reactivity effects of various materials of interest to epithermal and fast reactors were measured in a series of neutron spectra with median energies of neutrons causing fission ranging from 422 kev to 8.5 ev. An oscillator technique capable of sensitivities of 10^{-6} Δk was used. In addition, a power-history or reactivity-vs-time method was developed for fast, but less sensitive reactivity measurements. It was used to obtain radial reactivity traverses and rod calibrations and for other routine reactivity measurements. (auth)

Reactivities 53-63

53.

29024 EFFECT OF DISTRIBUTION OF FISSILE MATERIAL ON CRITICAL MASS. H. K. Clark (E. I. du Pont de Nemours and Co., Aiken, S. C.). Trans. Am. Nucl. Soc., 200-1(May 1965).

54.

29555 PRECISION REACTIVITY MEASUREMENT FACILITY AT ARGONNE NATIONAL LABORATORY. R. C. Doerner (Argonne National Lab., Ill.). Trans. Am. Nucl. Soc., 8: 261-2(May 1965).

55.

29550 INTERPRETATIONS OF REACTIVITY MEASUREMENTS ON CADMIUM-SHIELDED SAMPLES. W. K. Foell and J. J. Scoville (Phillips Petroleum Co., Idaho Falls, Idaho). Trans. Am. Nucl. Soc., 8: 257-8(May 1965).

56.

29556 THE RELIABILITY AND PHYSICAL INTERPRETATION OF MEASUREMENTS IN A REACTIVITY MEASUREMENT FACILITY. S. B. Gunst, J. C. Connor, E. D. McGarry, and D. E. Conway (Westinghouse Electric Corp., West Mifflin, Pa.). Trans. Am. Nucl. Soc., 8: 262-3(May 1965).

57.

24040 VOID REACTIVITY MEASUREMENTS IN TWO REACTOR LATTICES OF UO_2 ROD CLUSTERS IN D_2O. F. W. A. Habermann. Thesis, Amsterdam, Free Univ., 1965. 79p.

For understanding the characteristics of a boiling water reactor measurements were carried out of various coolant void reactivity effects in the zero power facility NORA at room temperature and in the Halden BWR between 55°C and 220°C. The measurements in NORA were carried out with a core of 7-rod UO_2 cluster fuel elements of the HBWR. The shroud of one fuel element was subdivided in sections in which the water level could be depressed by air. The temperature dependence of the void effect has been determined by removing coolant from one UO_2 cluster fuel element in the HBWR. The coolant channel was voided to various depths by depressing the D_2O level in it by means of argon under pressure. (auth)

58.

29542 CRITICAL FACILITY MEASUREMENTS OF THE SODIUM-VOID COEFFICIENT IN A LARGE DISC-SHAPED FAST-REACTOR CORE. F. H. Helm, W. Y. Kato, G. W. Main, H. H. Meister, and G. K. Rusch (Argonne National Lab., Ill.). Trans. Am. Nucl. Soc., 8: 245-6(May 1965).

59.

14283 (KY-L-258(Rev.I)) THE DENSITY EFFECT—IN REGARDS TO THE MINIMUM CRITICAL ENRICHMENT OF AQUEOUS HOMOGENEOUS SOLUTIONS. O. W. Hermann (Paducah Gaseous Diffusion Plant, Ky.). Jan. 16, 1962. Contract [W-7405-eng-26]. 6p. Dep.(mn); $1.00 (cy), 1(mn) OTS.

It is shown that the minimum critical enrichment of uranium in ^{235}U in aqueous homogeneous systems is independent of density. It is concluded that it should be possible to handle homogeneous or nominally homogeneous uranium salts without regard to density, so long as the minimum critical ^{235}U enrichment is not exceeded. (T.F.H.)

60.

29796 REACTIVITY MEASUREMENTS IN HECTOR. G. H. Kinchin (United Kingdom Atomic Energy Authority, Winfrith, Eng.). Trans. Am. Nucl. Soc., 8: 260-1(May 1965).

61.

3736 REACTIVITY VARIATION DURING STARTUP IN NUCLEAR ROCKETS. Winston W. Little, Jr., Edward A. Mason, and K. F. Hansen (Massachusetts Inst. of Tech., Cambridge). Trans. Am. Nucl. Soc., 7: 380-1(Nov. 1964).

62.

3676 FAST-REACTOR REACTIVITY COEFFICIENTS; THE EFFECT OF EXCESS REACTIVITY. W. B. Loewenstein (Argonne National Lab., Ill.). Trans. Am. Nucl. Soc., 7: 249-50(Nov. 1964).

63.

40158 (UNC-5123) SODIUM VOID COEFFICIENT OF REACTIVITY. A REVIEW AND STUDY OF ANALYTICAL WORK. Monroe, C. W. (United Nuclear Corp., White Plains, N. Y. Development Div.). May 1965. Contract AT(30-1)-2832. 126p. Dep. mn; CFSTI, $4.00 cy, $1.00 mn.

It is disclosed that the most widely accepted method for determining sodium reactivity effects employs a standard synthesis technique using diffusion theory in one-dimensional cylinder-and-slab geometries. The majority of multigroup diffusion theory codes used employ an option whereby group-and-region-dependent bucklings may be specified as input quantities, facilitating a relatively accurate iterative procedure. Effective group cross sections are derived from many sources of data; the high energy groups usually employ the cross sections of Yiftah, Okrent and Moldauer. Whenever possible, the most recent energy-dependent data for cross sections, $\nu(E)$, and the capture-to-fission ratios of fissile isotopes were used. A refined treatment of elastic scattering resonances (the ELMOE code) has established the necessity of using composition-dependent cross sections. Computations using several cross section sets indicate trends in the sodium void effect for a variety of conditions. Conclusions from these results are presented. (auth)

64.

10490 (UCRL-7696) A STUDY OF BERYLLIUM OXIDE MODERATED CRITICAL ASSEMBLIES CONTAINING VARIOUS METAL ABSORBERS. John R. Morton, III, James M. Plowaty, Joseph Petruzzi, and Loren Gardner (California. Univ., Livermore. Lawrence Radiation Lab.). Feb. 5, 1964. Contract W-7405-eng-48. 19p. Dep.(mn); $1.00(cy), 1(mn) OTS.

Reactivity worths of various metal plates were measured in a quasi-homogeneous enriched uranium-beryllium oxide subcritical assembly by means of the pulsed neutron method. The materials studied included nickel, iron, cobalt, gold, Hastelloy R-235, and René-41. The equivalences were determined between the various absorbers and fueled core material. Based upon the measurements it was concluded that the prompt lifetime is relatively insensitive to massive localized absorbers. (auth)

65.

3511 (CEA-R-2477) EXPERIENCES CRITIQUES DANS AQUILON PORTANT SUR DES COMBUSTIBLES LEGEREMENT ENRICHIS EN URANIUM 235 ET EN PLUTONIUM. (Critical Experiments in Aquilon With Fuels Slightly Enriched in Uranium 235 or in Plutonium). Roger Naudet, Michel Chabrillac, Guy Ledanois, and Pierre Lourme (France. Commissariat a l'Energie Atomique. Centre d'Etudes Nucleaires, Saclay). Aug. 1964. 27p. Dep.(mn).

Reactivity comparisons were made in Aquilon II between geometrically identical lattices differing only by the composition of the fuel. The fuel elements consist of metallic uranium single rods with either slight differences of the isotopic composition (0.69-0.71-0.83-0.86 per cent of uranium 235) or slight additions of plutonium (0.043 per cent). Five lattices pitches were used, in order to produce a large variation of spectrum. Two additional sets of plutonium fuels are prepared to be used in the same conditions. The double comparisons: natural-enriched versus natural-enriched plutonium, are made in such a way that a very precise interpretation is permitted. The results are perfectly consistent, which seems to prove that the calculation methods are convenient. Further it can be inferred that the usual data, namely for the ratio of the η of ^{235}U and ^{239}Pu seem reliable. (auth)

66.

45538 EFFECT OF A VOIDED GAP ON REACTIVITY COEFFICIENT FOR VOIDING CORE SODIUM IN A FAST REACTOR, FOR VARIOUS GEOMETRIES. Noderer, L. C.; Visner, Sidney (Combustion Engineering, Inc., Windsor, Conn.). 13p. (CONF-654-94). Gmelin, AED-CONF-64-219-138.

From American Nuclear Society Meeting, San Francisco, Nov.-Dec. 1964.

It is shown that a permanently voided gap between the core and blanket in a fast reactor substantially increases the leakage effect when sodium is removed from the core by decreasing the return probability of neutrons from the blanket to the core so that a negative voiding coefficient for the core sodium can be attained. Since the neutrons which are not returned to the core are nearly all absorbed in the blanket, there is no loss in overall breeding ratio. (L.B.S.)

67.

29551 NEUTRON-SCATTERING EFFECTS IN STATIC REACTIVITY MEASUREMENTS IN THE ARMF. J. W. Rogers, E. Fast, and D. A. Millsap (Phillips Petroleum Co., Idaho Falls, Idaho). Trans. Am. Nucl. Soc., 8: 258 (May 1965).

68.

29562 MEASUREMENT OF REACTIVITY NEAR DELAYED CRITICAL BY THE PULSED-SOURCE METHOD. S. A. Scott and J. R. Wolberg (Technion-Israel Inst. of Tech., Haifa). Trans. Am. Nucl. Soc., 8: 278(May 1965).

69.

33574 THE MEASUREMENT OF REACTIVITY IN MULTIREGION SUBCRITICAL SYSTEMS BY THE PULSED NEUTRON TECHNIQUE. Sherwin, J.; Leng, J. H. (United Kingdom Atomic Energy Authority, Sellafield, Cumb, Eng.). Vienna, International Atomic Energy Agency, 1965, Paper SM-62/25, 37p. (CONF-650504-31)

From IAEA Symposium on Pulsed Neutron Methods, Karlsruhe, Ger.

The prompt decay of a thermalized burst of neutrons in a multiregion subcritical system is examined using a two-group diffusion theory. It is shown that a relation can be established between the prompt decay constant of the fundamental mode and the effective multiplication factor of the system in terms of two parameters defined as the reactor response coefficient and the prompt decay correction and which depend to a large extent on the spatial flux patterns within the system. For the uniform bare system, the response coefficient can be identified with the mean neutron lifetime in the system; for a multiregion system it represents a compound of the lifetimes in each region weighted by perturbation type integrals. The second parameter, the decay correction, can have no physical meaning in that it arises from an attempt to relate the two scales of reactivity involved; that is the scale using the prompt decay constant and the scale derived using the effective multiplication factor. The properties of these parameters are examined here with reference to an enriched-uranium, graphite-moderated reactor consisting of a uniform core and reflector, and it is shown that the two parameters are not uniquely defined by the reactivity of the system, but depend on the method chosen to make the system subcritical. Two sets of measurements are treated by the theory. In one, the Windscale Advanced Gas-Cooled Reactor was shut down by a uniform poison. In the other, the reactivity of a core in the zero-energy reactor HERO was varied by altering the loaded radius. The neutron pulse measurements are all shown to be in good agreement with more conventional methods of reactivity determination. Some discussion of the experimental techniques and the difficulties encountered in the graphite-moderated systems is presented. (auth)

70.

17227 ON THE MEASUREMENTS OF SPATIAL DISTRIBUTION OF REACTIVITY COEFFICIENT. Nobuyuki Shirai (Meidensha Electric Mfg. Co., Ltd., Japan). Meidensha Jiho, No. 51, 25-32(Oct. 1963). (In Japanese)

The spatial distribution of reactivity coefficients in the epithermal region was measured in a graphite-moderated critical assembly (SHE). Unlike the foil activation method, the method used permits free selection of a sample regardless of the activation of nuclei, the mode of decay, and the workability of a foil. If the distribution of neutron fluxes is measured with foils of the same sample, the accompanying function (ϕ) can be estimated from the measurement of the reactivity coefficient. To improve the precision of measurements and shorten the time required for measurement, the sample contained in a Cd filter was pulled out of the core of the assembly by a manipulating system and the reactivity coefficient was measured radially from the core towards the periphery by the period method. In the investigation of the change in the reactivity coefficient due to the neutron absorption in the various energy regions, the following elements were used as samples: Au and In for the low energy region, I and Br for the medium energy region, and Co and Mn for the high energy region. The observed values of reactivity coefficients were compared with the theoretical. Both agreed very well in the core of the assembly, but in the reflector, the theoretical values were smaller than the observed. **(Nucl. Sci. Abstr. Japan, 3: (Sept. 1964))**

71.

8602 (ORNL-3691(p.89-98)) REACTIVITY EFFECTS OF SMALL FUEL DISPLACEMENTS FOR PLATE-TYPE REACTORS. E. G. Silver and S. B. Johnson (Oak Ridge National Lab., Tenn.).

A slab-geometry plate-type critical experiment was constructed to measure the reactivity effects of small perturbations introduced by fuel-element displacements. These perturbations were calculated by both diffusion-theory and transport-theory codes, with careful attention to sources of error in the calculation model, in order to test the relative abilities of the two calculation methods to predict the magnitude of such perturbations. The transport-theory results were in rather good agreement with the experiment, whereas the diffusion theory was far less accurate in its predictions. (auth)

72.

17313 NEUTRON-IMPORTANCE MEASUREMENTS IN THE ADVANCED EPITHERMAL THORIUM REACTOR (AETR) CRITICAL EXPERIMENTS. Robert J. Tuttle (Atomics International, Canoga Park, Calif.). Nucl. Sci. Eng., 21: 451-62(Apr. 1965).

The variation of neutron importance with energy and position was investigated in a multiregion critical assembly having a series of test regions typical of slightly epithermal to fast power reactors. Values of neutron importance at the center of the test regions were measured using neutron sources and a reactivity oscillator. The variation of neutron importance with position was determined using neutron sources in conjunction with a dynamic reactivity measurement technique. Analysis of data from similar beryllium- and carbon-moderated test regions indicates the significance of the Be(n,2n) reaction. The neutron sources used were Po–Be, Po–^{11}B, Po–CaF_2, Po–^{7}Li, Mock Fission and Sb–Be. Published source spectra were used in the analysis of the data; the Mock-Fission-source spectrum was determined by gamma spectroscopy. Relative yields of these sources were determined by calibration in a manganese sulfate bath. (auth)

73.

16060 (AROD-4434:2) THE STRUCTURE AND ANNEALING BEHAVIOR OF EXPLOSIVELY SHOCKED MgO CRYSTALS. Final Report. Mark J. Klein (Battelle Memorial Inst., Columbus, Ohio). Jan. 5, 1965. 13p.

The structure of explosively shocked magnesium oxide single crystals was investigated. Although ionic crystals with the rock-salt structure, such as MgO have been used as models to study the basic processes involved in plastic flow at low deformation rates, little is known about their response to high-velocity deformation. In this investigation, the structural changes induced in MgO crystals by plane shock waves were studied by etch-pit metallography and by electron-spin resonance. The experimental shock-loading procedures were designed to induce high dynamic pressures in MgO crystals for subsequent structural studies. (auth)

74.

1419 SOME COMMENTS ON THE ROSSI-α EXPERIMENT. Bryce, Donald H. (Kaman Nuclear, Colorado Springs, Colo.). Nukleonik, 7: 428(Aug. 1965).

The discrepancy between the accepted theoretical conditional detection probability for prompt neutrons and the experimental behavior lies in the experimental observation that the multiplying coefficient associated with the time-dependent portion of the prompt neutron conditional detection probability is independent of the prompt neutron population decay constant. An equation is derived for the prompt neutron conditional detection probability that expresses the dependence of the initial count upon the occurrence of the ancestor fission, thus directly relating the counting events at the two times of interest to their common antecedent. Experimental verification of this equation was obtained from the results of a series of experiments performed on the Cornell University zero power reactor. **(J.S.R.)**

75.

46995 PLUTONIUM IN NUCLEAR REACTORS. III. THE COEFFICIENT OF REACTIVITY OF Na IN LARGE FAST OXIDE REACTORS. Bucci, P. (CAMEN, Pisa, Italy). Ing. Nucl., 7: 30-41(Jan.-Feb. 1966). (In Italian).

The coefficient of reactivity of Na in Na-cooled fast Pu reactors is considered. The two factors most affecting the reactivity coefficient are the concentration of the fissile material and the presence of neutron capturing materials. The behavior of structural materials with respect to Na cooling is discussed. The nuclear and mechanical characteristics of stainless steel, Ti, Zr, Mo, W, Ta, Nb, and V are discussed. (J.S.R.)

76.

6279 (AI-65-225) TUNGSTEN RESONANCE INTEGRALS AND DOPPLER COEFFICIENTS. Quarterly Progress Report No. 1, July-September 1965. Carpenter, S. G.; Otter, J. M.; Paschall, R. K.; Royden, H. N. (Atomics International, Canoga Park, Calif.). Oct. 29, 1965. Contract NAS 3-7982. 21p. (NASA-CR-54769).

Axial flux and source traverses were made in the central void in the SGR-CA, the expected cosine shapes being obtained. Reactivity measurements under caumium were made on a number of samples, including natural tungsten, gold, lead, depleted uranium, aluminum, stainless steel, and lavite. The results indicate that a sensitivity of better than 0.01 cents is obtained for a single reactivity measurement taking 100 sec. This sensitivity is adequate for the resonance-integral measurements, and can readily be improved as required for Doppler measurements by taking data over a longer period of time. Calculated reactivities are consistently higher than those measured; however, the ratios of various calculated worths agree reasonably well with the corresponding ratios of experimental worths. The use of real and adjoint fluxes calculated with the cadmium sleeve taken into account should improve agreement with experiment. These results also indicate that reactivity effects associated with scattering and structural materials are negligible. Measurements with and without cadmium end caps have demonstrated that end effects are quite small. (auth)

77.

14082 PULSED NEUTRON REACTIVITY MEASUREMENTS IN THE NEAR-CRITICAL STATE. Einfeld, K.; Hennies, H.; Memmert, G.; Ullrich, W. (Interatom, Bensberg, Ger.). pp 299-312 of STI/PUB/104(Vol. 2).

Measured and calculated thermal neutron decay constants λ are reported for various 5% enriched zirconium hydride moderated lattices, using UO_2 and $ZrH_{1.7}$ rods in an aluminum matrix. By performing measurements in the critical and in near-critical states values of β/l could be determined. The resulting β/l values were used to calculate reactivity worths of different absorber rods and absorber arrangements from the measured decay constants. Several measurements were evaluated applying the modified pulsed source technique of Garelis and Russell. (auth)

78.

14075 SURVEY OF PULSED NEUTRON SOURCE METHODS FOR MULTIPLYING MEDIA. Garelis, E. (General Electric Co., Pleasanton, Calif.). pp 3-23 of STI/PUB/104(Vol. 2).

The methods by which the reactivity is derived from the α-measurement, e.g., the α-delayed critical measurement and the recent $(k\beta/l)$ method, are discussed. The fundamental modal treatments are examined in the light of the theory of the pulsed neutron source techniques as developed for the $(k\beta/l)$ model. The implications of the pulsed neutron source theory to obtain precise decay constants and suitable data for the analysis of pulsed systems are considered. Experimental work is reviewed that shows the advantages as well as the limitations of the $(k\beta/l)$ technique. The use of pseudo-random impulse response methods with cross correlation between the input and output for the determination of the Green's function of a multiplying assembly is also discussed. It is shown that the information obtained by the pseudo-random method is identical to that obtained from the repetitively pulsed method. Thus, this makes it possible to apply the methods developed for the repetitive to that of the pseudo-random technique. (auth)

79.

24709 (RPI-328-52) SUBCRITICAL REACTIVITY DETERMINATIONS—COMPARISON OF EXPERIMENTAL METHODS. Gozani, Tsahi (Rensselaer Polytechnic Inst., Troy, N. Y. Dept. of Nuclear Engineering and Science). [1965]. Contract AT(30-3)-328. 7p. (CONF-660606-1). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From American Nuclear Society Meeting, Denver.

Four methods of measuring the reactivity of subcritical systems are discussed and compared. (D.C.W.)

34964 SUBCRITICAL REACTIVITY DETERMINATIONS-COMPARISON OF EXPERIMENTAL METHODS. Gozani, Tsahi (Rensselaer Polytechnic Inst., Troy, N. Y.). Trans. Amer. Nucl. Soc., 9: 236-7(June 1966).

80.

3384 NUCLEAR REACTIVITY MEASUREMENTS OF IRRADIATED URANIUM DIOXIDE. Guon, J. (Atomics International, Canoga Park, Calif.). Trans. Amer. Nucl. Soc., 8: 384-5(Nov. 1965).

81.

3431 ON THE INTERPRETATION OF REACTIVITY MEASUREMENTS OF REACTOR MEDIA. Heineman, R. E. (Battelle-Northwest, Richland, Wash.). Trans. Amer. Nucl. Soc., 8: 532-3(Nov. 1965).

82.

34960 COMPARISON OF MEASURED AND CALCULATED WORTHS OF VARIOUS MATERIALS IN A ZONED FAST ASSEMBLY. Karam, R. A.; Kato, W. Y.; Rusch, G. K. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 9: 232-3(June 1966).

83.

1543 REACTIVITY MEASUREMENTS IN HECTOR. Kinchin, G. H. (Atomic Energy Establishment, Winfrith, Eng.). 3p. (CONF-650602-65). Gmelin, AED-CONF-64-219-134.

From American Nuclear Society, 11th Annual Meeting, Gatlinburg, Tenn.

HECTOR is a graphite-moderated zero-energy reactor with an annular driver region that is maintained at a temperature below 60°C and a central region that can be heated to a temperature of 450°C. Heating and cooling is carried out by circulating carbon dioxide. By changing the fuels loaded in the central region and by altering the temperature, a variety of neutron spectra can be produced. Small samples or single fuel elements can be introduced into the center of the reactor for making reactivity measurements.

84.

16118 (AE-212) CENTRAL REACTIVITY MEASUREMENTS ON ASSEMBLIES 1 AND 3 OF THE FAST REACTOR FR0. Londen, S-O. (Aktiebolaget Atomenergi, Stockholm (Sweden)). Jan. 1966. 47p. Dep. mn.

The reactivity effects of small samples of various materials were measured by the period method at the core center of Assemblies 1 and 3 of the fast zero power FR0. For some materials the reactivity change as a function of sample size was determined experimentally. The core of Assembly 1 consisted only of uranium enriched to 20% whereas the core of Assembly 3 was diluted with 30% graphite. The results were compared with calculated values obtained with a second-order transport-theoretical perturbation model and using differently shielded cross sections depending upon sample size. Qualitative agreement was generally found, although discrepancies still exist. The spectrum perturbation caused by the experimental arrangement was analyzed and found to be rather important. (auth)

These measurements can be made either by an oscillator method, in which the amplitude of reactor power modulation is determined while the sample is regularly moved in and out of the reactor, or by period measurements. (J.F.P.)

85.

34941 ROSSI-ALPHA MEASUREMENTS IN SUBCRITICAL URANIUM-METAL CYLINDERS. Mihalczo, John T. (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 9: 175-6(June 1966).

86.

26434 REACTIVITY EFFECT OF WATER GAP SPACING, AND ^{238}U ON THE SYSTEM ZIRCONIUM-WATER-^{235}U. Rutledge, G. P.; Condit, C. M.; Dobbe, F. A. (Bettis Atomic Power Lab., Pittsburgh). Chem. Eng. Progr., 61: Symp. Ser. No. 60, 54-69(1965).

The reactivity of single homogeneous systems of zirconium-water-U-235 has been previously developed. This information is quantitatively extended by assessing the reactivity effect that results when homogeneous regions are separated by water gaps, that is, heterogeneous systems. Several fuel regions which are stored adjacent to each other under water do not necessarily represent a maximum reactivity. Since these regions are separated by a water gap, which is often necessary in fuel storage, a maximum peak reactivity is reached at a certain spacing between the regions. It is necessary that this maximum value always be determined before the fuel is handled unless it is mechanically impossible to vary this water gap. (auth)

87.

44621 (CONF-660303; pp 773-81) THE INTERPRETATION AND EVALUATION OF INTEGRAL CROSS SECTIONS BY REACTIVITY MEASUREMENTS. Scoville, J. J.; Fast, E. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.).

The importance of reactivity measurement of integral cross sections is discussed. The investigator should either make the measurements in a pure 1/E dependent flux, or be able to correct for any departure from this idealistic condition. Calculational methods are employed to make these, along with other possible corrections for fast fission, (n,2n'), and inelastic scattering reactions. Since many materials do not activate upon neutron absorption, the reactivity technique is important in determining their resonance integrals. Some results determined with the ARMF reactors by static reactivity measurements are included. (auth)

88.

32790 (ANL-6930) PARAMETRIC STUDIES OF THE REACTIVITY COEFFICIENTS FOR LARGE ^{233}U-Th-FUELED FAST REACTORS. Singh, R. S.; Hummel, H. H. (Argonne National Lab., Ill.). Jan. 1966. Contract W-31-109-eng-38. 28p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

The feasibility of using ^{233}U-Th as a fuel in a large fast breeder reactor was studied from the point of view of the Doppler and sodium-void coefficients. The composition and temperature-dependent cross sections for thorium were evaluated by using the latest resonance parameters available. The sodium-void coefficient was studied in detail over a range of compositions and sizes of core for metal and oxide fuels, and the effect of the addition of ^{234}U and fission products on the coefficient is also investigated. The reactivity coefficients due to the Doppler effect are calculated for metal- and oxide-fueled systems of interest. The results indicate that, even with present cross-section uncertainties, the reactivity changes due to both the sodium loss and the Doppler effect are encouraging for the development of fast breeder reactors with very large single cores if ^{233}U and thorium are used as the fuel. (auth)

89.

12358 (LA-3406) SPECTRAL INDICES OF A 4.29 per cent ^{235}U-ENRICHED URANIUM METAL EXPONENTIAL COLUMN. Steinke, R. G. (Los Alamos Scientific Lab., Univ. of California, N. Mex.). Sept. 3, 1965. Contract W-7405-eng-36. 21p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

The spectral indices or detector ratios and buckling values characteristic of the equilibrium neutron energy spectrum in a 4.29% ^{235}U enriched bulk of uranium metal were investigated. The 21-inch-diameter Los Alamos outdoor exponential column was utilized with the column and "Hydro" source reactor elevated some 11 feet above ground level. (auth)

90.

5064 THE REACTIVITY EFFECT OF VOID AND HOLE. Sugimoto, Osamu (Chugoku Electric Power Co., Hiroshima). J. Nucl. Sci. Technol. (Tokyo), 2: 277-84 (Aug. 1965).

The reactivity loss caused by a hole introduced in a reactor is calculated by using the two group perturbation theory based on neutron balance. It is shown that better agreement with experimental results is obtained by the use of a more reasonable probability function than those of Martí and Schneeberger. The reactivity loss by the presence of a void is calculated by the same method, and results are obtained for several cases: When void volume is below

a certain limit, $\Delta k(z)$ is independent of z and proportional to the void volume; in the limit of small void, $\Delta k(z)$ is proportional to the square of flux. From these results the mechanism of reactivity loss is discussed. In order to justify the discussion, a comparison is made of the theoretical results with experiments on JRR-2. (auth)

5. DOPPLER

1956

1.
4920 BNL-1344
Brookhaven National Lab., Upton, N. Y.
TEMPERATURE COEFFICIENTS OF REACTIVITY OF REACTORS. Jack Chernick. Jan. 9, 1953. Decl. Dec. 2, 1955. 22p. Contract AT-30-2-Gen-16. $0.25(OTS).

Experimental and theoretical investigations of the uniform temperature coefficient of reactivity (the reactivity change per degree rise in temperature of all the reactor constituents in some limited and defined temperature range) are described. (L.M.T.)

1957

2.
13563 HW-47834
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

DOPPLER COEFFICIENT OF A DILUTE SYSTEM OF U^{235}. M. V. Davis. Jan. 18, 1957. Decl. Feb. 6, 1957. 3p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80 (mf OTS).

Experiments conducted in the Hanford Test Pile yield the result that cylinders containing U^{235} at densities less than 0.35 gm/cm³ will have a zero Doppler coefficient in the temperature range from 50 to 500°C. (auth)

3.
8758 TID-5183
Los Alamos Scientific Lab., N. Mex.
SEARCH FOR DOPPLER EFFECT IN THE HEATING OF U^{235}. David B. Hall and Jane H. Hall. Dec. 9, 1948. Decl. Feb. 7, 1957. 22p. Contract W-7405-eng-36. $0.35 (OTS).

Several experiments were undertaken in order to detect temperature effects on the Doppler broadening of fission and absorption resonance lines in U^{235}. It is pointed out that rapid temperature increases in a reactor fuel rod could cause a nuclear explosion if an increase in reactivity occurred concurrent with high local heating of the fuel rods. The effect was studied by heating a rod of U^{235} and an Al rod, noting differences in the reactivity of a Pu reactor. In all cases, the effect was not found. (K.S.)

4.
6100
A DIRECT MEASUREMENT OF THE URANIUM METAL TEMPERATURE COEFFICIENT OF REACTIVITY. R. M. Pearce and D. H. Walker (Atomic Energy of Canada Ltd., Chalk River, Ontario). Nuclear Sci. and Eng. 2, 24-32 (1957) Feb.

The U metal temperature coefficient of reactivity has been measured in ZEEP. A U sample was oscillated in the reactor and the resulting modulation of reactor power was measured as a function of the sample temperature. The temperature coefficient of uniformly heated U rods, 3.25 cm. in diameter, immersed in a constant temperature moderator (moderator-to-U volume ratio 22) is deduced from this experiment. Over the range +30°C to +230°C the coefficient is $dk/dT = -(1.25 \pm 0.09) \times 10^{-5}$ per °C. Over the range +10°C to −140°C the coefficient is $dk/dT = -(1.58 \pm 0.18) \times 10^{-5}$ per °C. (auth)

1958

5.
17752 AERE-R/M-168
United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.
A MEASUREMENT OF THE CONTRIBUTION OF THE DOPPLER EFFECT TO THE TEMPERATURE COEFFICIENT OF REACTIVITY IN A FAST REACTOR. A. R. Baker and T. A. J. Jaques. June 1958. 26p.

The Doppler effect in U^{235} and U^{238} in a fast reactor was studied by measuring the changes in reactivity when hot and cold samples were placed alternately at the center of Zeus. The results for U^{235} are consistent with zero Doppler effect. For natural uranium a small negative temperature coefficient was observed, corresponding to increased capture in U^{238} with increasing temperature, given by: $(d\sigma_c/dT) = 0.0006 \pm 0.0002$ mb/°C. The estimated Doppler temperature coefficient for the whole reactor Zeus was zero within experimental error. An upper limit could be placed on the magnitude of the Doppler coefficient (which might be positive or negative) of 10×10^{-7} per °C, which is negligible compared with the magnitude of the total negative temperature coefficient of fast reactors (namely about 150×10^{-7} per °C). (auth)

6.

15072 A/CONF.15/P/1777

Knolls Atomic Power Lab., Schenectady, N. Y. and Argonne National Lab., Lemont, Ill.

MEASUREMENT OF DOPPLER TEMPERATURE COEFFICIENT IN INTERMEDIATE AND FAST ASSEMBLIES. R. T. Frost, W. Y. Kato, and D. K. Butler. 12p. $0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

Measurements made of the Doppler temperature effect for material samples in intermediate and fast spectrum reactors and the experimental technique applied are described. The materials studied in the Be-moderated intermediate reactors were highly enriched U^{235}, U^{238}, and Hf. In the fast reactors, constructed alternatively of highly enriched U and Al or Pu and Al, experiments were performed with samples of enriched and natural U and Pu. Because of the extremely small magnitudes of the Doppler coefficient in the fast spectrum, only upper bounds could be measured for the effect in the latter experiments. (M.H.R.)

7.

15021 A/CONF.15/P/596

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

TEMPERATURE COEFFICIENT MEASUREMENTS OF LIGHT-WATER MODERATED HETEROGENEOUS CRITICAL ASSEMBLIES. S. H. Levine, B. H. Noordhoff, J. J. Taylor, and W. F. Vogelsang. 23p. $0.50(OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958.

A series of critical experiments on heterogeneous, zirconium, light water moderated assemblies in simple geometry is summarized. The assemblies consist of both uniform highly enriched media and seed-blanket slab arrays containing a natural uranium alloy blanket in plate form. A review is given of the typical experimental techniques and experimental data obtained in the process of validating power reactor designs involving these core media. The discussion is divided into two parts; the first covering experimental techniques and the second covering certain of the data obtained. (auth)

8.

2503 HW-51008

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

TEMPERATURE COEFFICIENT OF A GRAPHITE–URANIUM LATTICE. R. C. Lloyd and C. R. Richey. June 18, 1957. Decl. Nov. 7, 1957. 29p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

Material buckling measurements were made in a graphite-U exponential pile at temperatures ranging from 24 to 460°C. The large (101-inch cube) exponential pile was loaded with 1.36-inch diameter fuel elements of natural U. The temperature coefficient, over this range is $(1/B_m^2)(\Delta B_m^2/\Delta T) = -9.62 \times 10^{-4}$/°C and $(1/k_\infty)(\Delta k_\infty/\Delta T) = -4.58 \times 10^{-5}$/°C. Lattice parameters were calculated and the values of $\Delta\eta/\eta$ were determined which permit a fit to the experimental results. (auth)

1959

9.

445 KAPL-M-WS-3

Knolls Atomic Power Lab., Schenectady, N. Y.

ON THE TEMPERATURE DEPENDENT PROPERTIES OF AN 8 1/4 INCH SLAB REACTOR. J. A. Bistline, R. G. Luce, and W. Skolnik. Sept. 3, 1958. 30p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

Experiments to determine the critical load of an $8\frac{1}{4} \times 30 \times 32$ inch reactor as a function of moderator temperature are described. The reactor contained water, highly enriched uranium, and Zircaloy. The metal-water ratio was 1.53 to 1 and a temperature range of 79.5 to 513°F was covered in five steps. (W.D.M.)

1960

10.

23681 APDA-139

Atomic Power Development Associates, Inc., Detroit.

THE DOPPLER EFFECT IN FAST NEUTRON REACTORS (thesis). Richard Benjamin Nicholson. June 1960. 145p. OTS.

Submitted to Cornell Univ.

A new analysis of the Doppler effect in fast neutron reactors is made. The effect is evaluated for U^{235} and U^{238} at several neutron energies between 1 and 200 kev. A new method is presented for U^{238} at neutron energies below 30 kev where the previous method is shown to be invalid. The new method is based on the assumption that the individual resonances are isolated and do not overlap. The importance of inelastic scattering is evaluated and the effects of other neglects previously made without complete justification are studied in some detail. The newer data on low energy resonance parameters and high energy cross sections are examined to determine the best choice for the statistical distributions of neutron and fission widths and the distribution of resonance spacing. A crude numerical calculation for the Doppler temperature coefficient of reactivity in the Fermi Fast Breeder Reactor gives about -2×10^{-6}/C at 550°C. The results indicate that the Doppler effect from U^{235} can almost always be neglected as small but in some of the larger reactors such as a large fast oxide breeder U^{238} may contribute a negative Doppler effect of the order -10^{-5} which would be of considerable importance, especially with respect to reactor safety. (auth)

1961

11.

21742 DOPPLER RESONANCE INTEGRAL COEFFICIENT OF UO_2 AND ThO_2. R. L. Crowther (General Electric Co., Pleasanton, Calif.). Trans. Am. Nuclear Soc., 4: No. 1, 18-19(June 1961).

12.

21741 THE DOPPLER COEFFICIENT OF REACTIVITY IN A FAST U-233–Th-232 REACTOR. F. R. Nakache and M. H. Kalos (Nuclear Development Corp. of America, White Plains, N. Y.). Trans. Am. Nuclear Soc., 4: No. 1, 17-18(June 1961).

13.

16659 THE DOPPLER EFFECT IN THERMAL REACTORS. R. M. Pearce (Atomic Energy of Canada Ltd., Chalk River, Ont.). J. Nuclear Energy, Pt. A Reactor Sci., 13: 150-75(Jan. 1961). (In English)

Experimental and theoretical work on the Doppler effect in thermal reactors is reviewed for U metal, UO_2, Th metal, and ThO_2. The experimental values of α, the fractional increase in resonance capture per °C, have a spread many times the quoted errors. The use of different slowing-down spectra contributes to the discrepancies. For U metal, approximate corrections are made to obtain the coefficient α_0 appropriate to a 1/E spectrum. The spread in the corrected values α_0 is smaller than that for α, but remains unsatisfactory. Other experimental difficulties arise in reactivity normalizations, in obtaining the statistical weight of samples, and from spurious temperature effects. Theory and experiment agree on an increase of α_0 with increasing surface-to-mass ratio and that this is caused by an increase in the contribution of lower-energy resonances to the Doppler effect. It is also in agreement with the theoretical interpretation of the radial dependence of the Doppler effect in a lump. However in the region of practical interest where the surface-to-mass ratio is small, α_0 is almost constant. Experimental evidence on the temperature behavior of α_0 is unsatisfactory but indicates that α_0 decreases with increasing temperature. Theory predicts that α_0 will vary approximately as T^{-1} where T is the Kelvin temperature. In the case of non-uniform temperature distribution in a fuel element, both experimental and theoretical effort is needed. (auth)

1962

14.

18608 (AERE-M-911) CALCULATION OF THE TEMPERATURE COEFFICIENT OF REACTIVITY OF A GRAPHITE-MODERATED REACTOR. G. Brown, R. Richmond, and R. H. W. Stace (United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England). Mar. 1962. 6p.

Declassified version of NPCC/RPWP/P 66; AERE/R/M-132.

The temperature coefficients of reactivity of the BEPO, Windscale, and Calder reactors are calculated. The results are compared with experimental values. (auth)

15.

31326 (IDO-16772) TEMPERATURE COEFFICIENTS OF THE REACTIVITY MEASUREMENT FACILITY. E. Fast (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). July 27, 1962. Contract AT(10-1)-205. 15p.

The temperature coefficient of the Reactivity Measurement Facility was found to be 49 ± 1 μk/°C (1 μk = 10^{-6} $\Delta k/k$) in the range 15.4 to 17.8°C. The change in the net reactivity of a standard sample was −0.48 ± 0.02, −0.66 ± 0.03, and −0.78 ± 0.02 μk/°C in three measuring positions. These low values generally make temperature corrections insignificant. The above results are compared with previous determined values. This information developed in the RMF should be generally applicable to flux-trap-type reactors such as the Advanced Reactivity Measurement Facility (ARMF) and ARMF-II, now under construction. RMF was dismantled in April 1962. (auth)

16.

26562 THE DOPPLER EFFECT OF PLUTONIUM-239 IN FAST REACTORS. E. A. Fossoul (Belgonucleaire, Brussels). p.139-51 of "Physics of Fast and Intermediate Reactors. Vol. III." Vienna, International Atomic Energy Agency, 1962. (In French)

The Doppler effect of Pu^{239} in a fast spectrum was studied. Following Goertzel, Lane and Bethe, and the formulation proposed by Nicholson, the Doppler effect of Pu^{239} at 1 to 200 kev was evaluated. At these energies, the total macroscopic cross-section $\mu_t(E)$ scarcely deviates from its mean value $(\mu_t(E))$, and the contribution to the Doppler effect by the group of neutrons centered on energy E can be evaluated by Nicholson's method "A." The function $Q_E^{(s)}$ is calculated on the basis of findings regarding the resonances of Pu^{239}. (auth)

17.

1020 (IDO-16679) DOPPLER BROADENING OF THE LOW ENERGY TOTAL NEUTRON CROSS SECTION OF Pu-241. O. D. Simpson and N. H. Marshall (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Aug. 4, 1961. Contract AT(10-1)-205. 96p.

The low energy total neutron cross section of Pu^{241} was measured from 0.02 ev to 2 kev with the use of the Materials Testing Reactor (MTR) fast chopper. Curves and tables of the total cross section as a function of energy are given. Pronounced asymmetries were observed in several of the resonances, indicating the existence of interference in fission. The data were analyzed below 12 ev with the use of the Reich-Moore multilevel formula, under the assumption that the observed resonance asymmetries are due to interference in a small number of fission channels. A study was

conducted to determine the extent of the Doppler distortion of the resonances below 11 ev, at temperatures ranging from 0 to 2000 degrees centigrade. An estimate of the neutron strength function $\langle \Gamma_n^0/D \rangle$ obtained from the average cross section in the kev energy region is $(0.85 \pm 0.15) \times 10^{-4}$. (auth)

18.

27734 (HW-73116(p.71-81)) THE FUEL TEMPERATURE COEFFICIENT OF K_∞ FOR PLUTONIUM–ALUMINUM FUEL. R. I. Smith (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

The change in the reactivity of the Physical Constants Testing Reactor (PCTR) was measured as a section of Pu-Al fuel rods in the center test cell of the core was heated from room temperature to more than 400°C. Measurements were made with 1.88 wt % low exposure Pu-Al, with 2.07 wt % high exposure Pu-Al, and with dummy Al rods in the test cell. The signs and the magnitudes of the fuel temperature coefficients of k_∞ were obtained from the data. The effects of heating the fuel on the neutron multiplication of the lattice cells fueled with Pu-Al were found to be small in magnitude and negative in sign. The magnitude of the negative effect was seen to be a function of the concentration of Pu^{240} in the fuel. The higher Pu^{240} density material yielded a larger negative effect. The results indicate that for Pu-Al fuels of low Pu density, i.e. ~2 wt % Pu in Al, the major contributor to the fuel temperature coefficient is the Doppler effect in Pu^{240}. (auth)

1963

19.

15067 (GA-3422) MEASURED DOPPLER COEFFICIENT OF THORIUM DISPERSED IN GRAPHITE. J. R. Brown and J. B. Sampson (General Atomic Div. General Dynamics Corp., San Diego, Calif.). Feb. 6, 1963. Contract AT(04-3)-314. 71p.

The Doppler coefficient in thorium dispersed in graphite was measured in a reactor spectrum developed in the HTGR critical assembly, using a reactivity technique that compares the reactivity of a single cold element and a single hot element heated to 700°K. The considerable correction due to the small perturbation in neutron temperature resulting from the heating has been experimentally determined by auxiliary measurements. An activation technique has been used to check the results. In this activation technique, the thermal-base effect has also been experimentally subtracted by measuring the 1/v component of the captures at each temperature in vanadium. Measurements of the Doppler coefficient obtained by these two different techniques agree to within 10%. This agreement, demonstrated by applying both techniques for the first time in the same laboratory, shows that the previously reported spread of results and apparent disagreement between results of these two techniques are avoidable. The average Doppler coefficient for the temperature interval from 300 to 700°K, for a carbon-to-thorium ratio of about 50 in the thorium-bearing region, was measured to be 3.6×10^{-6}/°C to within about 10% uncertainty. The measured results are in satisfactory agreement with calculated values. (auth)

20.

3767 STUDIES OF THE DOPPLER EFFECT IN LARGE CERAMIC-FUELED FAST REACTORS. H. H. Hummel (Argonne National Lab., Ill.) and M. G. Bhide. Trans. Am. Nucl. Soc., 5: 366-7(Nov. 1962).

21.

3767 STUDIES OF THE DOPPLER EFFECT IN LARGE CERAMIC-FUELED FAST REACTORS. H. H. Hummel (Argonne National Lab., Ill.) and M. G. Bhide. Trans. Am. Nucl. Soc., 5: 366-7(Nov. 1962).

22.

4340 (TID-17271) ISOTHERMAL TEMPERATURE EFFECTS IN THE ADVANCED TEST REACTOR CRITICAL EXPERIMENTS. A. L. MacKinney and J. W. Poston (Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va.). Oct. 1962. For Ebasco Services, Inc., New York. Contract AT(10-1)-1075. 16p. (ATRCE-102)

Isothermal temperature effects were measured in an ATR Critical Experiment core containing two strips of borated mylar film on each plate, no neck draw rods, and control cylinders at an initial position of approximately 100 degrees. Two sets of measurements were made because the first set of data indicated uncertainties in both temperature and reactivity measurements. Both measurements indicated slightly positive temperature coefficients initially. The first measurement indicated a change of sign at higher temperatures, while the second indicated a slightly less positive effect at higher temperatures. An additional measurement of isothermal temperature effects was made in an ATRCE Core with no borated plastic in the fuel region. Core reactivity was held by two fully inserted safety rods in Lobe Positions 1 and 9, and an additional measurement was made with a third safety rod inserted in Lobe 5. A reactivity loss of 31.1 cents was obtained by increasing core temperature from 21.7 to 39.7 C with two safety rods inserted; a reactivity loss of approximately 67 cents was obtained by increasing core temperature to the same degree with the three safety rods fully inserted. (auth)

23.

17599 (TID-18254) SPATIALLY DEPENDENT TEMPERATURE COEFFICIENTS IN THE ATRCE CORE. A. L. MacKinney and J. W. Poston (Babcock and Wilcox Co. Critical Experiment Lab., Lynchburg, Va.). Mar. 1963. For Ebasco Services, Inc., New York. Contract AT(10-1)-1075. 41p. (ATRCE-109)

Temperature effects on reactivity caused by heating various regions of the Advanced Test Reactor Critical Experiment Core were measured by pumping heated water separately through isolated flux trap and fuel regions. The data necessary to plot reactivity as a function of tempera-

ture were obtained, and isothermal temperature coefficients were measured to correct for bulk temperature rise. As expected, the results indicated a negative temperature coefficient, −2.5¢/C, in the fuel region and a positive temperature coefficient, estimated to be +1.3¢/C, in all nine flux traps. These measurements are appropriate for a core containing six neck shim rods per neck and having the control cylinders unbalanced to approximately simulate the reference power split. (auth)

24.

9855 (TID-13139) REACTOR TEMPERATURE. C. B. Mills (Los Alamos Scientific Lab., N. Mex.). [1961]. 17p. (LADC-4938)

Beryllium oxide and graphite critical experiments operated in the temperature range 90° to 1200°F were analyzed with a simple gas model neutron scattering matrix. A comparison of room temperature and high temperature experiments showed no systematic trends clearly ascribable to the method of calculation. It was concluded that in this temperature range the gas model is sufficiently accurate for the analysis of this reactor type for the epithermal to intermediate flux spectra found here. (auth)

25.

36660 (NAA-SR-Memo-8690) THE DOPPLER EFFECT OF U-238, Th AND Au. J. M. Otter (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.). July 2, 1963. Contract [AT(11-1)-Gen-8]. 30p.

Theoretical values of resonance integrals and Doppler coefficients of U-238, Th, and Au were found using recently improved, yet efficient, calculational procedures. A range of effective scattering cross section per atom from infinity to very small values, and a wide temperature range were investigated. The mathematical representations and sources of physical parameters are reviewed. The Doppler coefficient was found to exhibit a maximum at an intermediate value of self-shielding, which is determined primarily by the properties of the lowest energy resonances. Two parameter fits were made to both resonance integrals and Doppler effect coefficients for practical rod sizes. Comparison with recent experiments showed significant lack of correlation with theoretical calculations for the thorium resonance integral and the U-238 Doppler coefficient. (auth)

1964

26.

44930 (ANL-6936(p.17-38)) LIQUID-METAL-COOLED REACTORS. (Argonne National Lab., Ill.).

Doppler measurements in ZPR-III are reported for $^{239}Pu-^{238}U$ metallic fuel mockups and urania–plutonia test samples. Axial flux transverses were made by the foil-activation technique in a 600-liter core containing U, Na, and graphite with a length-to-diameter ratio of about 1 to 1. Spatially dependent worth studies were made on Rh reflectors in ZPR-IX, along with studies of B worth enhancement by H and measurements of central worth coefficients of Au, Zr, and ZrH_4. Development work on Ti–V cladding materials is described, and preliminary corrosion tests of V-base binary alloys are reported. Developments in fast-reactor fuel processing are also reported. EBR-II measurements made during the period included heat balance and transfer functions in approach-to-power experiments and power and flow coefficients. Analyses of the Ar cover gas system in EBR-II revealed that about 2200 PPM He and 20 PPM H_2 were present. Operation of the EBR-II power plant is summarized. Development and operation of the fuel-cycle facility and FARET are reported. (J.R.D.)

27.

16891 (GA-4881) INTEGRAL NEUTRON THERMALIZATION. Quarterly Progress Report for the Period Ending December 31, 1963. J. R. Beyster, G. D. Joanou, J. Kirkbride, J. U. Koppel, J. M. Neill, J. A. Young, and J. C. Young (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Jan. 15, 1964. Contract AT(04-3)-167. 38p.

Data were presented on spectral and reactivity measurements in an assembly of UF_4 and paraffin, the sensitivity of BF_3 detector banks, fast neutron penetration in water, the scattering kernel for polyethylene, lattice vibrational spectrum of magnesium, and the specific heat of magnesium. (C.E.S.)

28.

14709 DOPPLER COEFFICIENT MEASUREMENTS FOR U^{238} IN FAST-REACTOR SPECTRA. G. J. Fisher, H. H. Hummel, J. R. Folkrod, and D. A. Meneley (Argonne National Lab., Ill.). Nucl. Sci. Eng., 18: 290-2(Feb. 1964).

Experimental results of measurements of the Doppler temperature coefficient of reactivity of U^{238} for fast-spectrum assemblies on the ZPR III are reported. (R.E.U.)

29.

18866 (ANL-6792(p.885-95)) EXPERIMENTAL RESULTS FOR U^{238} DOPPLER MEASUREMENTS IN FAST REACTOR SPECTRA. G. J. Fischer, H. H. Hummel, J. R. Folkrod, and D. A. Meneley (Argonne National Lab., Ill.).

Doppler reactivity measurements were made for uranium-238 heating in two-zoned core loadings of the ZPR-III fast critical assembly. The central zone of the first loading, in which the measurements were made, had the composition of a 5000-liter uranium monocarbide fast power breeder. The second central-zone loading differed from the first by replacement of 40% of the sodium with graphite. This change gave an increased Doppler reactivity effect and a second point for theoretical comparison. The design of the Doppler element and of the core loading, both of which were planned to minimize non-Doppler reactivity effects, is described. Supplementary

experiments which show that non-Doppler reactivity effects must represent at most a small part of the observed reactivity change are discussed. An experiment which tested the importance of resonance depression of the neutron flux incident upon the element, due to adjacent ^{238}U in the central-zone loading, is described and results are given. Calculational comparisons with these experimental results using recently developed ANL cross sections are reported. They show close agreement with the first experiment and poorer agreement with the softer spectrum case. (auth)

30.

42937 (TID-7050(p.1218-45)) HIGHLY ENRICHED CLEAN CRITICAL EXPERIMENTS AT ELEVATED TEMPERATURES. L. O. Herwig (Westinghouse Electric Corp. Bettis Atomic Power Lab., Pittsburgh); and W. Skolnik (Knolls Atomic Power Lab., Schenectady, N. Y.).

Experiment and theory are compared for some high-temperature clean cores; reactivity and power distributions were studied. Experimental results from both the Pressurized Test Reactor and High Temperature Test Facility are discussed. (R.E.U.)

31.

28712 DOPPLER EFFECT IN URANIUM-238 AT HIGH S/M VALUES. R. Sher (Stanford Univ., Calif.) and G. R. Pflasterer. Trans. Am. Nucl. Soc., 7: 27(June 1964).

32.

2964 FAST-SPECTRUM DOPPLER MEASUREMENTS. T. H. Springer and S. G. Carpenter (Atomics International, Canoga Park, Calif.). Trans. Am. Nucl. Soc., 6: 242(Nov. 1963).

33.

44984 FAST-SPECTRUM DOPPLER MEASUREMENTS. PART II. A MEASUREMENT OF A NEGATIVE DOPPLER EFFECT IN U^{235} AND NEW MEASUREMENTS IN Th^{232} AND U^{238}. T. H. Springer, S. G. Carpenter, and R. J. Tuttle (Atomics International, Canoga Park, Calif.). Nucl. Sci. Eng., 20: 272-80(Nov. 1964).

A negative Doppler effect was observed in measurements on a metallic uranium sample (enriched to 93% in ^{235}U) placed in a typical fast-reactor spectrum which has a median fission energy of 195 kev. The plausibility of the negative sign is supported on theoretical grounds, although with the use of standard analytical techniques and the limited number of resonance parameters at present available, it can not be calculated for this spectrum. The value for $|1/\rho|(d\rho/dt)$ was found to be -5.65×10^{-6}/°C. Measurements were made on ^{238}U and the data on ^{232}Th, previously published, were extended to 930°C. The agreement between analytical and experimental values for the latter materials is good. (auth)

34.

18867 (ANL-6792(p.897-912)) FAST SPECTRUM DOPPLER MEASUREMENTS OF Th^{232}, U^{238}, AND U^{235} METALS. T. H. Springer and S. G. Carpenter (Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif.).

Doppler effect measurements for ^{232}Th, ^{235}U, and ^{238}U performed in the AETR fast spectrum are presented and compared with theoretical results. (C.E.S.)

1965

35.

45642 (DP-999, pp 65-71) TEMPERATURE COEFFICIENTS OF REACTIVITY IN THE SAVANNAH RIVER HIGH FLUX CHARGE. Bailey, C. E.; Baumann, N. P.; Jewell, C. E. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Lab.). (CONF-650602-4).

From American Nuclear Society 11th Annual Meeting, Gatlinburg, Tenn.

Calculations of the temperature coefficients of reactivity in the high-flux charge are compared with measurements of the coefficients in both a subcritical facility and a zero-power critical facility and with measurements in the reactor. (D.C.W.)

36.

29334 A WIDE-RANGE STUDY OF THE EFFECTIVE RESONANCE INTEGRAL AND DOPPLER EFFECT IN Au-197. L. S. Beller and H. Farrar, IV (Atomics International, Canoga Park, Calif.). Trans. Am. Nucl. Soc., 8: 285-6(May 1965).

37.

1834 (AEEW-R-318) BUCKLING AND REACTION RATE EXPERIMENTS IN PLUTONIUM/URANIUM METAL FUELLED, GRAPHITE MODERATED LATTICES AT TEMPERATURES UP TO 400°C. PART I. EXPERIMENTAL TECHNIQUES AND RESULTS. D. H. Carter, R. Hobday, B. J. Puckett, W. G. Clarke, C. Hunt, C. R. Symons, M. Gibson, J. Marshall, and T. Wass (United Kingdom Atomic Energy Authority. Reactor Group. Atomic Energy Establishment, Winfrith, Dorset, England). July 1964. 138p. Dep.; $4.20(BIS), 21s.(HMSO).

The report presents experimental measurements of bucklings, flux fine structure, and fission rate distributions in graphite-moderated lattices fueled with plutonium/uranium metal at temperatures up to 400°C in the subcritical assemblies SCORPIO I and SCORPIO II. The experimental techniques employed are described in some detail. The accuracy of the experimental measurements appears to be adequate for testing methods of calculation being developed for the calculation of reactivity and temperature coefficient of reactivity for power reactors containing plutonium and uranium. (auth)

38.

3544 DOPPLER-EFFECT MEASUREMENTS IN PLUTONIUM-FUELED FAST ASSEMBLIES. G. J. Fischer, D. A. Mencley, and E. F. Groh (Argonne National Lab., Ill.). Trans. Am. Nucl. Soc., 7: 252(Nov. 1964).

39.

1864 (ANL-6838) DOPPLER EFFECT MEASUREMENTS, SODIUM VOID COEFFICIENTS, AND CRITICAL STUDIES OF A 5,000-LITER FAST POWER BREEDER REACTOR BY THE TWO-ZONE METHOD. ZPR-III Assembly 43. J. M. Gasidlo, J. K. Long, and W. P. Keeney (Argonne National Lab., Idaho Falls, Idaho). May 1964. Contract W-31-109-eng-38. 40p. Dep.; $2.00(cy), 1(mn) OTS.

Assembly 43 is the second two-zone core built in the ZPR-III facility. The central zone and the low-density axial blanket were designed to simulate a large power-breeder reactor. Criticality was achieved by an annular driver, while an interposed buffer modified the spectrum of the neutrons diffusing from the driver to the central zone. Experiments included sodium void coefficients for central substitutions and for full-length axial measurements. Other reactivity measurements included central reactivity coefficients and core-length column substitutions in the central zone and the annular driver. Spectral indices were measured, and reaction-rate traverses and activation experiments were performed. The reactivity change due to the Doppler effect caused by heating a large, depleted, uranium oxide sample was also determined in the experimental program. The experimental results indicate that the measurements can be used for checking calculations of dilute fast cores, as well as evaluating the two-zone experimental technique. (auth)

40.

29574 FAST-REACTOR MELTDOWN ACCIDENTS WITH DOPPLER EFFECTS. R. A. Meyer, B. Wolfe, and N. Friedman (General Electric Co., San Jose, Calif.). Trans. Am. Nucl. Soc., 8: 310-11(May 1965).

41.

29199 FAST-SPECTRUM DOPPLER MEASUREMENTS. T. H. Springer, S. G. Carpenter, L. A. Mountford, and R. J. Tuttle (Atomics International, Canoga Park, Calif.). Trans. Am. Nucl. Soc., 8: 242(May 1965).

42.

17243 THE DOPPLER COEFFICIENT. L. W. Nordheim (General Atomic, Div. of General Dynamics, San Diego, Calif.). p.205-43 of "The Technology of Nuclear Reactor Safety. Vol. 1." M.I.T. Press, Cambridge, 1964.

Neutron resonance absorption in thermal and fast reactors is surveyed, with emphasis on the Doppler effect. 77 references. (T.F.H.)

43.

3480 PHYSICS. Power Reactor Technol., 7: 329-34(1964).

The efficacy of the Doppler effect as a shutdown agency in power excursions is discussed, and recent work on Doppler evaluations are reviewed and some thermal calculations given. The programming of control rod withdrawal is also reviewed. Nuclear data for Zr, the age of fission neutrons to 1.46 ev in water, and neutron thermalization are discussed briefly. (D.L.C.)

44.

45532 (ORNL-TM-1219) A SURVEY OF DOPPLER COEFFICIENTS FOR UO_2 FUEL. Preskitt, C. A. (Oak Ridge National Lab., Tenn.). Apr. 27, 1965. Contract W-7405-eng-26. 10p. Dep. mn; CFSTI $1.00 cy, $0.50 mn.

Four experimental and two theoretical studies of the Doppler coefficient of UO_2 fuel were compared. The results appear to divide into two distinct groups with about the same dependence on surface-to-mass ratio, but separated by a difference of ~15%. (auth)

45.

29332 MEASUREMENT OF THE DOPPLER COEFFICIENT OF THORIUM OXIDE RODS TO 1550°C. F. C. Schoenig, K. S. Quisenberry, D. P. Stricos, and H. Bernatowicz (Knolls Atomic Power Lab., Schenectady, N. Y.). Trans. Am. Nucl. Soc., 8: 284(May 1965).

46.

3388 DOPPLER EFFECT IN URANIUM-238 CAPTURE AND URANIUM-235 FISSION IN A FAST NEUTRON SPECTRUM. R. Sher (Stanford Univ., Calif.) and G. R. Pflasterer, Jr. Trans. Am. Nucl. Soc., 7: 272-3(Nov. 1964).

47.

3569 MEASUREMENT OF $k_\infty(T)$ FOR A URANIUM FUEL ELEMENT. R. I. Smith (General Electric Co., Richland, Washington). Trans. Am. Nucl. Soc., 7: 297-8 (Nov. 1964).

48.

17229 THE VARIATION OF k_∞ WITH FUEL TEMPERATURE FOR A URANIUM FUEL ELEMENT. R. I. Smith (General Electric Co., Richland, Wash.). Nucl. Sci. Eng., 21: 481-9(Apr. 1965).

The change in k_∞ of a heterogeneous lattice caused by a uniform change in the temperature of the fuel was measured, using the Physical Constants Testing Reactor. The test lattice was moderated with graphite and fueled with concentric-tube elements of slightly enriched uranium metal. The temperature of the fuel was varied from 297 to 1241°K. The change in k_∞ with high temperature was nonlinear and could be represented by the relation. $\Delta k_\infty = C[\alpha(T^4 - T_0^4) + \beta U(T_{\alpha\beta}) + \gamma U(T_{\beta\gamma})]$, where T is in degrees Kelvin. The experimentally measured values of the constants were $\alpha = (-0.308 \pm 0.004)$, $\beta = (-0.120 \pm 0.004)$, $\gamma = (-0.085 \pm 0.004)$. The unit functions, U, represent the changes in k_∞ caused by the isothermal volume expansion of the fuel element when the uranium metal undergoes transformations in its crystal structure from alpha to beta and from beta to gamma phases. The term C is a normalization factor related to the lattice under study. The reactivity techniques employed here are shown to be four times more sensitive than activation methods for determining the functional relationship between the effective resonance integral of a fuel element and the temperature of the element. The constant, α, was experimentally separated into two components: $\alpha_v = (-0.240 \pm 0.04)$, which is associated with the average interior temperature of the fuel, and $\alpha_s = (-0.068 \pm 0.04)$, which is associated with the temperature of the surface of the fuel. This separation allows treatment of nonuniform temperature distribution in the fuel. (auth)

49.

3782 MEASUREMENTS OF THE ISOTHERMAL TEMPERATURE COEFFICIENT OF REACTIVITY FOR THE LIGHT-WATER-MODERATED, 2.1%-ENRICHED UO_2 LATTICES OF THE CORNELL ZERO POWER REACTOR. W. P. Wynn, Jr., David D. Clark, and David J. Osias (Cornell Univ., Ithaca, N. Y.). Trans. Am. Nucl. Soc., 7: 213-14(Nov. 1964).

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50.

5072 DOPPLER COEFFICIENT MEASUREMENTS FOR ^{235}U IN FAST-REACTOR SPECTRA. Amundson, P. I.; Gasidlo, J. M. (Argonne National Lab., Idaho Falls, Idaho). Nucl. Sci. Eng., 23: 392-3(Dec. 1965).

The Doppler coefficient of ^{235}U was measured in the ZPR II Assembly 45A, a zoned reactor containing a central region whose composition represents a large sodium-cooled fast power-breeder reactor having ^{239}Pu-^{238}U monocarbide fuel. The reactor regional parameters are tabulated. The reactivity coefficient was found to be approximately $+3 \times 10^{-6}$/°C. (L.B.S.)

51.

20310 (AEEW-R-465) DOPPLER COEFFICIENT MEASUREMENTS IN ZEBRA CORE 5. Baker, A. R.; Wheeler, R. C. (Atomic Energy Establishment, Winfrith (England)). Nov. 1965. 30p. Dep. BIS $0.90. HMSO 4s. 6d.

Measurements using a central hot loop in Zebra Core 5 are described. Results are given for the Doppler coefficients found in a number of assemblies with PuO_2 and 16% PuO_2/84% depleted UO_2 pins, loaded with different combinations of steel, sodium or void pins. The mixed oxide results are in general about 20% more negative than was calculated using the FD2 data set, but agreement is good if the plutonium contributions in the calculations are omitted. The small positive Doppler coefficient calculated for ^{239}Pu was not observed, and two measurements indicated instead a small negative effect. The Doppler effect in the mixed oxide systems was found to vary approximately as 1/T. The results from the empty loop and non-fissile assemblies indicate either a small negative Doppler effect in steel or alternatively the presence of an unexplained expansion effect. (auth)

52.

38429 (ANL-7120, pp 614-15) RECENT RESULTS OF DOPPLER MEASUREMENTS IN FAST-NEUTRON SPECTRA. Carpenter, S. G.; Mountford, L. A.; Springer, T. H.; Tuttle, R. J. (Atomics International, Canoga Park, Calif.).

Doppler measurements in fast-neutron spectra were made using small samples in a multiregion reactor. The variation of reactivity of ^{235}U metal due to temperature and expansion and the derived Doppler reactivity for different calculated median fission energies are shown. It is concluded that the Doppler effect in fast cores is positive for ^{235}U, though it does go negative in the low-kev region. The change in reactivity of ^{238}U metal with temperature is shown. (A.G.W.)

53.

20197 (AEEW-R-438) BUCKLING AND REACTION RATE MEASUREMENTS IN GRAPHITE MODERATED LATTICES FUELLED WITH PLUTONIUM-URANIUM OXIDE CLUSTERS AT TEMPERATURES UP TO 400 DEG C. Carter, D. H.; Gibson, M.; King, D. C.; Marshall, J.; Puckett, B. J.; Richards, A. E.; Wass, T.; Wilson, D. J. (Atomic Energy Establishment, Winfrith (England)). July 1965. 147p. Dep. BIS $4.00. HMSO 20s.0d.

A series of experiments carried out in SCORPIO I and II on sub-critical graphite moderated lattices fueled with 21-rod clusters of PuO_2/UO_2 fuel are described. Three fuel batches with nominal plutonium: uranium ratios of 0.25%, 0.8% and 1.2% were investigated at temperatures between 20°C and 400°C. Because of the limited amounts of the three fuels, exponential measurements were made in 2-zone stacks, the outer regions of which were loaded with suitably matched "reference fuel." Fine structure distributions in the lattice cell were obtained with manganese and indium foils. $^{239}Pu/^{235}U$ fission ratios were determined both by fission chambers and by fission-product counting techniques. (auth)

54.

10332 (EURFNR-12F) INVESTIGATION OF THE REACTIVITY COEFFICIENT CONNECTED TO THE SODIUM TEMPERATURE IN BIG FAST REACTORS. Chaumont, Jean-Marie; Martin, Jacques (Commissariat a l'Energie Atomique, Cadarache (France). Centre d'Etudes Nucleaires). Translation of report CEA-R-2555. June 1964. 34p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

Work Performed under United States–Euratom Fast Reactor Exchange Program.

The reactivity of large, sodium-cooled, fast reactors is investigated. Separate discussions are presented on the components of the reactivity coefficient connected to the sodium temperature, influence of fuel percentage in the reactor, influence of reactor geometry, influence of sodium cross sections, improvement of the coolant temperature coefficient, and a comparison between experimental and theoretical values of the reactivity coefficient of sodium is presented. (M.O.W.)

55.

44582 (NAA-SR-11963) EFFECTIVE GROUP ABSORPTION CROSS-SECTIONS AND RESONANCE OVERLAP. Fillmore, F. L. (Atomics International, Canoga Park, Calif.). July 15, 1966. Contract AT(11-1)-Gen-8. 46p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

The effect of resonance overlap on multigroup resonance cross sections and the Doppler coefficient is examined. Consistent definitions are given for the group flux and effective cross section using both the TRUE FLUX and the 1/E FLUX methods. Approximate expressions are developed for the effective resonance absorption cross section in a homogeneous mixture. The validity of the approximations is discussed for typical fast reactor cores, and an example is presented of a correction for self-overlap. The calculation of reactivity coefficients by means of first-order perturbation theory is interpreted, and it is shown that the same result is obtained with either the TRUE FLUX or the 1/E FLUX method of defining the effective cross sections; however, the TRUE FLUX method is the one recommended. Approximate expressions for the change in the effective absorption cross section due to Doppler broadening of resonances are developed, and con-

sideration is given to the effect of resonance overlap on this change in the effective cross section. It is concluded that the error in the Doppler coefficient for typical fast reactor cores due to neglecting resonance overlap should usually not exceed a few percent provided that the TRUE FLUX definitions are used for the effective cross sections. It is recognized, however, that situations may exist where the neglect of resonance overlap is not justified. (auth)

56.

38427 (ANL-7120, pp 603-9) MEASUREMENT AND ANALYSIS OF DOPPLER EFFECT IN PLUTONIUM-FUELED FAST REACTOR ASSEMBLIES. Fischer, G. J.; Meneley, D. A.; Hwang, R. N.; Groh, E. F.; Till, C. E. (Argonne National Lab., Ill.).

Measurements made with the ZPR-3 fast critical assembly are described. Zone-loading procedures were used to produce a mockup of a large fast power breeder using ^{239}Pu, ^{238}U monocarbide fuel, Na coolant, and stainless steel clad and structure. In another assembly 40% of the Na cans were replaced by graphite. The measurements indicated that the positive Doppler reactivity contribution of ^{239}Pu in the fuel and in the neutron-energy spectrum of a large power breeder reactor is small in magnitude compared to the strong negative Doppler effect of ^{238}U. Experimental results are compared with theoretical calculations; experimental and theoretical uncertainties are discussed. It is concluded that much more work must be done with the fissile isotopes in varying fast reactor spectra. (A.G.W.)

57.

2958 DOPPLER COEFFICIENTS OF FERTILE AND REFRACTORY MATERIALS. Mountford, L. A.; Carpenter, S. G.; Springer, T. H.; Tuttle, R. J. (Atomics International, Canoga Park, Calif.). Trans. Amer. Nucl. Soc., 8: 454(Nov. 1965).

58.

2976 MEASUREMENT OF THE EFFECTIVE RESONANCE INTEGRAL AND DOPPLER COEFFICIENT OF THORIUM-OXIDE RODS. Palowitch, B. L.; Hardy, J. Jr. (Bettis Atomic Power Lab., Pittsburgh). Trans. Amer. Nucl. Soc., 8: 472-3(Nov. 1965).

59.

44627 (CONF-660303, pp 933-8) MEASUREMENTS AND CALCULATIONS OF THE DOPPLER EFFECT ON THE REACTIONS $^{238}U(n,\gamma)$, $^{235}U(n,f)$ AND $^{239}Pu(n,f)$ WITH NEUTRONS IN THE ENERGY RANGE 0-25 kev. Perkin, J. L.; Fieldhouse, P.; Brickstock, A.; Davies, A. R. (Atomic Weapons Research Establishment, Aldermaston (England)).

Spherical samples, 2 cm in diameter, of ^{238}U, ^{235}U, and ^{239}Pu were irradiated in turn at various temperatures ranging from 170 to 770°K in a central cavity of a spherically symmetric Sb-Be photo neutron source. The $^{238}U(n,\gamma)$ reaction rate was measured by counting the ^{239}U activity produced, and the (n,f) reactions were monitored by counting the fission neutrons emitted. The results obtained for the dependence of reaction rate on temperature were compared with those from a computer calculation based on a program developed by Brissendon and Durston. (ANL-7050, p.51 (1965)). (auth)

60.

45088 DOPPLER COEFFICIENT MEASUREMENTS ON Th AND ThO_2 RODS WITH NONUNIFORM TEMPERATURE DISTRIBUTIONS. Pettus, W. G.; Baldwin, M. N. Nucl. Sci. Eng., 26: 34-46(Sept. 1966).

Measurements of the Doppler effect in resonant neutron capture were made for samples having a nonuniform temperature distribution. These measurements were made on Th and ThO_2 rods of approximately $^3/_4$-in. diam. An activation technique was used, and the samples were exposed in a cadmium thimble at the center of a pool research reactor. The activated samples were dissolved and the ^{233}Pa was separated out and gamma counted. The Doppler coefficients for identical samples were determined with an axial heat source and with a peripheral heat source. In the axially heated cases, measurements were made with radial temperature drops ranging up to 185°C for the metal samples, and up to 1000°C for the oxide samples. In the peripherally heated cases, the temperature was uniform through the samples, and measurements were made with the temperature ranging up to about 350°C for both metal and oxide samples. The results show that the Doppler coefficient as a function of the average sample temperature is essentially the same for both axial and peripheral heating over the temperature range investigated. The measured values of the Doppler coefficients for the nonuniform temperature cases were $(85 \pm 5) \times 10^{-4}$ and $(95 \pm 19) \times 10^{-4}$ $(°K)^{-½}$ for Th metal and oxide, respectively. (auth)

61.

38430 (ANL-7120, pp 616-24) DOPPLER EFFECT IN ^{238}U CAPTURE AND ^{235}U FISSION IN A FAST-NEUTRON SPECTRUM. Pflasterer, G. R. Jr.; Sher, R.; Mayer, W. A.; Warzek, F. G. (General Electric Co., Pleasanton, Calif. Vallecitos Atomic Lab.).

The equipment and method used for activation measurements of the Doppler effect in the fast-neutron spectrum of the Mixed Spectrum Critical Assembly are described. Measurements were made on natural and enriched U metal foils. Results indicate a significant underprediction of the Doppler effect. This discrepancy is believed to be at least partly due to the calculated neutron spectrum being substantially too hard. Random and systematic errors in the measurements are discussed. The technique requires careful control of foil position and counting procedure, but reproducible measurements can be performed to approximately 10% of the magnitude of the observable Doppler effect. (A.G.W.)

62.

15732 (TID-21654) DOPPLER COEFFICIENT OF THORIUM OXIDE. [Quisenberry, K. S.] (Knolls Atomic Power Lab., Schenectady, N. Y.). [nd]. Contract W-31-109-eng-52. 18p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

The neutron Doppler coefficient is briefly discussed. The method of measurement for ThO_2 in the KAPL Thermal Test Reactor is briefly reviewed. The results are given. (M.O.W.)

63.

32869 INTERPRETATION OF DOPPLER AND SODIUM REACTIVITY WORTH MEASUREMENTS IN ZERO POWER REACTORS, WITH PARTICULAR REFERENCE TO THE ZEBRA DOPPLER LOOP MEASUREMENTS. Rowlands, J. L.; Wardleworth, D. (Atomic Energy Establishment, Winfrith, Eng.). 47p. (CONF-651009-28). ORAU. Gmelin, AED-CONF-65-307-46.

From Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, Argonne, Ill.

A study is made of the effect on the Doppler coefficient from plutonium-239 and of the present uncertainties in the data from 0.1 to 10 kev and, in particular, in the resonance parameter data in the unresolved resonance region. Also some of the approximations involved in the methods for comparing calculations with experimental data on the sodium coefficient are examined. (J.F.P.)

64.

2959 DOPPLER-EFFECT MEASUREMENTS ON A DILUTE FAST-CARBIDE ASSEMBLY, ZPR-6 ASSEMBLY 4Z. Till, C. E.; Lewis, R. A.; Groh, E. F. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 8: 454-5 (Nov. 1965).

65.

34627 DOPPLER-EFFECT OF $^{235}U/^{238}U$ MIXTURES. Till, C. E.; Lewis, R. A.; Groh, E. F. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 9: 259-60 (June 1966).

66.

38428 (ANL-7120, pp 610-13) RECENT DOPPLER MEASUREMENTS WITH ZPR-6. Till, C. E.; Lewis, R. A.; Groh, E. F. (Argonne National Lab., Ill.).

Improved data-gathering techniques, consisting largely of providing a servo-controlled regulating rod for the reactivity measurement fully automated with the oscillator system, improved the precision of Doppler measurements in the ZPR-6 by better than an order of magnitude. The improved techniques were used to study expansion effects and the temperature dependence of the Doppler effect. Data are summarized on the effects of environment of the sample in the reactor on the measurements, the ^{238}U Doppler effect as a function of temperature, reactivity changes for ^{235}U oxide with freely expanding and axially constrained samples as a function of temperature, reactivity changes for various isotopic ratios of ^{235}U to ^{238}U as a function of temperature, and reactivity effect of expansion of U metal. (A.G.W.)

67.

5067 TEMPERATURE COEFFICIENT OF WATER LATTICES. Wajima, J. Tsunetaka; Yamamoto, Kazuko (Hitachi, Ltd., Kawasaki-shi, Japan). J. Nucl. Sci. Technol. (Tokyo), 2: 331-9(Sept. 1965).

Temperature coefficients of slightly enriched UO_2-H_2O lattices were measured in the Ozenji Critical Facility and analyzed in terms of critical parameters. The lattices studied were mainly those of 2.5% enriched UO_2, 10 mm in diameter and clad in 0.8 mm thick Al tube, and with water to fuel volume ratios of 2.5 and 1.5. Analyses were made with Deutsch's four factor critical equation and also with three group one dimensional diffusion calculations. The result shows the importance of the influence of the reflector, which is the largest positive contribution to the temperature coefficient, being about 1×10^{-4} $\Delta k/k/°C$. This was experimentally demonstrated by measuring the coefficient for a core with an inner reflector water gap. Analyses also show that the discrepancy between experiment and calculation, which increases from 0.2 to 0.7×10^{-4} $\Delta k/k/°C$ as the water to fuel volume ratio decreases, may be mostly resolved by applying corrections for the space-dependent spectrum within the cell using the THERMOS code and for the effect of Dancoff factor with the TUZ-ZUT code. Calculation with these corrections applied reproduces experiment within an accuracy of about 0.2×10^{-4} $\Delta k/k/°C$ over the temperature range of 20 to 70°C. (auth)

68.

20198 (AEEW-R-464) FURTHER BUCKLING MEASUREMENTS ON A HEATED GRAPHITE LATTICE FUELLED WITH 0.8 PERCENT PuO_2/UO_2 CLUSTERS. Wilson, D. J. (Atomic Energy Establishment, Winfrith (England)). Oct. 1965. 26p. Dep. BIS $0.90. HMSO 4s. 6d.

The buckling of a graphite moderated stack fueled with 0.8% PuO_2/UO_2 21 rod clusters in a 14" lattice pitch was measured by the exponential method. The stack was built in the SCORPIO I facility which enabled the experiments to be made at 200 and 390°C as well as at ambient temperature. The experimental results are shown to be in very satisfactory agreement with predictions using the ARGOSY IV method. (auth)