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The Data of Nuclear Reactor Physics, 1967-1968: A Bibliography

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Compiled by

Jean Furnish

THE DATA OF NUCLEAR REACTOR PHYSICS, 1967-1968:
A BIBLIOGRAPHY

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INTRODUCTION

This bibliography is a continuation of LA-3740-MS. Nuclear Science Abstracts for 1967 and 1968 have been searched and the pertinent abstracts arranged in the following order:

I. Critical Experiments, Reasonably Homogeneous

II. Critical Experiments, Lattices

III. Reactivity Measurements

IV. Neutron Flux Spectra

V. Neutron Cross Sections

1. Sources of data
2. ENDF/B tapes and other evaluated lists
3. Wide ranges in energy
4. Capture-to-fission ratios
5. Resonances
6. Doppler effects

VI. Laboratory Summary Reports and Miscellaneous

Within each section the abstracts are grouped first by year of appearance in Nuclear Science Abstracts and then alphabetically by first author (or corporate author if no individual author is given).

Critical masses of several small metal assemblies, reported in the 1956-1966 search, are being reevaluated by H. C. Paxton and G. E. Hansen, who will publish results later. Changes in organization of the previous search are due to an expansion of neutron cross section and flux work. Of particular importance are the ENDF/B and other similar evaluated data files. Los Alamos sources for light isotope cross sections and the computer handling of data files are L. S. Stewart and R. J. LaBauve, respectively.

I. CRITICAL EXPERIMENTS

Reasonably Homogeneous

1967

35673 (ANL–7320, pp 159–84) EXPERIMENTAL AND THEORETICAL WORK AT THE ZERO-ENERGY FAST REACTOR FRO. Andersson, T. L.; Bjoereus, L.; Hellstrand, E.; Haeggblom, H.; Londen, S-O.; Tiren, L. I. (Aktiebolaget Atomenergi, Studsvik (Sweden)); Kockum, J. (Forsvarets Forskningsanstalt, Stockholm (Sweden)).

The fast reactor FRO is a split-table machine with vertical fuel elements. A quantity of 120 kg ^{235}U is available as fuel, fabricated into metallic plates of 20% enrichment. Of the first five cores studied three (numbers 1, 2, and 4) consisted of undiluted fuel. Core No. 3 was diluted with graphite (29 v/o) and No. 5 with graphite (29%) and polythene (7.5%). All assemblies had a thick reflector of Cu except for No. 4, which had a U reflector. Among the experiments carried out with FRO the following items are dealt with: critical mass, reaction ratios, central reactivity coefficients, and heterogeneity effects. Measurements of reactivity worths of multiplying as well as absorbing control rods are also summarized, including studies of interaction effects and streaming in empty channels. Recently, spectrometry work with proton-recoil counters and Doppler measurements by means of foil activation have been initiated. The latter have been carried out at 590°C, and preliminary results for ^{235}U and ^{238}U have been obtained. The experimental results have been analyzed with various models. The spectrum program SPENG, which includes a data library, has been used to calculate core and reflector fine-structure spectra and to derive multigroup cross-section sets for the different assemblies. For the critical-size and flux calculations the DSN and TDC transport theory programs have generally been employed and a second-order perturbation code has been used to analyze the central reactivity measurements. A list of 41 references is included. (auth)

35667 (ANL–7320) PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON FAST CRITICAL EXPERIMENTS AND THEIR ANALYSIS [HELD AT ARGONNE NATIONAL LABORATORY, ILLINOIS], OCTOBER 10–13, 1966. (Argonne National Lab., Ill.). Contract W-31-109-eng-38. 818p. (CONF-661019). Dep. CFSTI.

The various phases of fast reactor physics considered in the conference were: differential and group cross sections, tests of cross section sets, clean critical experiments and their analysis, evaluation and computation techniques, Doppler and Na reactivity effects, special experiments and their analysis, spectrum measurements, experimental techniques and equipment, and future programs. A total of 73 papers was presented. Abstracts were prepared for 58 of the papers; 8 papers are included under CONF-661019; abstracts for 6 papers appear in Nuclear Science Abstracts, Vol. 21, under the following abstract numbers 19151, 17520, 19469, 21765, 19468, 5683; one paper is abstracted under the report number TRG-Report-1384. (M.L.S.)

For abstracts of individual papers see: 34461, 35481–35484, 35497, 35546–35563, 35645, and 35668–35700.

35675 (ANL-7320, pp 205-14) ZPR-3 ASSEMBLY 48: STUDIES OF A DILUTE PLUTONIUM-FUELED ASSEMBLY. Broomfield, A. M.; Amundson, P. I.; Davey, W. G.; Gasidlo, J. M.; Hess, A. L.; Kenney, W. P.; Long, J. K. (Argonne National Lab., Ill.).

A series of relatively simple Pu-fueled assemblies with well-degraded spectra has been designed for study in ZPR-3. Assembly 48, the first in the series, was chosen as the subject of an international comparison of fast reactor calculation techniques. Each assembly in the proposed program will consist of a cylindrical core surrounded by a depleted U blanket. In Assembly 48 the material constituents of the core are limited to Pu, depleted U, Na, graphite, and the stainless steel present in the structure and canning of the Na and Pu plates. Graphite is included to degrade the neutron spectrum. The ratio of U to Pu in the core is 4.2. The program of measurements with Assembly 48 is still in progress. The results of the critical-mass evaluation, fission cross-section ratio, and central perturbation measurements are described. Also included is a brief statement of the results of the neutron spectrum and Doppler coefficient measurements. A list of 11 references is included. (auth)

35645 (ANL-7320, pp 186-93) MEASUREMENTS AND ANALYSIS OF Al-, Al_2O_3-, AND BeO-REFLECTED FAST CRITICAL EXPERIMENTS. Butler, D. K.; Doerner, R. C.; Knapp, W. G. (Argonne National Lab., Ill.).

The series of critical assemblies discussed was performed in connection with the program for development of a fast W reactor for nuclear propulsion. A program of critical experiments was planned to obtain some basic information about fast W-based systems. The approach selected was to begin with systems which had already been studied, such as Assemblies 11 and 22 of ZPR-3 and Assembly 1 of ZPR-6. A series of critical configurations was constructed in which only a single change of composition was made from one assembly to the next. The experiments were performed with the split-bed assembly ZPR-9. The cores were made as close approximations to cylinders as possible in the rectangular drawers of the assembly. Criticality was normally achieved by varying the core radius. In each assembly a number of measurements were made. In addition to criticality, determinations of fission-rate distribution, neutron lifetime, and central and spatial reactivity worths were made. Some time was also spent studying properties related to control, including the effect of inserting a ring of B into the reflector near the core boundary. Included are: criticality, the spatial distribution of fission rates, and the worths of various materials at the core center and as a function of radius. (auth)

35668 (ANL-7320, pp 57-64) INTERCOMPARISON OF CALCULATIONS FOR A DILUTE PLUTONIUM-FUELED FAST CRITICAL ASSEMBLY (ZPR-3 ASSEMBLY 48). Davey, W. G. (Argonne National Lab., Idaho Falls, Idaho).

An international group of establishments which are active in the field of fast reactors was invited to participate in a comparison of the calculated and measured parameters of a Pu-fueled, soft-spectrum, simple-geometry critical assembly to be constructed in Argonne National Laboratory's Zero Power Reactor-3 (ZPR-3). The study of this assembly, No. 48, in the ZPR-3 series, is still in progress. A summary of the available calculated data and the measurements obtained is presented. (auth)

30090 (ANL-7007) PHYSICS MEASUREMENTS IN TUNGSTEN-BASED, ALUMINUM-REFLECTED FAST REACTORS. Doerner, R. C.; Knapp, W. G.; Almenas, K. K.; Karam, R. A. (Argonne National Lab., Ill.). Mar. 1967. Contract W-31-109-eng-38. 38p. Dep. CFSTI.

The results of measurements made on four fast critical experiments performed in ZPR-9 summarized in support of the rocket-design effort. The fuel is highly enriched ^{235}U, the major diluent is a W-Re alloy, and the reflector material is Al. Data on cross section sets for W, Re, and ^{10}B are included. (J.C.W.)

35663 (AAEC/E-177) BUCKLING AND INTEGRAL SPECTRUM MEASUREMENTS IN ^{239}Pu/BeO SUB-CRITICAL ASSEMBLIES. Duerden, P. (Australian Atomic Energy Commission Research Establishment, Lucas Heights). June 1967. 46p. Dep.

The materials buckling of four BeO moderated ^{239}Pu-Al alloy fuelled systems having BeO-^{239}Pu atomic ratios of 1707, 2499, 3749 and 4999 have been measured by the exponential method. Relative fission rates of ^{233}U, ^{235}U and ^{239}Pu were also measured in the equilibrium spectrum region of the same assemblies. Because of the heterogeneous nature of the assemblies, fine structure corrections were applied. Some calculations using the CRAM diffusion code and the GYMEA code are included. (auth)

38481 ANALYSIS OF PuO_2-UO_2 CRITICAL EXPERIMENTS. Eich, W. J. (Westinghouse Electric Co., Pittsburgh). Trans. Amer. Nucl. Soc., 10: 306-7(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

44946 (RFP-1017) CRITICAL MASSES OF OIL REFLECTED, ENRICHED URANIUM METAL ASSEMBLIES WITH POLYURETHANE CENTERS. Ernst, B. B. (Dow Chemical Co., Golden, Colo. Rocky Flats Div.). Sept. 6, 1967. Contract AT(29-1)-1106. 9p. Dep. CFSTI.

The critical masses of oil reflected, enriched uranium spherical assemblies of inside uranium radii, 0.0, 4.017, 8.010, and 12.011 cm with a low-density foam in the central cavity were measured to be 24.3, 31.65, 52.1, and 81.3 kg. Critical masses were determined by reciprocal multiplication measurements on fully reflected assemblies and are compared with calculations. (auth)

46944 (IN-1120) COMPARISON OF ORNL CLEAN CRITICAL EXPERIMENTS WITH CALCULATIONS. Fox, J. K. (Idaho Nuclear Corp., Idaho Falls). Sept. 1967. Contract AT(10-1)-1230. 23p. Dep. CFSTI.

The accuracy of several reactor codes has been determined for a variety of criticality problems that are of interest in criticality safety. This was done by comparisons of the calculations with data on clean critical experiments performed at ORNL. All of the systems studied were moderated to some degree by hydrogen. Most of the comparisons are with data on highly enriched U fueled cores, although a few were with 2 to 5 percent enriched fuels. A four-group structure was used in all cases. Transport theory was used only for obtaining flux-weighted cross sections. The tabulated results indicate that with properly weighted constants eigenvalue calculations using diffusion theory agree well with experiments. 25 references. (auth)

27667 (GA-5805) ADVANCED BERYLLIUM OXIDE CONCEPTS. Progress Report for the Quarter Ending September 30, 1964. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Oct. 30, 1964. Contract AT(04-3)-187. 49p. Dep. CFSTI.

Efforts during the quarter were concentrated on specific problems related to the specifications for the first test module and its subsequent insertion into EBOR. The development of a suitably fueled BeO matrix also continued with particular emphasis on determining the type of samples to be incorporated in the next irradiation capsules. Cross section work and critical assembly analytical calculations were also done as preliminary checks on Pu isotope cross sections data prior to initiating a conceptual design study of a Pu-fueled EBOR core. An examination of some alternate cladding materials was undertaken. Information is included concerning Th utilization, Pu utilization, and EBOR test module development. (J.R.D.)

1967

35669 (ANL-7320, pp 66-78) IMPLICATIONS OF RECENT FAST CRITICAL EXPERIMENTS ON BASIC FAST REACTOR DESIGN DATA AND CALCULATIONAL METHODS. Greebler, P.; Gyorey, G. L.; Hutchins, B. A.; Segal, B. M. (General Electric Co., San Jose, Calif.).

Recent experimental information from dilute, Pu-fueled critical assemblies was used to test and to provide guidance for improving fast reactor design data and calculational methods. Most of the experimental data used in this study are taken from ZPR-III Assembly 47 (the SEFOR core mockup). The important nuclear reactor parameters were calculated with a number of variations in nuclear data and calculational techniques. An analysis of the experimental and calculated results shows that by careful adjustment of the important cross sections, a much closer agreement can be achieved between calculations and experiments than that heretofore reported. A complete evaluation was made of the cross-section and resonance parameter data for Pu-239, the most important isotope in this case. The new data yield good agreement with the critical mass of Assembly 47. They result in a calculated Pu-239 Doppler effect that is essentially zero, in agreement with the measured values. A two-dimensional calculation of the neutron lifetime, using group constants that adequately account for the spatial variation of the neutron spectrum, yields a significant improvement over that based on a one-dimensional model. A list of 40 references is included. (auth)

8077 MEASUREMENT OF THE FAST-IMPORTANCE FUNCTION DISTRIBUTION AND OF THE GEOMETRIC BUCKLING EMPLOYING A ^{252}Cf SPONTANEOUS FISSION SOURCE. Greenspan, Ehud; Cady, K. B.; Aderhold, H. C. (Cornell Univ., Ithaca, N. Y.). Trans. Amer. Nucl. Soc., 9: 491-2(Oct.-Nov. 1966).

24183 (BNWL-SA-983) CRITICALITY OF PLUTONIUM COMPOUNDS IN THE UNDERMODERATED RANGE, H/Pu ≤ 20. Hansen, L. E.; Clayton, E. D. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Oct. 27, 1966. Contract AT(45-1)-1830. 21p. DTIE.

Nuclear criticality control involves the application of criticality data to those operations involving fissile materials. A variety of plutonium compounds are known to exist for which experimental criticality data are particularly lacking. In order to obtain data for nuclear safety guidance, a series of calculations were made to determine the bare and water reflected spherical critical masses for 12 of these compounds in the undermoderated range (H : Pu ≤ 20). Verification of the calculations was made for the limited criticality data on homogeneous PuO_2 systems in the undermoderated range, and comparisons also made for three heterogeneous plutonium fueled assemblies. (auth)

34161 CRITICALITY OF PLUTONIUM COMPOUNDS IN THE UNDERMODERATED RANGE, H : Pu ≤ 20. Hansen, L. E.; Clayton, E. D. (Battelle Memorial Inst., Richland, Wash.). Contract AT(45-1)-1830. Nucl. Appl., 3: 481-7(Aug. 1967).

Experimental criticality data do not exist for most plutonium compounds. To obtain guidelines for nuclear criticality safety use, a survey utilizing transport-theory calculations was made to determine the critical masses of bare and water-reflected spheres as a function of density and H : Pu ratio for 12 of these compounds in the undermoderated range (H : Pu ≤ 20). The compounds considered were: PuH_2, PuH_3, PuN, PuC, Pu_2C_3, PuO_2, Pu_2O_3, PuF_3, PuF_4, $PuCl_3$, $Pu(NO_3)_4$, $Pu(C_2O_4)_2$. Also derived were core density exponents which permit critical masses to be predicted for compounds with densities ranging down to one-fifth of their theoretical values. The validity of the calculations was examined by comparing results with the limited criticality data on homogeneous PuO_2 systems in the undermoderated range. Comparisons were also made for Pu metal systems and for three heterogeneous Pu-fueled assemblies. (auth)

38760 CRITICALITY OF PLUTONIUM COMPOUNDS IN THE UNDERMODERATED RANGE, H : Pu ≤ 20. Hansen, L. E.; Clayton, E. D. (Battelle-Pacific Northwest Lab., Richland, Wash.). Trans. Amer. Nucl. Soc., 10: 307-8(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11–15, 1967. See CONF-670602.

35384 (ANL-7320, pp 88-106) OPTIMISATION OF NEUTRON CROSS-SECTION DATA ADJUSTMENTS TO GIVE AGREEMENT WITH EXPERIMENTAL CRITICAL SIZES. Hemment, Pamela C. E.; Pendlebury, E. D. (Atomic Weapons Research Establishment, Aldermaston (England)).

A method is described which enables adjustments of group cross sections to be calculated in an optimum way to fit experimental critical sizes when resonance self-shielding is not important. The method uses a least-squares fitting procedure and takes into account the experimental uncertainties on the cross-section data and the critical sizes. It is fully mechanized for use with the IBM-7030 and has been shown to work satisfactorily. The machine programs involved are briefly described and an account given of some results obtained. The extension of the method to take into account other integral data, such as spectra and reaction-rate measurements, in critical systems is discussed along with a way of dealing with resonance self-shielding. A list of 17 references is included. (auth)

8093 MEASUREMENTS OF k ∞ FOR A PuAl–THORIA SUPERCELL. Hill, N. A. (Battelle-Northwest, Richland, Wash.). Trans. Amer. Nucl. Soc., 9: 447(Oct.–Nov. 1966).

38696 EXPERIMENTAL RESULTS FROM LARGE-CAVITY REACTOR CRITICAL EXPERIMENT. Hyland, R. E. (Lewis Research Center, Cleveland); Pincock, G. D.; Kunze, J. F.; Wood, R. E. Trans. Amer. Nucl. Soc., 10: 8-9(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11–15, 1967. See CONF-670602.

23606 (LA-3651) CRITICAL MASS REDUCTION. Jarvis, George A.; Mills, Carroll B. (Los Alamos Scientific Lab., N. Mex.). Dec. 1, 1966. Contract W-7405-eng-36. 14p. Dep mn. CFSTI $3.00 cy, $0.65 mn.

Further study into the critical mass of nuclear reactors is outlined. Results of the study are presented. (M.L.S.)

10507 (ORNL-P-2741) THE NATIONAL CRITICALITY DATA CENTER. Johnson, E. B. (Oak Ridge National Lab., Tenn.). [1966]. Contract W-7405-eng-26. 10p. (CONF-661206-2). Dep. mn. CFSTI $3.00 cy, $0.65 mn.

From American Nuclear Society, National Topical Meeting, Nuclear Criticality Safety, Las Vegas, Nev.

A Criticality Data Center has been established at the Oak Ridge National Laboratory under the sponsorship of the USAEC for the purpose of collecting from sources, both in the United States and abroad, information applicable to criticality safety problems. The principal output of the Center is typified by TID-7028, "Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U," and TID-7016, "Nuclear Safety Guide," both of which were originally the results of group efforts not under the sponsorship of the Center. The former document summarizes most of the data available at the time of its publication and will require frequent expansion. The latter document is presently under revision to incorporate the results of more recent measurements and to reflect the development of reliable theoretical analysis. Both documents are internationally known and used. (auth)

8069 CRITICAL EXPERIMENTS WITH PuO_2-UO_2 FUEL AND D_2O MODERATOR. Kutcher, J. W.; Lauby, J. H.; Purcell, W. L.; Schmid, L. C.; Williams, L. D.; Worden, J. R. (Battelle-Northwest, Richland, Wash.). Trans. Amer. Nucl. Soc., 9: 418-9(Oct.-Nov. 1966).

8065 ANALYSIS OF PLUTONIUM/URANIUM ALLOY-FUELED HEATED-GRAPHITE EXPONENTIAL EXPERIMENTS. Lepecki, W. (Instituto de Pesquisas Radioativas da Universidade, Belo Horizonte, Brazil). Trans. Amer. Nucl. Soc., 9: 516-17 (Oct.-Nov. 1966).

14068 (BNWL-347) MULTIGROUP ANALYSIS OF SELECTED FAST CRITICAL ASSEMBLIES. Little, W. W. Jr.; Hardie, R. W.; Maas, L. L. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Dec. 1966. Contract AT(45-1)-1830. 29p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

A modified version of the Russian data compilation is used to compute nuclear parameters for various fast critical assemblies. Calculated and experimental values of critical mass, fission ratios, and reactivity coefficients are compared. For the moderate volume Pu and U assemblies analyzed, the data predict k_{eff} within about 1.5%. (auth)

8063 CAVITY REACTOR CRITICAL EXPERIMENT. Lofthouse, J. H. (General Electric Co., Idaho Falls, Idaho); Pincock, G. D.; Kunze, J. F.; Wood, R. E.; Hyland, R. E. Trans. Amer. Nucl. Soc., 9: 340(Oct.-Nov. 1966).

40671 (AWRE-R-4/63) VERA ^{235}U-GRAPHITE REACTOR EXPERIMENT. Interim Report. McTaggart, M. H.; Goodfellow, H.; Paterson, W. J.; Weale, J. W. (Atomic Weapons Research Establishment, Aldermaston (England)). [nd]. 55p. Dep. CFSTI. UK.

Experimental work on a number of small fast reactor assemblies with U-235/graphite cores and natural U reflectors is presented. The object of the experiments was to provide detailed information against which the nuclear data for U-235 in the energy region from 500 to 50 keV could be checked. Critical masses were measured for five assemblies. (auth)

46643 (ORNL-4134) NEUTRON PHYSICS DIVISION ANNUAL PROGRESS REPORT FOR PERIOD ENDING MAY 31, 1967. (Oak Ridge National Lab., Tenn.). Aug. 1967. Contract W-7405-eng-26. 150p. Dep. CFSTI.

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CRITICALITY STUDIES—critical mass and volume of reflected and unreflected paraffin-moderated uranium-tetrafluoride, (E)
NEUTRONS, PROMPT—decay constants in reflected and unreflected paraffin-moderated uranium-tetrafluoride assemblies, (E)
URANIUM FLUORIDE UF_4—criticality studies on reflected and unreflected paraffin-moderated assemblies of, (E)

REACTORS, THERMAL—criticality calculations for, using slightly enriched uranium
CRITICALITY STUDIES—critical mass of polyethylene-moderated and unmoderated enriched-uranium assemblies, (E)
URANIUM—critical mass of polyethylene-moderated and unmoderated assemblies of enriched, (E)
URANIUM-235—critical mass of polyethylene-moderated and unmoderated assemblies of, (E)
CRITICAL ASSEMBLIES—critical mass of polyethylene-moderated and unmoderated enriched-uranium, (E)

35549 (ANL-7320, pp 270-5) USE OF INTEGRAL MEASUREMENTS AS SUPPLEMENTARY DATA IN NEUTRON CROSS-SECTION EVALUATION. Pazy, A.; Rakavy, G.; Reiss, Y.; Yeivin, Y. (Hebrew Univ., Jerusalem (Israel)).

The formulation of an exact method for improvement of microscopic cross-section evaluation by means of integral experiment data is presented. This formulation utilizes a generalized least squares method. A simple numerical example is used to illustrate the method. (M.L.S.)

35693 (ANL-7320, pp 550-9) THE CRITICAL EXPERIMENTS AND PRELIMINARY ANALYSIS OF MULTIFUELED, NONUNIFORM CORE LOADINGS FOR THE FARET PROGRAM. Persiani, P. J.; Hess, A. L.; Kucera, D. (Argonne National Lab., Ill.).

A series of small, Pu-plus-U-235-fueled fast reactor cores with steel radial and axial reflectors were constructed in Argonne's Zero Power Reactor-3 as part of the design program for the FARET reactor. These studies, designated ZPR-3 Assembly 46, were essentially mockups of possible loadings of the FARET core. The primary objective of the studies was to confirm the physics analysis of multifueled, nonuniform core loadings as were envisaged for the FARET system. A principal interest in the studies was to establish experimentally the predicted reactivity control afforded by the control-rod designs for FARET. The agreement obtained between calculations and the results of experiments for control-rod worth was sufficient to establish the range of control possible in the FARET reactor. Of equal importance was the substantiation of the neutronic behavior of a mixed core in FARET when subassemblies of different types of fuels were interchanged. Analytical calculations were done for all experiments, and the methods for analysis that were adopted are discussed. (auth)

15765 (CONF-661019-10) RECENT FAST CRITICAL EXPERIMENTS IN THE MSCA AND 710-CE. Petersen, G. T.; Kunze, J. F.; Wall, I. B.; Hallum, J. W.; Henderson, W. B.; Warzek, F. G. (General Electric Co., Pleasanton, Calif. Nuclear Technology Dept.). Oct. 1966. Contracts AT(04-3)-189; AT(40-1)-2847. 28p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

From International Conference on Fast Critical Experiments and Their Analysis, Argonne, Ill.

The fast core of the Mixed Spectrum Critical Assembly (MSCA) or the Vallecitos Atomic Laboratory, contains a loading of enriched UO_2, natural UO_2, and Inconel. The neutron spectrum in this assembly is spatially asymptotic and representative of a dilute fast ceramic reactor. Measurements reported include fission rates of ^{233}U, ^{234}U, ^{235}U, ^{236}U and ^{237}Np. Neutron life-time determined by pulsed neutron and 1/V poison substitution, and reactivity worths determined by pile oscillator and direct period measurements are reported. At the Idaho Test Station, work in the 710-CE concentrated on small, hard spectrum, refractory metal critical experiments related to space power reactor applications. Be reflected cores in the 20 to 50 liter range, containing essentially equal volume mixtures of W, fully enriched U metal and Ta are studied. Data from the first basic critical experiment include central fission ratios, reflector effects on power distribution and neutron lifetime, and relative reactivity worths. (auth)

35674 (ANL-7320, pp 195-203) RECENT FAST CRITICAL EXPERIMENTS IN THE MSCA AND 710-CE. Petersen, G. T.; Kunze, J. F.; Wall, I. B.; Hallam, J. W.; Henderson, W. B.; Warzek, F. G. (General Electric Co., San Jose, Calif. Nuclear Technology Dept.).

The fast core of the Mixed Spectrum Critical Assembly contains a loading of 405 kg $^{235}UO_2$, 1540 kg $^{238}UO_2$, and 1060 kg of Inconel in its 400-liter fast-core matrix. Both calculation and experiment demonstrate that the neutron spectrum in this assembly is spatially asymptotic and representative of a dilute fast ceramic reactor. Recent measurements to be reported in this paper include fission rates of ^{233}U, ^{234}U, ^{235}U, ^{236}U, and ^{237}Np, and neutron lifetime as determined by: pulsed neutron and 1/v poison substitution, and reactivity worths as determined by pile oscillator, and direct period measurements. Work in the 710-CE at the Idaho Test Station has been concentrated on small, hard-spectrum, refractory metal critical experiments related to space power reactor applications. In particular, Be-reflected cores in the 20- to 50-liter range, containing essentially equal volume mixtures of W and fully enriched U metal and some Ta, have been stûdied Some of these assemblies provide necessary design data for compact space reactors, while others are primarily devoted to basic physics measurements and furnish very useful integral data for evaluation of higher-energy cross sections and specialized transport problems Data reported in this paper are taken from the first basic critical experiment and include: central fission ratios, reflector effects on power distribution and neutron lifetime, and relative reactivity worths. A list of 14 references is included. (auth)

14072 (GEMP-472) TEST PROGRAM AND PROCEDURES FOR A MOCKUP AND UF_6 NASA CAVITY REACTOR CRITICAL EXPERIMENT AT THE LOW POWER TEST FACILITY. Pincock, G. D. (General Electric Co., Cincinnati, Ohio. Nuclear Materials and Propulsion Operation). Dec. 23, 1966. Contract AT(40-1)-2847. 19p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

The test program and procedures for an accurate mock-up of a gaseous UF_6 experiment in the cavity reactor using sheet fuel followed by an actual UF_6 experiment are documented. The fuel radius in both cases will be 24 inches or 0.67 of the cavity radius. Measurements will be made to determine k excess, rod worth, reactor material worths such as fuel, Al, etc., and power and flux distributions in the cavity and reflector regions. The purpose of the experiments is to determine absolute differences between the mock-up reactor using solid sheet fuel and the same system containing gaseous UF_6. (auth)

44934 (BNWL-SA-780) THEORETICAL ANALYSES OF HOMOGENEOUS PLUTONIUM CRITICAL EXPERIMENTS. Richey, C. R. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Oct. 3, 1966. Contract AT(45-1)-1830. 16p. Dep. CFSTI.

A computational analysis of data from an accumulation of clean experiments with plutonium fueled assemblies is presented. Simplification approximations of the transport equation and cross section data are evaluated. Associated errors are predicted. Critically safe masses and dimensions are presented for aqueous $Pu(NO_3)_4$ solutions and ^{239}Pu-water mixtures. A theory-experiment comparison is presented. Calculated eigenvalues are tabulated as a function of S_N order and as a function of anisotropic scattering order. 11 references. (M.L.S.)

44947 (RFP-1021) CRITICAL MASSES FOR PARTIALLY STEEL REFLECTED ENRICHED URANIUM METAL ASSEMBLIES. Rothe, Robert E. (Dow Chemical Co., Golden, Colo. Rocky Flats Div.). Sept. 18, 1967. Contract AT(29-1)-1106. 16p. Dep. CFSTI.

Sixty-one critical masses were measured for spherical or hemispherical enriched U assemblies. The assemblies differed in the amount of mild-steel reflector outside the U. They also differed in the size of a central cavity in the U. The cavity was formed by the omission of small-radius U components from the assembly. This central cavity contained either air or mild steel. The critical masses were determined by the extrapolation of reciprocal multiplication data for subcritical assemblies. (auth)

35676 (ANL-7320, pp 231-6) PHYSICS PARAMETERS OF LARGE DILUTE URANIUM CARBIDE CORES. Rusch, G. K.; Karam, R. A.; Kato, W. Y.; Main, G. W. (Argonne National Lab., Ill.).

Descriptions of the cores, critical masses, sodium-coefficient studies, and radial reactivity worths of core materials of large dilute uranium carbide Assemblies 4Z and 5 in ZPR-6 and Assembly 11 in ZPR-9 are presented. Assemblies 4Z and 11 are zoned core systems. Assembly 5 is a 2600 liter carbide reactor. The cores of all three Assemblies are similar. (M.L.S.)

8072 FAST-SPECTRUM REFRACTORY-METAL CRITICAL-EXPERIMENT MEASUREMENTS. Sims, F. L. (General Electric Co., Idaho Falls, Idaho); Kunze, J. F.; Walsh, W. P.; Henderson, W. B. Trans. Amer. Nucl. Soc., 9: 488-9(Oct.-Nov. 1966).

1967

21653 (WAPD-TM-621) ANNULAR SEED-BLANKET REACTOR CRITICAL EXPERIMENTS. Smith, G. G.; Beck, J. W.; Glickstein, S. S.; (and others) (Bettis Atomic Power Lab., Pittsburgh, Pa.). Feb. 1967. Contract AT(11-1)-Gen-14. 59p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

A series of one-, two-, and three-module cores containing highly enriched UO_2 seeds and ThO_2 blankets have been studied. The purpose of this program was to compare design calculations with various measured core parameters. These parameters included the critical eigenvalues, seed power distributions, fast and thermal activation traverses, epithermal to thermal Th capture ratios, epithermal to thermal ^{235}U fission ratios, thermal disadvantage factors, and fast advantage factors. In addition, some information on core intermodule coupling was obtained. Each module of the assemblies consisted of a narrow hexagonal annular ^{235}U seed and an inner and outer ThO_2 blanket. The cores were designed to be nearly clean critical with no significant internal structure except fuel rod cladding in the active portion of the core. This allowed a fair test of the design model on highly absorbing narrow seed regions in a ThO_2 blanket. The design model was found to agree quite well with experimental results. The critical eigenvalues for all cores were consistent and close to unity. Near the seed-blanket interface, discrepancies between calculated and experimental traverses were noted and have been explained by spectrum-weighted cross sections and a higher order approximation to transport calculations. Monte Carlo calculations gave good agreement with experiment for thermal disadvantage factors and fast advantage factors. Fast leakage effects were found to be important in the calculation of the fast advantage factor in the seed region. (auth)

38533 CRITICAL EXPERIMENTS WITH THE UO_2-2 WT PERCENT PuO_2 BATCH CORE IN THE PRTR. Smith, R. I.; Kutcher, J. W.; Lauby, J. H. (Battelle-Pacific Northwest Lab., Richland, Wash.). Trans. Amer. Nucl. Soc., 10: 185-6(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

10592 (LA-DC-8386) CORRELATIONS OF EXPERIMENTS AND CALCULATIONS. Stratton, William R. (Los Alamos Scientific Lab., N. Mex.). [1966]. Contract W-7405-eng-36. 43p. (CONF-661206-3). Dep. mn. CFSTI $3.00 cy, $0.65 mn.

From American Nuclear Society, National Topical Meeting, Nuclear Criticality Safety, Las Vegas, Nev.

A theoretical technique to interpret critical data using one-dimensional codes for spheres, slabs, and infinite cylinders of ^{235}U, ^{233}U, and ^{239}Pu was compared with data determined experimentally. Calculations for highly enriched ^{235}U thermal and fast systems, even though cross sections for fast and epithermal cores were chosen, gave satisfactory results. A tendency for progressive overestimates of critical radii with decreasing enrichment was seen. A trend of increasing error with decreasing $H/^{235}U$ ratio for systems enriched to 5% ^{235}U or less was noted. For a given ^{235}U enrichment, a calculational bias may be determined and applied with confidence. (F.S.)

44941 (LA-3612) CRITICALITY DATA AND FACTORS AFFECTING CRITICALITY OF SINGLE HOMOGENEOUS UNITS. Stratton, W. R. (Los Alamos Scientific Lab., N. Mex.). July 1964. Contract W-7405-eng-36. 53p. Dep. CFSTI.

The critical parameters of single homogeneous units are examined and tabulated. The study includes both theoretical and experimental results which are compared extensively in order to establish the accuracy of the theoretical method. The experimental data are reduced to standard conditions to facilitate this comparison and to investigate the consistency of the large number of critical experiments. Given the validity of the calculational scheme, the various affects of diluents (including moderators), reflectors, density charges, and poisons are studied. Finally, by application of the theory, results are obtained which are inaccessible or very difficult to obtain by experimental methods. (auth)

38669 (RFP-907) ENRICHED URANIUM-METAL MEASUREMENTS, NO. 1. Tuck, Grover (Dow Chemical Co., Golden, Colo. Rocky Flats Div.). July 26, 1967. Contract AT(29-1)-1106. 38p. Dep. CFSTI.

The critical masses of oil-reflected and moderated enriched uranium (93.12%, uranium-235) spherical and hemispherical shell assemblies have been measured at inside radii of 0, 4, 6.67, 8, and 12 centimeters (cm). The measurements are described and compared to calculated values for spherical assemblies. (auth)

8067 CRITICALITY OF AQUEOUS SOLUTIONS OF 5 PER CENT ENRICHED URANIUM. Webster, J. W.; Johnson, F. R. (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 9: 514-15(Oct.-Nov. 1966).

38664 (KAPL-M-6701) CATALOGUE OF SHA EXPERIMENTS AND CALCULATIONS. Supplement I. Weinstein, S.; Reiter, R. A. (Knolls Atomic Power Lab., Schenectady, N. Y.). May 1967. Contract W-31-109-eng-52. 18p. Dep. CFSTI.

Eight additional SHA critical configurations were assembled using the first and second solid homogeneous fuel materials and an assortment of internal and external reflectors. The experimental values of the effective multiplication constants, dimensional drawings of the systems, and details of their construction are presented as a supplement to the SHA catalogue. (auth)

1968

22799 (AHSB(S)Handbook-5(Pt.1)) HANDBOOK OF EXPERIMENTAL CRITICALITY DATA. PART I. Chapters 1 to 4. Abbey, F. (comp.) (United Kingdom Atomic Energy Authority, Risley (England). Authority Health and Safety Branch). 1967. 122p. Dep. CFSTI; UK.

Criticality data from the literature are compiled into tabulated form. These tables are broken down into: single unmoderated ^{235}U cores; single unmoderated plutonium cores; single ^{235}U cores moderated by deuterium, beryllium, or carbon; and single plutonium cores moderated by deuterium, beryllium, or carbon. Data are given various geometric configurations. 67 references. (M.L.S.)

32958 (AHSB(S)Handbook-5(Pt.2)) HANDBOOK OF EXPERIMENTAL CRITICALITY DATA. PART 2. CHAPTERS 5 AND 6. Abbey. F. (comp.) (United Kingdom Atomic Energy Authority, Risley (England). Authority Health and Safety Branch). 1968. 124p. Dep. CFSTI. UK.

Criticality data are tabulated for: single ^{235}U cores moderated by hydrogen and single Pu cores moderated by hydrogen. The results for highly enriched systems are categorized according to: unreflected spheres of aqueous UO_2F_2 and $UO_2(NO_3)_2$ and spherical systems reflected with water and polyethylene; aqueous cylindrical systems—both reflected and unreflected; reflected and unreflected aqueous parallelepiped cores; uranium metal cores diluted with lucite and with lucite-graphite; and annular cylindrical aqueous cores. The results for low and intermediate enrichment systems are categorized as: bare aqueous cores of spherical, cylindrical, and rectilinear parallelepiped configurations; single material reflected systems; and composite reflectors. Criticality data for heterogeneously poisoned aqueous systems are tabulated. Pu fueled aqueous systems are categorized according to spherical, cylindrical, or parallelepiped. 57 references. (M.L.S.)

35423 (APDA-224) QUARTERLY TECHNICAL PROGRESS REPORT ON AEC-SPONSORED ACTIVITIES, JANUARY–MARCH 1968. (Atomic Power Development Associates, Inc., Detroit, Mich.). Contract AT(11-1)-865. 87p. Dep. CFSTI.

CRITICAL ASSEMBLIES—physics measurements for plutonium-fueled ZPR-3 Assembly 48, analysis of one-dimensional, 24-group spherical-geometry; physics measurements for uranium-fueled ZPR-6 Assembly 2, analysis of one-dimensional 24-group spherical-geometry

12087 (BNWL-472, pp 5.1-16) CRITICAL MASS PHYSICS. *(Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).*

Criticality experiments, bare and water reflected, were continued with the 42 × 42 in. slab assembly having adjustable thickness. The plutonium concentration was 58 g/liter and acid molarities were 2.3 and 5. The plutonium contained 4.8 wt % ^{240}Pu. Evaluation of the effect of vessel walls, lattice reinforcement, and hood walls was made. *Correlation of slab experimental data and initial calculation theory were in poor agreement.* The clean critical bare and water reflected infinite slab was experimentally estimated to be 15.7 and 10.1 cm respectively for 58 g Pu/liter at an acid molarity of 2.3. Criticality experiments were performed to provide data for nuclear safety guidance on handling, storing, and shipping of *United Kingdom nut type casks containing* 6.6 kg plutonium metal each. Experiments were carried out with the nine casks bare and reflected with Lucite. Effect of Lucite moderator and cadmium plating was studied qualitatively. The bare array indicated criticality at about 19 casks; the reflected array gave 10½ casks for criticality. Critical bucklings and masses were measured for a range of *lattice spacings of* 2.1 wt % enriched U fuel tubes in light water. Criticality experiments were performed in support of the Gas Cooled Fast Breeder Reactor (GCFR) program. The experiments were designed to simulate water entry into the GCFR core and to check basic neutronic data and computational techniques. An experimental program to provide data for determining the *minimum critical* ^{235}U *enrichment* of hydrogenous uranyl nitrate systems was completed in the PCTR. Data reduction and analysis are currently in progress. Some experimental results are presented as raw data. The dead time problem connected with Rossi-alpha measurement using multi-channel equipment was circumvented by using, essentially, a multiple single channel approach. A *system was assembled employing* fast solid state equipment in hope that it will be useful for epithermal and fast neutron systems. A system for experimentally measuring the P_0 probability of reactor noise for various time intervals was assembled in hopes of providing an independent measurement of the prompt neutron decay constant. A new series of critical experiments was begun at the Critical Mass Laboratory with PuO_2-polystyrene compacts and the Remote Split-Table Machine. These experiments are a continuation of the basic research program to provide data for evaluating the effects of moderation and ^{240}Pu on intermediate neutron spectra plutonium systems. The current series of experiments are concerned with fuel having an atomic H/Pu of 5 and a ^{240}Pu isotopic content of 11.5 wt %. (auth)

22804 (BNWL-685) REACTOR PHYSICS DEPARTMENT TECHNICAL ACTIVITIES QUARTERLY REPORT, OCTOBER, NOVEMBER, DECEMBER 1967. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Mar. 1968. Contract AT(45-1)-1830. 108p. Dep. CFSTI.

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CRITICAL ASSEMBLIES—critical mass measurements in the Critical Approach Facility; neutron buckling measurements in the Critical Approach Facility; control rod reactivity worth measurements in the Critical Approach Facility; neutron spectrum measurements in heterogeneous plutonium fueled, integral

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CRITICALITY STUDIES—critical measurements using plutonium nitrate in slab geometry

1968

6114 STUDIES OF CRITICAL ASSEMBLIES OF HOMOGENEOUS MIXTURES OF PLUTONIUM OXIDE AND POLYSTYRENE. Baxter, Alan M. (General Atomic Div., General Dynamics Corp., San Diego, Calif.); Clayton, E. D.; Hansen, L. E. Trans. Amer. Nucl. Soc., 10: 536(Nov. 1967).
From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

6119 MINIMUM CRITICAL ^{235}U ENRICHMENT FOR URANYL NITRATE HYDROGENOUS SYSTEMS. Bierman, S. R.; Hess, G. M. (Battelle-Pacific Northwest Lab., Richland, Wash.). Trans. Amer. Nucl. Soc., 10: 539-40(Nov. 1967).
From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

21149 MINIMUM CRITICAL ^{235}U ENRICHMENT OF HOMOGENEOUS, HYDROGENOUS URANYL NITRATE SYSTEMS. Bierman, S. R.; Hess, G. M. (Battelle Memorial Inst., Richland, Wash.). Nucl. Sci. Eng., 32: 135-9(Apr. 1968). (BNWL-SA-1379).
The data presented establish the minimum critical enrichment of ^{235}U in homogeneous uranyl nitrate at 2.104 with a standard deviation of 0.010 wt %. This results in a lower limit of 2.07 wt % at the 99% confidence level. Optimum neutron moderation for 2.14, 2.26, and 3.04 wt % enriched uranyl nitrate homogeneous systems occurs at H/U ratios of 8.0 ± 1.0, 9.3 ± 0.5, and 10.5, respectively. The minimum critical enrichment is the enrichment required to obtain an infinite neutron multiplication factor of unity under conditions of optimum moderation. (M.C.G.)

33062 CRITICAL EXPERIMENTS WITH HOMOGENEOUS PuO_2-POLYSTYRENE AT 5 H:Pu. Bierman, S. R.; Hansen, L. E.; Lloyd, R. C.; Clayton, E. D. (Brookhaven National Lab., Upton, N. Y.). Trans. Amer. Nucl. Soc., 11: 380-1(June 1968).
From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

37530 (BNWL-SA-1630) CRITICAL EXPERIMENTS WITH HOMOGENEOUS PuO_2-POLYSTYRENE AT 5 H:Pu. Bierman, S. R.; Hansen, L. E.; Lloyd, R. C.; Clayton, E. D. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). May 21, 1968. Contract AT(45-1)-1830. 13p. (CONF-680601-18). Dep. CFSTI.

From 14th Annual Meeting of the American Nuclear Society, Toronto, Ontario.
Criticality parameters for mixtures of fuel having an 11.5 wt % ^{240}Pu isotopic concentration and an atomic H/Pu ratio of 5 are presented. Experimental data are obtained from both bare and reflected rectangular parallelepipeds of PuO_2-polystyrene fuel. (D.C.C.)

35655 COMPARISON OF MEASUREMENTS IN SNEAK-1 AND ZPR III-41. Boehme, R.; Barleon, L.; Boehnel, K.; (and others) (Kernforschungszentrum, Karlsruhe, Ger.). pp 55-77 of Fast Reactor Physics. Vol. II. Vienna, International Atomic Energy Agency, 1968.
From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.2); CONF-671043-(Vol.2).
The experimental program of the Karlsruhe Fast Zero Power Reactor SNEAK started in the autumn, 1966 with measurements on a 400-liter uranium assembly, a mock-up of ZPR III-41. During a four-month period the experimental installations and techniques of SNEAK were successfully tested. The installations include a movable drawer connected to an automatic sample changer operating in a horizontal experimental channel, a vertical drive unit, a pile oscillator, and a pulsed neutron generator. The techniques used include spectra measurements with proton recoil counters and foil activation, and several techniques for determining reactor power and $\beta/1$, e.g., Rossi- and pulsed neutron source measurements. In the experimental program quantities such as critical mass, reaction rate ratios, neutron spectrum, material worths in the center, radial and axial traverses, $\beta/1$, and reactor power were determined The results were generally in good agreement with those of the ZPR III experiments, and the remaining discrepancies are discussed. These are partly due to small deviations in the material composition of the two assemblies. The experimental data are also compared with calculations using the 16-group YOM, 26-group KFK, and 26-group ABN cross-section sets. While critical mass is best calculated with YOM, the YOM spectrum is too hard, and both KFK and ABN give better agreement with the experimental spectrum. (auth)

25078 (CEA-R-3367) EXPERIENCES DE CRITICITE REALISEES AVEC UNE SOLUTION HOMOGENE DE PLUTONIUM. RESULTATS EXPERIMENTAUX. INTERPRETATIONS THEORIQUES. (Critical Experiments Carried Out with a Homogeneous Plutonium Solution. Experimental Results. Theoretical Interpretations). Bouly, Jean Claude; Caizergues, Robert; Deilgat, Edouard; Houelle, Michel; Lecorche, Pierre (Commissariat a l'Energie Atomique, Saclay (France). Centre d'Etudes Nucleaires). Dec. 1967. 90p. (In French). Dep.
Results of a series of experimental and theoretical criticality studies on plutonium are given. A comparison of theoretical and experimental values for critical heights of solutions is made; effects of nitrogen, introduced in the form of the nitrate ion, on the reactivity of the fissile media are evaluated; the effects of ^{240}Pu on the reactivity of the media are analyzed. Influence of moderators which are introduced into the solution is investigated; effects of dimensions of the inner cavity of annular cylinders are analyzed. (auth)

48735 (NP-17606) CRITICALITY OF THE LIQUID MIXTURES OF HIGHLY-ENRICHED UF_6 AND HF. Caizergues, Robert; Deilgat, Edouard; Lecorche, Pierre; Maubert, Louis; Revol, Henri (Commissariat a l'Energie Atomique, Saclay (France). Centre d'Etudes Nucleaires). Apr. 1968. 88p. (In French). (R-68.1). Dep.
Critical mass is determined for a UF_6-HF mixture as a function of ^{235}U concentration; the liquid-vapor equilibrium is established for the system. Schematics of the UF_6-HF circuit are shown, experimental apparatus is described. Density of the binary system is tabulated as a function of temperature; critical uranium concentration is shown as a function of temperature. Variation of the effective multiplication coefficient is shown as a function of sphere diameter. Effects of the wall of the sphere and temperature effects on the reactivity of the mixture are determined. 23 references. (M.L.S.)

47022 MULTIREGION FAST REACTOR EXPERIMENTS. Carpenter, S. G.; Mountford, L. A.; Springer, T. H.; Strominger, D.; Tuttle, R. J. pp 91-109 of I Reattori Veloci. Rome, Comitato Nazionale Energia Nucleare, 1967.

From 8th Nuclear Congress, Rome, June 1963. See CONF-193-(Vol.2).

A description of an Al$-^{235}$U-fueled critical assembly is presented. The critical assembly uses reduced-density Al for Na and graphite and Be for moderators. Calculations for the kinetic properties of the nine critical assemblies are presented. (auth)

50760 DETERMINATION OF EFFECTS OF CROSS SECTION ERRORS ON FAST REACTOR CALCULATIONS. Celentano, Romano; Gandini, Augusto (CNEN, Rome). pp 71-85 of Fisica del Reattore. Rome, Consiglio Nazionale delle Ricerche, 1966. (In Italian).

From Conference on Physics of Reactors, Milan. See CONF-469.

Fuel volume fractions for six reference fast reactor cores are tabulated; core volumes range from 400–2500 l. Sensitivity of reactivity to change in cross section is evaluated; results are tabulated for each core volume. Effects of cross section variation on initial conversion factors is discussed. (M.L.S.)

35637 INVESTIGATION OF THE CRITICALITY OF LOW-ENRICHMENT URANIUM CYLINDERS. Chezem, C. G. (Los Alamos Scientific Lab., N. Mex.). Nucl. Sci. Eng., 33: 139(1968).

An attempt to achieve a near-critical assembly with a minimum average ^{235}U enrichment of an unreflected, uranium metal, 21-in.-diam cylinder has been completed. Data were required for the design of a low-enrichment, uranium metal, reflected system. Utilizing only the materials on hand, 21-in.-diameter plates of 93.3% enriched uranium and normal uranium, four low-enrichment cylinders were investigated. The thickness of the normal uranium plates dictated the exact enrichments attainable by interleaving the plates in a cyclic manner along the axis. The critical parameters were obtained by extrapolation of inverse multiplication curves, which extend to 93 to 95% of the critical height. Corrections for the reflecting properties of the vertical support structure and the building itself were applied. The significant results are tabulated. A least-squares analysis (quadratic) of the data, inverse critical height squared vs percent enrichment, yields results which extrapolate to an infinite-height enrichment of 10.5 ± 0.2%. (auth)

3896 (RFP-1033) CRITICAL MASSES OF STEEL-MODERATED, ENRICHED URANIUM METAL ASSEMBLIES WITH COMPOSITE STEEL-OIL REFLECTORS. Coonfield, Donald C.; Tuck, Grover; Clark, Harold E.; Ernst, Bruce B. (Dow Chemical Co., Golden, Colo. Rocky Flats Div.). Nov. 7, 1967. Contract AT(29-1)-1106. 13p. Dep. CFSTI.

Critical masses have been determined, experimentally and calculated, for enriched U metal spherical assemblies, moderated internally with a sphere of mild steel of radius 8.01 centimeters. The assemblies were reflected with various thicknesses of mild steel followed by an effectively infinite amount of oil. An irregularity was noted in the graph of the experimental and calculated critical masses as a function of reflector steel thickness. (auth)

33035 A "BENCHMARK" SERIES OF PLUTONIUM-FUELED FAST CRITICAL ASSEMBLIES. Davey, W. G. (Argonne National Lab., Idaho Falls, Idaho); Broomfield, A. M.; Amundson, P. I.; (and others). Trans. Amer. Nucl. Soc., 11: 239-40(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

33123 KRITICHESKIYE PARAMETRY SISTEM S DELYASHCHIMISYA VESHCHESTVAMI I YADERNAYA BEZOPASNOST'; SPRAVOCHNIK. (Critical Parameters of Systems With Fissionable Materials and Nuclear Safety; a Handbook). Dubovskii, B. G.; Kamaev, A. V.; Kuznetsov, F. M.; (and others). Moscow, Atomizdat, 1966. 225p.

Presented is a handbook intended for specialists concerned with the problems of assuring nuclear safety, for persons calculating, designing, operating, and studying the physics of nuclear reactors of various types, and for students in associated departments. Methods of creating and maintaining conditions which will exclude the possibility of an accidental chain reaction during the processing, storage, and transportation of fissionable materials are discussed. The book is based mainly on the results of studies published before 1965. In addition to information on critical parameters of systems with fissionable materials, the fundamental concepts of criticality, principles for assuring nuclear safety, a review of cases of the occurrence of uncontrolled chain reactions, and the basic standards for nuclear safety are included. (ATD)

32975 (LA-3883) CRITICAL DIMENSIONS OF HOMOGENEOUS SPHERES CONTAINING ^{235}U, ^{238}U, AND CARBON FOR VARIOUS C/^{235}U RATIOS AND ^{235}U ENRICHMENTS. Engle, L. B.; Stratton, W. R. (Los Alamos Scientific Lab., N. Mex.). Dec. 15, 1967. Contract W-7405-eng-36. 13p. Dep. CFSTI.

The critical dimensions of homogeneous spheres containing ^{235}U, ^{238}U, and carbon at various C/^{235}U moderating ratios and ^{235}U enrichments are presented. Some values of k_∞ for these mixtures are included. (auth)

2324 (RFP-1025) CRITICAL MASSES OF SPHERICAL AND HEMISPHERICAL STEEL-MODERATED, OIL-REFLECTED ENRICHED URANIUM ASSEMBLIES. Ernst, Bruce B.; Tuck, Grover (Dow Chemical Co., Golden, Colo. Rocky Flats Div.). Nov. 6, 1967. Contract AT(29-1)-1106. 8p. Dep. CFSTI.

Critical masses were experimentally determined for steel-moderated, oil-reflected, spherical and hemispherical enriched uranium assemblies having inside radii from 0.0 to 12.0 cm. (auth)

18573 EXPERIMENTS ON THE ZR-3 CRITICAL ASSEMBLY IN CONNECTION WITH THE DEVELOPMENT OF THE WWR-SM CORE. Frankl, Laszlo; Gacsi, Lajos; Szabo, Ferenc; Szaklajda, Laszlo; Varkonyi, Lajos. KFKI (Kozp. Fiz. Kut. Intez.) Kozlem., 16: 3-33(1968). (In Hungarian).

The ZR-3 critical system was built as a part of the international cooperation for the reconstruction of the WWR-S reactors. The optimum configuration of the WWR-SM reactor and a possible load for the second operational cycle were evaluated on this zero power critical assembly. The results of measurements carried out on these two core configurations are given. (auth)

33059 MEASUREMENT OF THE CRITICAL MASS OF A WATER-REFLECTED PLUTONIUM SPHERE. Geer, W. U.; Smith, David R. (Los Alamos Scientific Lab., N. Mex.). Trans. Amer. Nucl. Soc., 11: 378(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

1968

53417 (LA-DC-9370) MEASUREMENT OF THE CRITICAL MASS OF A WATER-REFLECTED PLUTONIUM SPHERE. Geer, W. U.; Smith, David R. (Los Alamos Scientific Lab., N. Mex.). [nd]. Contract W-7405-eng-36. 3p. (CONF-680601-39). Dep. CFSTI.

From 14th Annual Meeting of the American Nuclear Society, Toronto, Ontario.

Criticality measurements for a water-reflected Pu sphere are presented. The high purity plutonium was fabricated and machined into a mass of 5546 g. The sphere was coated with Cu and enclosed in a Plexiglass shell. Monitoring of the neutron counting rate due to the Pu provided frequent checks of the multiplication of the sphere as water was slowly added to the sphere-containing 4 × 4 ft tank. One-dimensional DTF calculations are presented. The sphere was remachined to remove 200 g and the water-adding procedure repeated. Results are discussed. (D.C.C.)

18547 (DOCKET-50231-1) SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR. Facility Description and Safety Analysis Report, Amendment 6, Suppl. 1. (General Electric Co., Sunnyvale, Calif. Advanced Products Operation). Dec. 7, 1967. 57p. Dep. CFSTI.

Supplementary information on analysis of SEFOR mockup critical experiments in ZPR-3 is given. Information includes: ratio of prompt neutron lifetime to effective delayed neutron fractions, reaction ratios, reaction-rate traverses, ^{239}Pu traverses, and an evaluation of reflector leakage probabilities. 24 references. (M.L.S.)

30303 (GEAP-5271) IMPLICATIONS OF RECENT FAST CRITICAL EXPERIMENTS ON BASIC FAST REACTOR DESIGN DATA AND CALCULATIONAL METHODS. Greebler, P.; Gyorey, G. L.; Hutchings, B. A.; Segal, B. M. (General Electric Co., Sunnyvale, Calif. Advanced Products Operation). Oct. 1967. Contract AT(04-3)-189. 37p. Dep. CFSTI.

Recent experimental information from dilute, Pu-fueled critical assemblies is used to test and to provide guidance for improving fast reactor design data and calculational methods. The experimental data are taken from ZPR-III Assembly 47 (the SEFOR core mockup) and ZPR-III Assembly 48. The important nuclear reactor parameters are calculated with a number of variations in nuclear data and calculational techniques. An analysis of the experimental and calculated results shows that by careful adjustment of the important cross sections, well within experimental uncertainties, and by improved accuracy in the calculational methods, a much closer agreement can be achieved between calculations and experiments than that heretofore reported. A complete evaluation is made of the cross-section and resonance parameter data for Pu-239, the most important isotope in this case. The new data yield good agreement with the critical masses of ZPR-III Assemblies 47 and 48. They result in a calculated ^{239}Pu Doppler effect that is essentially zero, in agreement with the measured values. A two-dimensional calculation of the neutron lifetime, using group constants that adequately account for the spatial variation of the neutron spectrum, yields a significant improvement over that based on a one-dimensional model. (50 references are included). (auth)

33058 UNREFLECTED PLEXIGLAS-GRAPHITE-URANIUM CRITICAL MEASUREMENTS. Hoogterp, J. Carlton (Los Alamos Scientific Lab., N. Mex.). Trans. Amer. Nucl. Soc., 11: 389-90(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

25089 (LA-DC-8521) MINIMUM CRITICAL MASS. Jarvis, George A.; Mills, Carroll B. (Los Alamos Scientific Lab., N. Mex.). [nd]. Contract W-7405-eng-36. 10p. Dep. CFSTI.

A cubical core of enriched (93.15%) ^{235}U foil in a cubical Be reflector was used to establish minimum critical mass measurements. Three core sizes, of bases approximately 8 in., 6.5 in., and 6 in. square, were made critical by adjusting core height. Results of the experimental study are presented. (D.C.C.)

33065 EFFECT OF STEEL-WATER REFLECTORS ON THE CRITICALITY OF LOW-ENRICHED URANYL FLUORIDE SOLUTION. Johnson, E. B.; Newlon, C. E. (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 11: 383-4(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

35599 (K-D-2006) EFFECT OF STEEL-WATER REFLECTORS ON THE CRITICALITY OF LOW-ENRICHED URANYL FLUORIDE SOLUTION. Johnson, E. B. (Oak Ridge National Lab., Tenn.); Newlon, C. E. (Oak Ridge Gaseous Diffusion Plant, Tenn.). [1967]. Contract W-7405-eng-26. 8p. (CONF-680601-16). Dep. CFSTI.

From 14th Annual Meeting of the American Nuclear Society, Toronto, Ontario.

The effect of composite reflectors of steel and water on the reactivity of single cylinders of aqueous solution of low-enriched UO_2F_2 is described. The results are applicable to evaluation of criticality safety of shipping containers and for verification of calculational models. (D.C.C.)

35605 (ORNL-4263) CRITICAL EXPERIMENTS FOR THE REPETITIVELY PULSED REACTOR SORA. Kistner, G.; Mihalczo, J. T. (Oak Ridge National Lab., Tenn.). June 1968. Contract W-7405-eng-26. 105p. Dep. CFSTI.

A series of static critical experiments has been performed on an accurate mock-up of the SORA Reactor. SORA is a NaK-cooled repetitively pulsed fast reactor which will be used as a high intensity neutron source for time-of-flight experiments. The reactivity of this reactor is varied by a movable reflector. Those parameters which are related to the kinetics of the reactor have been investigated thoroughly in the critical experiments. They have been measured for beryllium and for iron reflectors of several sizes and for various core and fixed reflector configurations. The total reactivity of the movable reflectors varied from 3.7 dollars for a 11-cm-wide iron reflector to 12 dollars for a 26.2-cm-wide beryllium reflector. The reactivity of the movable reflector as a function of its position has been shown to have a parabolic dependence on position characterized by the parameter α_x which varied from 4 to 9.9 cents/cm^2. The prompt neutron time decay is described by a fast decay constant which varied between 0.30 and 0.55 μsec^{-1} and a slow decay constant which varied between 0.05 and 0.10 μsec^{-1}. The critical masses for the various experiments was between 50.3 to 57.3 kilograms of uranium enriched to 93.2 wt % ^{235}U. Using space independent neutron kinetics with one delayed neutron group, it has been shown that with a 24-cm-high, 7-cm-thick, 21-cm-wide beryllium reflector the assembly will produce pulses approximately 50 μsec wide at half maximum power with a peak-to-minimum power ratio of approximately 4000. (auth)

35657 MASURCA 1-A AND 1-B-PRELIMINARY RESULTS. Kremser, J.; Barberger, M.; Bruna, J. G.; Brunet, J. P.; Mougniot, J. C.; Schmitt, A. P.; Verriere, P. (CEA, Cadarache, France). pp 3-33 of Fast Reactor Physics. Vol. II. Vienna, International Atomic Energy Agency, 1968. (In French).

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.2); CONF-671043-(Vol.2).

The MASURCA fast-neutron critical model at Cadarache went critical in December 1966 with a plutonium configuration, using graphite as diluent so as to obtain a neutron spectrum lying within the spectral range of high-power fast reactors. The construction of the first core, composed of the MASURCA 1-A lattice using a U - Pu - Fe alloy with 25% Pu, and experimental techniques are described. Spectral index measurement data and reactivity coefficients for various materials are quoted. The material buckling value of the assembly was determined experimentally using spatial distributions for various reaction rates. Some measurement data are compared with the corresponding theoretical values calculated from transport theory or by a Monte Carlo method. The causes of error and improvements which should be made in experimental methods and in interpretation of the measurements are discussed. (auth)

44874 (NASA-CR-72329) CAVITY REACTOR GAS-CORE CRITICAL EXPERIMENT. Kunze, J. F.; Masson, L. S.; Pincock, G. D.; Wood, R. E.; Hyland, R. E. (General Electric Co., Idaho Falls, Idaho. Nuclear Materials and Propulsion Operation). Nov. 6, 1967. Contract C-67747-A. 21p. (CONF-671102-42).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill.

A gaseous UF_6 core configuration was made critical in a 4 foot diameter, 43 inch long, tank and critical loading measurements were made. The results from the real gas configuration are compared with those from a mockup U-foil configuration of the same dimensions. It is concluded that the mockup provides an adequate representation of the gas core and can be used to simplify studies of variations in materials and geometry for the cavity reactor concept. (H.D.R.)

18570 EXPERIMENTAL AND THEORETICAL INVESTIGATIONS ON THE PHYSICS OF FAST REACTORS. Leipunskii, A. I.; Abagyan, L. P.; Bazazyants, N. O.; (and others). pp 445-91 of Fast Breeder Reactors. Evans, P. V. (ed.). Oxford, Pergamon Press, 1967.

From British Nuclear Energy Society Conference on Fast Breeder Reactors, London. See CONF-660502.

A reactor of the BN-350 type was studied by the use of critical assemblies with two enrichment zones. Critical parameters were calculated and cross section ratios measured in the center of the assemblies. Perturbations caused by different materials, including ^{235}U, Ta, Re, Fe, Nb, Mo, and W, were investigated. Control rod effectiveness was studied and heterogeneity effects analyzed. Prompt neutron lifetime was measured by means of two enriched BF_3 proportional counters. Neutron propagation in UO_2 and the Doppler effect in ^{235}U were studied. Space-energy neutron distribution in the thick oxide blanket of the BR-1 reactor was measured. Some problems in calculation theory and methods are described, including theories of neutron propagation and transfer. (UK)

33063 CRITICALITY OF PLUTONIUM NITRATE SOLUTIONS IN SLAB GEOMETRY. Lloyd, R. C.; Clayton, E. D.; Hansen, L. E. (Brookhaven National Lab., Upton, N. Y.). Trans. Amer. Nucl. Soc., 11: 381-2(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

16289 (ORNL-TM-2082) CRITICAL EXPERIMENTS WITH ENRICHED URANIUM METAL-POLYETHYLENE, -PLEXIGLAS, AND -TEFLON MIXTURES. Magnuson, D. W. (Oak Ridge National Lab., Tenn.). Feb. 1968. Contract W-7405-eng-26. 38p. Dep. CFSTI.

Reflected and unreflected critical experiments were performed at H: ^{235}U atomic ratios from 0 to 5 in rectangular geometry with layers of enriched uranium and polyethylene. Base dimensions of the assemblies were 5 × 10 and 10 × 10 in. In some unreflected experiments, the metal was interleaved with Plexiglas and with Teflon at only the latter base dimensions. Heterogeneity effects were found to be small from experiments assembled from the same or approximately the same materials but with different layer thicknesses. Values of k_{eff} calculated with the KENO Monte Carlo code, are in excellent agreement with the experimental values. By equating geometrical bucklings for rectangular and spherical geometry, these data were converted to critical masses for spheres. Comparisons of the experimental values were made to the critical masses calculated by the ANISN transport code for homogeneous spheres. (auth)

33064 CRITICAL EXPERIMENTS AT H:^{235}U RATIOS FROM 0 TO 5. Magnuson, D. W. (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 11: 383(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

6108 SUPERPROMPT CRITICAL BEHAVIOR OF A URANIUM-MOLYBDENUM ASSEMBLY. Mihalczo, J. T. (Oak Ridge National Lab., Tenn.); Lynn, J. J.; Watson, J. E.; Dickinson, R. W. Trans. Amer. Nucl. Soc., 10: 611-12(Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

14267 EXPERIMENTAL REACTOR WITH GASEOUS UF_6. NO. 1. THE CALCULATION OF THE CRITICAL MASS. Ogushi, Terumune (Chuo Univ., Japan); Yumoto, Ryozo. Chuo Daigaku Rikogakubu Kiyo, 8: 125-33(Dec. 1965). (In Japanese).

The critical mass of experimental reactor with gaseous UF_6 is determined. The reactor was a partly heterogeneous type, with a Be metal moderator and a graphite reflector. The core of the reactor was of cylindrical shape, 116 cm in diameter and 109 cm height. The Be moderator was in the form of tubes of 4.0 cm square in cross-section. Gaseous UF_6 was filled in 148 channels arranged in a square lattice with 8.0 cm pitch. Aluminum tubes of 4.0 cm square in cross-section and 0.1 cm thick were used to make those channels. The side reflector was 50.0 cm thick, and the upper and the lower reflectors were both 60.0 cm thick. From the results of the calculation by one-group theory of four factor formula, it was shown that the reactor can reach critical mass with that of 2.3 kg 90% enriched UF_6 at 0.9 atm. and 80°C. (auth) (NSA of Japan)

3881 (NASA-CR-72234(Vol.1)) CAVITY REACTOR CRITICAL EXPERIMENT. VOLUME 1. Final Report. Pincock, G. D.; Kunze, J. F. (General Electric Co., Idaho Falls, Idaho. Nuclear Materials and Propulsion Operation). Sept. 6, 1967. Contract C-67747-A. 380p. Dep. CFSTI.

A series of experiments were conducted at the Idaho Test Station on a large "cavity" reactor consisting of a cavity 183 cm in diameter of 122 cm long surrounded by 91 cm of heavy water. The cavity was fueled with uranium-235. Measurements were made on various configurations, including such variations as fuel diameter and shape, beryllium baffles in the reflector and insertion of various structural and operating materials characteristic of a nuclear rocket reactor. 11 references. (auth)

6123 A COMPARISON OF CALCULATION AND EXPERIMENT FOR ZPR-3 ASSEMBLY 48. Pitterle, T. A. (Atomic Power Development Associates, Inc., Detroit); Yamamoto, M. Trans. Amer. Nucl. Soc., 10: 530-1(Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5–9, 1967. See CONF-671102.

25075 (APDA-201) CROSS SECTION EVALUATION AND CRITICAL EXPERIMENT ANALYSIS FOR FAST REACTORS. Pitterle, T. A.; Yamamoto, M. (Atomic Power Development Associates, Inc., Detroit, Mich.). June 1967. 101p. Dep.

For Edison Electric Inst., New York.

A neutron cross section evaluation and critical experiment analysis conducted in support of PuO_2 core studies for the Enrico Fermi Reactor are presented. Cross sections of primary importance for fast reactor analysis were evaluated for use with the Argonne National Laboratory (ANL) MC^2 code which generates multigroup data from basic cross section data. The resulting multigroup cross sections were used for diffusion theory calculations of ZPR-III Assemblies 47 and 48. A comparison of calculation with experiment is presented for these critical experiments. (auth)

35344 RE-EVALUATION OF ^{235}U, ^{238}U, AND ^{239}Pu CROSS-SECTIONS BASED ON MICROSCOPIC AND INTEGRAL DATA. Rakavy, G.; Reiss, Y.; Samoucha, D.; Yeivin, Y. (Hebrew Univ., Jerusalem). pp 255-66 of Fast Reactor Physics. Vol. I. Vienna, International Atomic Energy Agency, 1968.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.1); CONF-671043-(Vol.1).

A generalized least-squares method to improve microscopic cross-section evaluations by means of integral data was applied to re-evaluate the cross sections of ^{235}U, ^{238}U, and ^{239}Pu, using critical-mass data of 24 simple metallic systems composed of these isotopes. It was found that, after some minor modifications of the original cross section set, most of the experimental integral data could be reproduced. The cross-section modifications, as a rule, were of the order of a few per cent and well within the uncertainties in the cross-sections. The exception to the rule was the ^{239}Pu fission cross section in the energy range up to about 150 keV, which had to be decreased by 15 to 20%. This result independently confirms the recent measurements of White et al. (auth)

10069 THEORETICAL ANALYSES OF HOMOGENEOUS PLUTONIUM CRITICAL EXPERIMENTS. Richey, C. R. (Battelle Memorial Inst., Richland, Wash.). Nucl. Sci. Eng., 31: 32-9(1968).

A computational analysis was made for the large number of available critical experiments with hydrogenous mixtures. The calculations were made using both multigroup S_4 and diffusion theory with 18 energy groups obtained with the GAMTEC-II code. Resonance capture by the isotope ^{240}Pu was treated in the NR and NRIA approximations. The results are given as a parametric survey for Pu densities ranging from 0.015 to 1.0 g/cm^3. The calculated minimum critical mass of ^{239}Pu is 547 g for water-reflected aqueous $Pu(NO_3)_4$ solutions and 531 g for similar mixtures of ^{239}Pu and water. 14 references. (auth)

37407 (BNWL-801) REACTOR PHYSICS DATA FOR THE UTILIZATION OF PLUTONIUM IN THERMAL POWER REACTORS. Schmid, L. C.; Leonard, B. R. Jr.; Liikala, R. C.; Smith, R. I. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). May 1968. Contract AT(45-1)-1830. 143p. Dep. CFSTI.

Experimental reactor physics data have been and are currently being obtained in the United States to study the utilization of plutonium in present-day thermal reactors. A reference for what data exist and where it can be found is presented. References to data for lattices moderated with H_2O, D_2O, and graphite are included. However, discussions are centered around the use of plutonium in H_2O reactors because these reactor types are of most interest in the United States. Problems connected with calculating H_2O-plutonium systems are illustrated using the data, and areas are mentioned in which needs for additional data still exist. Cross sections, criticality, reactivity coefficients, kinetics, and burnup data are referred to and conclusions are made about the use of the data in evaluating methods and cross sections for H_2O moderated reactors. 391 references. (auth)

6113 ANALYSIS OF SOME ^{235}U-GRAPHITE CRITICAL EXPERIMENTS. Sehgal, Bal Raj (Brookhaven National Lab., Upton, N. Y.). Trans. Amer. Nucl. Soc., 10: 535-6(Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5–9, 1967. See CONF-671102.

32997 EXPERIMENTAL COMPARISON OF FAST- AND THERMAL-NEUTRON BUCKLINGS IN A CRITICAL ASSEMBLY. Shaw, Robert A. (Clarkson Coll. of Tech., Potsdam, N. Y.); Clark, David D. Trans. Amer. Nucl. Soc., 11: 48-9(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

33080 ANALYSIS OF A SET OF HOMOGENEOUS U-H_2O SPHERES. Staub, Alan; Harris, D. R.; Goldsmith, M. (Westinghouse Electric Corp., West Mifflin, Pa.). Trans. Amer. Nucl. Soc., 11: 305-6(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

1968

20410 (GA-8466) NEUTRONIC MEASUREMENTS IN NON-CRITICAL MEDIA. Stevens, C. A. (Gulf General Atomic, Inc., San Diego, Calif.). [1967]. Contract AT(04-3)-167. 34p. (CONF-680307-19). Dep. CFSTI.

From 2nd Conference on Neutron Cross Sections and Technology, Washington, D. C.

Several types of subcritical experiments that are intended to provide integral checks of cross sections and computational methods in fast reactors and fast neutron shielding are discussed. These include studies of the time response to a pulsed neutron source, steady-state neutron spectrum measurements, and transmission measurements. The methods which are used to analyze these experiments are described, with the emphasis placed on what cross section information can be extracted from them. A description of how cross section averaging procedures are actually performed in some of the more sophisticated codes designed for this purpose is included. (auth)

6117 CRITICALITY OF ^{233}U AQUEOUS NITRATE SOLUTION IN REFLECTED AND UNREFLECTED ARRAYS. Thomas, J. T. (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 10: 538-9(Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

1564 (ORNL-CDC-2) CALCULATED NEUTRON MULTIPLICATION FACTORS OF UNIFORM AQUEOUS SOLUTIONS OF ^{233}U AND ^{235}U. Webster, J. Wallace (Oak Ridge National Lab., Tenn.). Oct. 1967. Contract W-7405-eng-26. 39p. Dep. CFSTI.

Computations of the effective neutron multiplication factor of single units of aqueous solutions of $^{233}UO_2F_2$ and $^{235}UO_2F_2$ are reported for guidance in the specification of limits applicable to processes, such as storage and transport, for these fissile isotopes. Graphs are presented of k_{eff} as a function of such parameters as the mass of fissile material, the chemical concentration, the dimensions of spheres and infinitely long cylinders, and the thickness and areal density of infinite slabs. Transport theory (DTF) codes in the S_n approximation with Hansen-Roach cross sections were utilized and the results agree with relevant experiments to within 0.01 in k_{eff}. (auth)

6118 THE NUCLEAR SAFETY OF AQUEOUS SOLUTIONS OF ^{233}U AND ^{235}U. Webster, J. Wallace (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 10: 539(Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

II. CRITICAL EXPERIMENTS

Lattices

1967

12232 (BAW-3647-2) PHYSICS VERIFICATION PROGRAM. Quarterly Technical Report No. 2, July–September 1966. (Babcock and Wilcox Co., Lynchburg, Va.). Nov. 1966. Contracts AT(30-1)-3647; 41-2007 (RDE-1526). 71p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

The results of critical experiments on cores IV, V, and VI are reported. In all assemblies the basic lattice consisted of 0.475-inch-OD rods of 2.45%-enriched UO_2 arrayed on a square pitch of 0.644 inch and moderated by H_2O. The water was poisoned with H_3BO_3 (about 1.5 gm B/l) to obtain critical assemblies approximately 5 feet in diameter. The reactivity worth of Ag–In–Cd control pins arranged in configurations typical of power reactors was measured in these cores. The results of measurements of ρ_{28}, the modified conversion ratio, and the epithermal neutron spectrum are also reported. (auth)

35565 (BAW-393-10) THORIUM FUEL CYCLE FOR HEAVY WATER MODERATED ORGANIC COOLED REACTORS. Technical Progress Report No. 7, October 1966–March 1967. (Babcock and Wilcox Co., Lynchburg, Va. Atomic Energy Div.). May 1967. Contract AT(38-1)-393. 18p. Dep. CFSTI.

Work directly relating to the thorium – HWOCR design and development was closed out. Assistance was given to the Evaluated Nuclear Data File Task Force at Brookhaven. This included supplying nuclear data for ^{238}U, ^{232}Th, and lumped fission products. Evaluation work was done on various reactor concepts being considered. (M.C.G.)

27730 ANALYSIS OF GRAPHITE MODERATED URANIUM AND PLUTONIUM/URANIUM OXIDE FUEL CLUSTERS USING THE LATTICE CODE WIMS. Barclay, F. R. (Atomic Energy Establishment, Winfrith, Eng.). J. Brit. Nucl. Energy Soc., 6: 155-60(Apr. 1967).

The performance of the lattice code WIMS was studied by the analysis of graphite moderated exponential experiments fuelled with clusters of UO_2 or Pu/UO_2 rods at temperatures up to 390°C. Earlier work on single metal rod systems showed that the best agreement in reactivity between U and Pu/U fuel was obtained by using a ^{239}Pu η value of 2.098 (at 2200 m/s). The use of the IAEA recommended η of 2.114 in the work showed a significant dependence of reactivity on ^{239}Pu enrichment, which was largely removed by using an η value of 2.098. (UK)

15764 (CONF-660221-, pp 287-90) ANALYSIS OF UNIFORM LATTICE EXPERIMENTS WITH THORIA–URANIA FUEL IN HEAVY WATER AND LIGHT WATER AS MODERATORS. Bhatta, H. K. (Atomic Energy Establishment, Trombay (India)).

The METHUSELAH-I and CAROL codes for $ThO_2-^{235}UO_2$ or $ThO_2-^{233}UO_2$ lattices with heavy or light water as moderator were assessed. Experimental bucklings were used to calculate the K_{eff} for uniform critical lattice experiments performed at Argonne National Laboratory, Brookhaven National Laboratory, and the Babcock and Wilcox Company. (H.D.R.)

25722 ANALYSIS OF CRITICAL EXPERIMENTS WITH ORGANIC-MODERATED ASSEMBLIES. Bitelli, G.; Martinelli, R.; Orestano, F. V.; Santandrea, E. Nucl. Sci. Eng., 28: 270-6(May 1967).

The results of critical experiments, performed with organic-moderated plate-type assemblies containing U enriched to 90% in ^{235}U, in the zero-power reactor ROSPO, are reported. Several cores, differing in critical radius and in ratio of U-to-stainless-steel plate number, have been investigated. The comparison with the reactivities calculated by a standard two-group calculation procedure shows an overestimate of the k_{eff}'s with a systematic dependence on the core radius. A satisfactory agreement is found for large-size cores. It is shown that simple calculational improvements, such as a four-group evaluation of the nuclear constants, and a more detailed treatment of core-radial reflector interface zone, lead to a homogeneously good agreement over the whole range of core dimensions. (auth)

1967

30249 (ANL-7203) HIGH CONVERSION CRITICAL EXPERIMENTS. Boynton, A. R.; Baird, Q. L.; Plumlee, K. E.; Redman, W. C.; Robinson, W. R.; Stanford, G. S. (Argonne National Lab., Ill.). Jan. 1967. Contract W-31-109-eng-38. 81p. Dep. CFSTI.

A program of critical experiments has extended the range of measured nuclear parameters of light-water-moderated, slightly enriched, UO_2 lattices further into the undermoderated region. The H-to-^{238}U atom ratios in the cores assembled ranged from about 5 to 0.5. The initial conversion ratio, ^{238}U-to-^{235}U fission ratio, ^{238}U capture Cd ratio, thermal disadvantage factor, and Cd ratios of various materials reported were measured in the full range of lattices; the bucklings, critical masses, and reflector savings reported were measured in the looser assemblies. Information on temperature coefficient and control-element and various material worths is included. Although the experimental results are not compared with calculations, these results are correlated with results obtained at other laboratories with similar fuel at higher H-to-^{238}U ratios. (auth)

12246 MATERIAL BUCKLINGS FOR 1.002, 1.25, and 1.95 WT PERCENT ^{235}U ENRICHED URANIUM TUBES IN LIGHT WATER. Brown, C. L.; Lloyd, R. C. (Battelle Memorial Inst., Richland, Wash.). Nucl. Sci. Eng., 27: 10-15(Jan. 1967). (BNWL-SA-267).

Material bucklings and extrapolation distances were measured for several slightly enriched U-metal tube lattices and tube-in-tube assembly lattices in light water. The tubes measured were: 1.002 wt % ^{235}U enriched U (2.34-in. OD; 1.79-in. ID); 1.25 wt % ^{235}U enriched U (2.37-in. OD; 1.80-in. ID); and 1.95 wt % ^{235}U enriched U (2.28-in. OD; 1.41-in. ID). The tube-in-tube assemblies measured were: 1.002 wt % ^{235}U outer tubes (2.34-in. OD; 1.79-in. ID) containing 1.002 wt % ^{235}U inner tubes (1.18-in. OD; 0.49-in. ID); and 1.25 wt % ^{235}U outer tubes (2.37-in. OD; 1.80-in. ID) containing 0.95 wt % ^{235}U inner tubes (1.18-in. OD; 0.48-in. ID). Maximum bucklings for the tubes were found to be 25.00, 47.00, and 83.00 m^{-2}, respectively; and for the tube-in-tube assemblies, 23.50 and 38.50 m^{-2}, respectively. Based on the measurements, critical parameters for use in nuclear safety analyses were calculated. (auth)

38715 EXPERIMENTAL BUCKLING MEASUREMENTS WITH 2.1 WT PERCENT ENRICHED URANIUM TUBES IN LIGHT WATER. Brown, C. L. (Battelle-Pacific Northwest Lab., Richland, Wash.); Hansen, L. E.; Toffer, H. Trans. Amer. Nucl. Soc., 10: 191-3(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

38713 TEMPERATURE DEPENDENCE OF NUCLEAR REACTION RATES IN A Pu-H_2O LATTICE. Carver, J. G.; Porter, C. R. (General Electric Co., Pleasanton, Calif.). Trans. Amer. Nucl. Soc., 10: 190(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

14091 κ_∞ MEASUREMENT OF A NATURAL URANIUM GRAPHITE LATTICE IN RB-1 BY THE NULL REACTIVITY METHOD. Casali, F. (CNEN, Bologna); Ghilardotti, G. Energ. Nucl. (Milan), 13: 480-90(Sept. 1966).

Measurement of the infinite multiplication constant is described for a natural U graphite lattice, similar to the old Brookhaven lattice, carried out in the critical assembly RB-1, by the null reactivity method. With respect to past measurements of this type, the experimental procedure and the interpretation of the results were made more consistent and complete. The modifications introduced are discussed in detail. A preliminary assessment was also made of the effect of spectrum mismatching. (auth)

25712 (BAW-3647-03) PHYSICS VERIFICATION PROGRAM, Final Report. Clark, R. H.; Baldwin, M. N.; Pettus, W. G.; Pitts, T. G.; Flickinger, R. B. (Babcock and Wilcox Co., Lynchburg, Va. Research and Development Div.). Mar. 1967. Contract AT(30-1)-3647. 188p. Dep. CFSTI.

A program to determine the nuclear properties of water-moderated, slightly enriched, uranium dioxide lattices perturbed by clusters of Ag-In-Cd poison pins and dissolved boric acid is described. Core parameters and configurations typical of those in water-moderated power reactors were studied. The unperturbed basic lattice consisted of 0.475-inch-OD rods of 2.46%-enriched UO_2 on a square lattice pitch of 0.644 inch, and core radii varied from 19 to 65 cm. Boric acid, in concentrations from 0 to about 1500 ppM, was dissolved in the moderator to control excess reactivity. Power reactor fuel elements, incorporating clusters of Ag-In-Cd poison pins and typical structural material, were simulated in the central region of the largest core. Measurements were made of critical size, reactivity worth of poison pins, buckling, reflector savings, thermal disadvantage factor, epithermal absorption in ^{238}U, conversion ratio, neutron spectrum, radial and axial power distribution, and temperature coefficient. Experimental results of the program, many of which have been reported in quarterly technical reports, are presented. In some cases the data have been re-analyzed, and the reported results may differ slightly from those reported earlier. (J.R.D.)

1967

27731 EVALUATION OF SOME UNCERTAINTIES IN THE COMPARISON BETWEEN THEORY AND EXPERIMENT FOR REGULAR LIGHT WATER LATTICES. Fayers, F. J.; Kemshell, P. B.; Terry, M. J. (Atomic Energy Establishment, Winfrith, Eng.). J. Brit. Nucl. Energy Soc., 6: 161-81(Apr. 1967).

The principal factors influencing the accuracy of comparisons between theory and experiment for regular water-moderated lattices are examined. By the use of more elaborate theoretical methods, the accuracy of the physics methods used in the WIMS lattice code is established with regard to leakage, fast fission events in ^{238}U, resonance captures in ^{238}U and thermal disadvantage factors. The difficulty of making corrections for the non-asymptotic nature of the flux to bucklings inferred from measurements in small exponential cores is examined and the validity of a one-dimensional analysis of this effect is questioned. Subject to this limitation it is shown that the computational methods are sufficiently accurate to allow deductions to be made concerning the gross characteristics of fundamental nuclear data through comparison of the predicted reactivities and reaction rates with experiment. Results for a series of UO_2 lattices using both critical and exponential techniques are given and further comparisons are made for a selection of experiments using both UO_2 and U metal fuel. The results of reducing the leakage by poisoning with B in these experiments are also described. The results obtained support the need to introduce a modification to the resonance integrals for ^{238}U as computed from fundamental data, and information is provided on a preferred value for the ratio of epithermal capture to fission in ^{235}U. A pitch-dependent error in the fast fission factor and a discrepancy in temperature coefficient are identified, but efforts to isolate the cause of these errors were not successful. (UK)

15768 (N-66-33021) 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. 25p. (NASA-TN-D-3555). CFSTI $3.00 cy, $0.65 mn.

The NASA Zero Power Reactor II (ZPR-II) has been used to determine experimentally several critical cylindrical configurations of aqueous fuel solutions that contain heterogeneous arrays of voids. These voids are cylindrical, are symmetrically arranged parallel to the axis of the reactor, and extend the height of the core. The study covered a wide range of highly enriched aqueous UO_2F_2 fuel concentrations. The specific reactor void configurations consisted of symmetrical arrays of 1, 7, 19, 31 and 37 tubes approximately 7.6 cm in diameter arranged in hexagonal geometry with pitches of 9.652 or 10.922 cm. In addition to the critical mass and geometry, data are presented on the thermal neutron flux distributions in the central radial plane, and on the variation of void reactivity importance with radial position. These data explain qualitatively some of the reactivity effects associated with the different void spacings. (STAR)

23715 (EUR-3126.i) MISURA DELLA COSTANTE DI MOLTIPLICAZIONE INFINITA DI RETICOLI A URANIO NATURALE E GRAFITE NELL'INSIEME CRITICO RB-1 CON IL METODO DELLA REATTIVITA' NULLA. (Measurement of the Infinite Multiplication Constant of Natural Uranium–Graphite Lattices in the RB-1 Critical Assembly by Means of the Zero Reactivity Method). Ghilardotti, G. (Societa Nazionale Metanodotti, Milan (Italy). Laboratori Reuniti Studi e Ricerche). Oct. 1966. Contract 038-64-3 TEGI. 212p. (In Italian). Dep. mn.

The development of the zero reactivity technique (PCTR) applied to natural U/graphite lattices and its comparison with the substitution method are described. The experiments were carried out in the critical assembly RB-1 at the Montecuccolino laboratory, Bologna. Two series of measurements were conducted on two lattices differing only with regard to the fuel element, one consisting of a solid 29.2 mm dia rod in a can, the other being a tube 30 mm to 50 mm OD; channel dia 70 mm square lattice pitch of 224 mm. These lattices were chosen from those tested in the Marius critical assembly by the substitution method. Results show that a consistent and complete experimental procedure has been devised for measuring the K_{00} of natural U/graphite lattices by means of zero reactivity method. The same applies to the procedure for analysis of the experimental data. The error in (K_{00} — 1) inherent in measurement can in our opinion be reduced to 2%. This limit was reached in the last experiment on lattices consisting of tubular elements. Agreement proved to be good with the results obtained by the CEA in the critical assembly Marius. (tr-auth)

35666 (AECL-2691) CANDU-BLW EXPERIMENTS IN ZED-2. PART III. BUCKLING AND LOSS OF COOLANT EXPERIMENTS. Green, R. E.; Kay, R. E.; Colpitts, C. W. (Atomic Energy of Canada Ltd., Chalk River (Ontario)). May 1967. 44p. Dep. CFSTI. CAN $1.00.

Experiments have been performed in simulated CANDU-BLW lattices in ZED-2 (square arrays of 28-rod UO_2 clusters at a spacing of 27.94 cm) to determine (a) the material buckling of the lattice with H_2O or air as "coolants," and the flux perturbation and reactivity effects of removing the H_2O coolant from 50% of the fuel assemblies in three geometric arrangements. The buckling for the H_2O-cooled lattice was $1.166 \pm 0.018\ m^{-2}$, and for the air-cooled lattice $3.949 \pm 0.039\ m^{-2}$. The loss of coolant experiments indicated a significantly smaller increase in reactivity when alternate fuel assemblies or alternate rows of fuel assemblies were voided than when one-half of the lattice about a dia was voided. (auth)

1967

38655 (AECL-2707) INTEGRAL NEUTRON SPECTRUM MEASUREMENTS IN HEAVY WATER LATTICES AND A COMPARISON WITH THEORY. Green, R. E.; Kay, R. E.; Halsall, M. J. (Atomic Energy of Canada Ltd., Chalk River (Ontario)). June 1967. 18p. (CONF-670707-16). Dep. CFSTI. CAN $1.00.

From IAEA Symposium on Neutron Thermalization and Reactor Spectra, Ann Arbor, Mich.

Experiments have been performed in the ZEEP and ZED-2 critical facilities at Chalk River to obtain information about the neutron spectrum in lattices of natural U fuel in heavy water moderator. Studies have been made in lattices containing clusters of U, UO_2, and UC having D_2O, air and organic liquids as coolants, all at room temperature, and some experiments with UO_2 fuel have investigated the effect of changing the coolant temperature. Integral neutron spectrum measurements that were made in both the fuel and moderator regions of lattice cells by activating foils containing In, Mn, and Lu are described. The relative $^{115}In/^{56}Mn$ and $^{176}Lu/^{56}Mn$ activity ratios have been compared with calculated ratios obtained from HAMMER, a multigroup, multiregion integral transport theory code, SOLO, a multigroup, multiregion collision probability code and MULTIGRO, a multigroup, two-region diffusion theory code. The activity ratios have also been interpreted in terms of the Westcott epithermal index, r, and effective neutron temperature, T_n. Experiment and theory are compared for selected lattices to illustrate the effect upon the neutron spectrum of fuel material, cluster geometry, coolant composition, and coolant temperature. (auth)

14097 SPATIAL VARIATION OF THE FAST NEUTRON FLUX IN CELLS OF SLIGHTLY ^{235}U-ENRICHED-URANIUM WATER-MODERATED LATTICES. Hardy, J. Jr. (Westinghouse Electric Corp., West Mifflin, Pa.); Klein, D.; Dannels, R. Nucl. Sci. Eng., 26: 462-71(Dec. 1966).

The intra-cell structure of the fast neutron flux was measured in several TRX lattices with ^{238}U (n,fission) and Al (n,α) detectors. The lattices were light-water-moderated, with cylindrical, 0.387-in.-dia fuel rods of slightly enriched U. The fuel rods were arranged in hexagonal arrays, with H_2O: U volume ratios of 1.0, 2.35, 4.02, and 8.1. Measured activation shapes and integral fast advantage factors were compared with calculated results obtained with the MOCA2A Monte Carlo program. Agreement was very good. A one-group Monte Carlo calculation and a one-group collision-probability model were found to perform well in comparison with MOCA2A. (auth)

44944 (NP-16953) NEUTRON DIFFUSION AND MULTIPLICATION IN REACTOR LATTICES. Haroon, Muhammad Rafiq (Imperial Coll. of Science and Technology, London (England). Nuclear Power Group). Sept. 1966. 378p. Dep. Thesis.

Neutron flux distribution and buckling measurements were made as a function of fuel loading in a subcritical assembly. The measurements were made in a square configuration. Control rod effectiveness measurements were also made. Results were correlated on the basis of diffusion theory. Values for k_∞ and B_m^2 are shown as a function of number of fuel elements; reflector savings are plotted as a function of reflector thickness. Analysis of control rod effects is based on super-cell calculations and the heterogeneous theory. (M.L.S.)

7895 (ORNL-TM-1566) CRITICALITY OF LATTICES OF HEAT TRANSFER REACTOR EXPERIMENT FUEL ELEMENTS. Johnson, E. B.; Reedy, R. K. Jr. (Oak Ridge National Lab., Tenn.). July 20, 1966. Contract W-7405-eng-26. 14p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

A series of experiments were completed to determine the critical parameters of lattices of HTRE fuel elements, primarily in geometries and environments of interest for transport, storage, and chemical dissolution. Arrays of these elements were made critical with water and with dilute aqueous $UO_2(NO_3)_2$ solution of two concentrations (to simulate dissolver environments) as moderator and reflector; one solution concentration was 3.97 g of ^{235}U/l and the other was 8.02 g/l. In some of the slab lattices in water, sheets of cadmium were placed between rows to serve as a neutron absorber as they might in a shipping container. (auth)

30259 (ORNL-TM-1808) CRITICAL LATTICES OF HIGH FLUX ISOTOPE REACTOR FUEL ELEMENTS. Johnson, E. B. (Oak Ridge National Lab., Tenn.). Mar. 20 1967. Contract W-7405-eng-26. 10p. Dep. CFSTI.

Lattices of High Flux Isotope Reactor (HFIR) fuel elements were assembled in order to determine the critical spacing between elements when moderated and reflected by water. It was found that seven elements spaced 6.37 in. in a triangular pattern were critical when submerged. Seven outer annuli in the same pattern were critical when separated 1.50 in., and seven inner annuli were subcritical even when in contact. (auth)

38714 CRITICAL LATTICES OF U(4.89) METAL RODS IN WATER. Johnson, E. C. (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 10: 190-1 (June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

23709 (AECL-2651) LATTICE MEASUREMENTS WITH 7-ROD CLUSTERS OF NATURAL URANIUM CARBIDE IN HEAVY WATER MODERATOR. PART II. NEUTRON SPECTRUM PARAMETERS IN A LATTICE CELL. Kay, R. E.; Green, R. E. (Atomic Energy of Canada Ltd., Chalk River (Ontario)). Dec. 1966. 46p. Dep. mn. CFSTI $3.00 cy, $0.65 mn. AECL $1.00.

Experiments have been performed in the ZED-2 critical facility to determine In-Mn and Lu-Mn activity ratios at various positions in the central cell of lattices of seven rod clusters of UC in heavy water moderator. The measurements were made in square lattices at several pitches using different coolant materials. The activity ratios have been interpreted in terms of the Westcott epithermal index r, and effective neutron temperature T_n, and compared with values predicted by the Chalk River lattice recipe code LATREP. (auth)

5833 (HW-51937) PLUTONIUM METAL DISSOLUTION, PART II. Ketzlach, N. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Aug. 14, 1957. Contract AT(45-1)-1350. 8p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

A study was made of the nuclear safety involved in the dissolution of plutonium metal. The study indicated that the minimum critical mass of a plutonium metal-plutonium solution system may be no less than that for the plutonium metal-water system or homogeneous solution system, whichever is smaller. The enriched uranium metal-uranium solution system was also studied. Experiments are suggested to verify the results. (M.C.G.)

35702 (BNWL-CC-1205) PRELIMINARY CALCULATIONS FOR H_2O-MODERATED LATTICES OF UO_2–4 WT. PERCENT PuO_2 FUEL RODS. Kobayashi, S.; Uotinen, V. O. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). May 17, 1967. Contract AT(45-1)-1830. 32p. Dep. CFSTI.

Physics calculations, preliminary to experiments, for determining cell-averaged lattice constants are presented. Calculations were made for 7.65 wt % and 23.5 wt % ^{240}Pu lattices for three geometries. All parameters are tabulated. Cell-averaged macroscopic cross sections are shown as a function of lattice spacing. Numerical descriptions are given for each of the ^{240}Pu cores. The calculated thermal neutron energy spectrum is shown. (M.L.S.)

40351 EXPERIMENTAL STUDY OF INTERACTION IN ARRAY OF FISSILE SPHERES. Kuvshinov, M. I.; Stsiborskii, B. D. At. Energ. (USSR), 22: 312-13(Apr. 1967). (In Russian).

The angular distribution of neutrons emerging from a sphere of Oralloy-90 (metallic uranium containing 90% ^{235}U) and the critical mass of two interacting spheres of Oralloy-90 were measured. The following assemblies were investigated: a solid sphere 13.5 cm in dia without a reflector; a solid sphere 15.1 cm in dia without a reflector; a solid sphere 16.7 cm in dia without reflector; a sphere 18.3 cm in dia with a hole 6.3 cm in dia in the center without a reflector; and a sphere 16.7 cm in dia with a hole 2.8 cm in dia in the center surrounded with an aluminum shell 30 cm in dia. A Po neutron source was placed outside the sphere at a distance L from the center of the sphere at various angles between the neutron detector and the sphere. The neutron multiplying coefficient and the average cosine of the angular neutron distribution were determined. A neutron source in the center of one sphere, and the other sphere were rigidly fixed at a distance L. By measuring the neutron flux from such a system and extrapolating the data to the critical state, it was found that the critical distance L was equal to 18.3 cm and L = 25 cm described above. (TTT)

38672 (WCAP-3726-1) PuO_2–UO_2 FUELED CRITICAL EXPERIMENTS. Leamer, R. D.; Orr, W. L.; Stover, R. L.; Taylor, E. G.; Tobin, J. P.; Vukmir, A. (Westinghouse Electric Corp., Pittsburgh, Pa. Atomic Power Div.). July 1967. Contract AT-(30-1)-3726. 129p. Dep. CFSTI.

A series of critical experiments using mixed-oxide (PuO_2–UO_2) plutonium fuels was carried out at the Westinghouse Reactor Evaluation Center (WREC). Two plutonium fuels with a variation in the ^{240}Pu isotopic content and one low enrichment uranium fuel were used in an experimental program which included buckling, reactivity, and power distribution measurements. Buckling measurements were made in five clean lattices with the 8% ^{240}Pu fuel and two clean lattices with the 24% ^{240}Pu fuel. With the 8% ^{240}Pu fuel, buckling measurements were made in two lattices at two different boron concentrations. The reactivity worths of voids, water holes, and control rods in different test arrays were determined in single and multi-region cores. Water hole and water slot power peaking effects were measured in clean and borated cores. Power distribution measurements were made in cores containing concentric regions of the different fuels, multi-region slab cores, and in cores containing interspersed fuels. (auth)

38720 INTERACTING ARRAYS OF CONTAINERS WITH ^{233}U SOLUTION. Lloyd, R. C.; Clayton, E. D. (Battelle-Pacific Northwest Lab., Richland, Wash.). Trans. Amer. Nucl. Soc., 10: 188-9(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

38725 ANALYSIS OF CRITICAL-FUEL-SOLUTION REACTORS CONTAINING ARRAYS OF VOID TUBES. Mayo, Wendell (Lewis Research Center, Cleveland). Trans. Amer. Nucl. Soc., 10: 232(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

2879 FEW GROUP DIFFUSION CALCULATIONS OF A LIGHT WATER MODERATED LATTICE. Mizuta, Hiroshi; Makino, Kakuji. J. Nucl. Sci. Technol. (Tokyo), 3: 309-19 (Aug. 1966).

An analysis was made of a light water moderated lattice based on few group diffusion calculations. The lattice under investigation consisted of 468 fuel rods in a square lattice arranged in a cylindrical core of 22.46 cm effective radius and 127.2 cm core height, with a water to fuel volume ratio of 2.918. The fuel was 2.02% enriched UO_2, clad in 0.8 mm thick Al tube. The theoretically calculated values for thermal, epithermal and fast neutron flux distributions, as well as the effective multiplication constant λ of the lattice, were compared with experimental data. After detailed analysis of the problems encountered in the course of the study the value of 0.9935 was determined for λ. Uncertainty in the nuclear data for fast neutrons would appear to constitute the greatest factor of error in λ. The discrepancies between the calculated and experimental activation distributions of the thermal, epithermal and fast neutrons amount to about 20, 10, and 15%, respectively, in the reflector region adjoining the core. The fact that these discrepancies cannot be removed by multigroup P_1 calculations would point toward insufficiency of the diffusion or P_1 calculation in this region. (auth)

24419 (UCRL-50175) SUMMARY REPORT OF CRITICAL EXPERIMENTS PLUTONIUM ARRAY STUDIES, PHASE I. Morton III, J. R.; Pierce, G. A.; Gardner, L. L.; Ball, C. J. (California Univ., Livermore. Lawrence Radiation Lab.). Dec. 22, 1966. Contract W-7405-eng-48. 56p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

A series of small-scale 3-dimensional arrays of 3-kg cylindrical α-plutonium billets was studied. The parameter measured was the surface-to-surface spacing of the billets at the critical condition. The following systems were studied: 2^3 ($2 \times 2 \times 2$) arrays bare, reflected and moderated; 2×2 arrays of 6-kg pairs bare and moderated; 3^3 arrays bare and reflected; and 3^3 arrays containing several perturbations. Multiplication measurements are also reported for an extensive group of irregular arrays, which were relevant to the operational safety of the program. (auth)

30245 (AECL-2593) DETERMINATION OF LATTICE PARAMETERS USING A FEW RODS. Okazaki, A.; Craig, D. S. (Atomic Energy of Canada Ltd., Chalk River (Ontario)). Mar. 1967. 83p. Dep. CFSTI. CAN $2.00.

A study has been made to determine whether lattice parameters of natural U, heavy water-moderated lattices can be obtained using as few as seven test fuel rods placed at the center of a driver or reference lattice of known properties. Up to seven test rods were substituted successively for reference rods and the critical height changes measured. Measurements were made for fuel clusters of natural UO_2 and U metal with D_2O, He and organic (HB-40) coolants, in 18 and 22 cm triangular pitch lattices. A two-group heterogeneous reactor calculation program (MICRETE) was used to determine the bucklings of the test rod lattices from the critical height changes. For D_2O-cooled rods the values are in good agreement with those obtained by conventional flux mapping in lattices containing a large number of test rods, even for buckling differences of 6 m^{-2} between test and reference lattices. The agreement is worse for He and HB-40 cooled rods. Measurements were made of the neutron flux distribution, Westcott spectrum parameters r and T, initial conversion ratio, and fast fission ratio using seven test rods and in general the results are in agreement with those made in lattices containing a large number of rods. (auth)

15791 DETERMINATION OF LATTICE PARAMETERS OF D_2O-MODERATED SYSTEMS FROM PULSED AND STATIC NEUTRON MEASUREMENTS. Parkinson, T. F.; Diaz, N. J.; Perez, R. B. (Univ. of Florida, Gainesville). Nucl. Energ., 262-70 (Nov.-Dec. 1966).

Experiments are described in which both the neutron die-away method and the static exponential method have been applied to a variety of natural-U, D_2O-moderated lattices. Three different fuel assemblies were used and data were obtained in both bare and side reflected systems. From the measured decay constants, k_∞ the infinite reproduction constant, and B_m^2, the material buckling, were evaluated. Theoretical studies were made of the multigroup, multiregion subcritical and the experimental decay constants for both bare and reflected systems agreed well with a two-region, two-group model. From the combined pulsed and static experiments, the dispersion law for multiplying media was derived. (auth)

14061 (AE-254) BUCKLING MEASUREMENTS UP TO 250 DEG C ON LATTICES OF AGESTA CLUSTERS AND ON D_2O ALONE IN THE PRESSURIZED EXPONENTIAL ASSEMBLY TZ. Persson, R.; Andersson, A. J. W.; Wikdahl, C.-E. (Aktiebolaget Atomenergi, Stockholm (Sweden)). Nov. 1966. 58p. Dep. mn.

Buckling determinations by means of flux mapping were performed in TZ up to 250 C on two lattices of Agesta fuel assemblies in D_2O and on D_2O alone. Most of the flux measurements were made with fission counters in pressure thimbles. The perturbations caused by the thimbles were studied experimentally in various ways and compared with two-group diffusion-theory calculations. In one of the lattices the effectiveness of a control rod (AgInCd) was also investigated. The results of the diffusion length experiments indicated some systematic error of the order of 0.15–0.10 m^{-2} in the bucklings measured, though the temperature dependence should be well established. The bucklings of the two lattices studied (square pitches 24 and 27 cm) were found to be less sensitive to temperature than theoretical calculations predict, the temperature coefficient being more than 10% smaller. The buckling changes from 20 to 250°C were about −2.4 and −1.8 m^{-2}, respectively, for the two lattices. During part of the experimental period about 30% unexplained excess absorption occurred in the heavy water. (auth)

38666 (NP-16873) EXPERIMENTAL AND THEORETICAL STUDIES OF MATERIAL BUCKLING AND DIFFUSION COEFFICIENTS IN SINGLE- AND MULTI-REGION NUCLEAR REACTOR LATTICES. Persson, Rolf (Chalmers Tekniska Hogskola, Goteborg (Sweden)). 1966. 15p. Dep.

Thesis.

An interpretation is given to the various kinds of buckling measurements performed in heavy water moderated exponential and critical facilities. Problems connected with boundary transients, anisotropy, and heterogeneity are discussed. The experiments adapt to the one-group and two-group models; single-region cores regarding higher harmonics, spectral transients, and heterogeneity effects in flux distributions; multi-region cores regarding differences in diffusion coefficients, spectral transients between regions, and buckling equivalence of control rods. Individual analytical abstracts of preprints appear in Nuclear Science Abstracts as NSA 11: 2139; NSA 16: 15766; NSA 16: 29993; NSA 18: 33035; NSA 19: 17200; NSA 20: 40377; NSA 21: 5677; NSA 21 14061; and Preprint No. V, which is going to be published in a revised form in Nukleonik and will be abstracted when it appears in public. (Sweden)

8101 ORGANIC-COOLED HEAVY-WATER-MODERATED ^{233}U-FUELED LATTICE EXPERIMENTS. Price, G. A.; Windsor, H.; Tunney, W.; Hellstrand, E. (Brookhaven National Lab., Upton, N. Y.). Trans. Amer. Nucl. Soc., 9: 518-19(Oct.–Nov. 1966).

19457 (BNL-50012) ORGANIC-COOLED, HEAVY WATER-MODERATED, ^{233}U FUELED LATTICE EXPERIMENTS. Price, G. A.; Windsor, H. H.; Tunney, W. J.; Hellstrand, E. (Brookhaven National Lab., Upton, N. Y.). Aug. 25, 1966. Contract AT(30-2)-Gen-16. 38p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Material neutron bucklings and thermal flux activation rates were measured for four lattices containing ^{233}U and ThO_2 with D_2O moderation. Accuracy of the buckling measurements is essentially limited by the short fuel length and the small number of clustered fuel elements. The METHUSELAH calculations underestimate the flux depression in the clustered elements by quite a large factor However, the effect on reactivity is relatively small because of compensating absorption in the stainless steel canister. METHUSELAH overestimates the buckling with a corresponding overestimate in k_{eff}. There is a need for improvement in the reactor calculations, although the source of error is not obvious. (auth)

38659 (BNL-50035) URANIUM-WATER LATTICE COMPILATION. PART I. BNL EXPONENTIAL ASSEMBLIES. Price, Glenn A. (Brookhaven National Lab., Upton, N. Y.). Dec. 30, 1966. 279p. Dep. CFSTI.

The Brookhaven National Laboratory exponential experiments with the lattice specifications are recorded for the further development of theoretical models and neutron cross section libraries necessary for the design of power reactors. The experiment to determine cadmium ratios is described. An appendix summarizes some of the ρ_{28}^{2} measurements. Measurements of the thermal disadvantage factor ζ^{Dy} are accomplished by irradiating small Dy foils in a fuel rod and in water. This technique is explained in an appendix. A fission catcher technique is used to measure the fast-to-thermal fission ratio, δ_{25}^{28}. Two techniques for measuring material buckling in exponential assemblies are described. A bibliography of 100 references is included. (J.C.W.)

8100 ANALYSIS OF 7- AND 19-ROD UO_2 CLUSTER EXPERIMENTS WITH THE HAMMER CODE. Risti, H. A. (Combustion Engineering, Inc., Windsor, Conn.). Trans. Amer. Nucl. Soc., 9: 517-18(Oct.–Nov. 1966).

12241 CRITICAL DIMENSIONS OF WATER-REFLECTED SYSTEMS CONTAINING ^{235}U–H_2O–Zr. Rutledge, G. P.; Dobbe, F. A.; Price, C. H. (Westinghouse Electric Corp., Idaho Falls, Idaho). Nucl. Appl., 2: 461-7(Dec. 1966).

Calculated values of the extrapolation distance for water-reflected ^{235}U–H_2O–Zr ternary systems are presented. This extrapolation distance, together with previously published critical buckling data, permits the determination of critical dimensions for all possible compositions of this system. Limited data were previously available for the extrapolation distance for the ^{235}U–H_2O binary system, and no data existed for the ternary system. A quantitative determination of the extrapolation distance was achieved utilizing, in a unique manner, nuclear codes developed for reactor design purposes. Accuracy of the results was confirmed at compositions for which experimental data are available. The extrapolation distance was found to be essentially independent of the shape of the system but strongly dependent upon composition. A single diagram that presents critical buckling and extrapolation distance as a function of composition was developed. With this diagram it is possible to determine critical dimensions for a given shape and composition and optimum conditions for criticality. As an important practical example, the minimum critical limits for optimally water-moderated cylindrical arrays of ^{235}U–Zr fuel elements are presented as a function of fuel-element length and composition. (auth)

40688 MEASUREMENT OF MATERIAL BUCKLING IN VARIOUS SUBCRITICAL ARRANGEMENTS OF NATURAL URANIUM AND LIGHT WATER. Schade, Diethard (Technische Hochschule, Darmstadt, Ger.). Nukleonik, 10: 54-8(July 1967). (In German).

Material buckling was measured in 10 subcritical arrangements. The fuel elements consisted of metallic natural U rods 2 cm diameter; the moderator was light water. Five of the investigated arrangements were crossed grids, which formed a Cartesian coordinate system by superposition of three parallel grids in the three axial directions, the remaining grids were formed by parallel rods in the moderator. The crossed grids yielded a smaller maximum material buckling than the parallel grids. The results indicated that material buckling can be increased by use of a smaller rod radius. Calculation of material buckling in the investigated parallel grids by methods used for interpretation of light enriched U–water reactors gave a satisfactory agreement with experimental values. (tr-auth)

29831 (MIT–2344–08) THE MEASUREMENT OF REACTOR PARAMETERS IN SLIGHTLY ENRICHED URANIUM, HEAVY WATER MODERATED MINIATURE LATTICES. Sefchovich, E.; Kaplan, I.; Thompson, T. J. (Massachusetts Inst. of Tech., Cambridge. Dept. of Nuclear Engineering). Oct. 1966. Contract AT(30-1)-2344. 204p. (MITNE–76). Dep. CFSTI.

Reactor physics parameters were measured in six heavy-water lattices which were miniature versions of lattices investigated extensively in the exponential assembly at M.I.T. The lattices consisted of 0.25-inch-diameter rods in two ^{235}U concentrations, 1.143 and 1.027%, and three spacings, 1.25, 1.75, and 2.50 in. The following quantities were measured in each lattice: the ratio of epicadmium to subcadmium capture rates in ^{238}U (ρ_{28}); the ratio of epicadmium to subcadmium fission rates in ^{235}U (δ_{25}); the ratio of the total capture rate in ^{238}U to the total fission rate in ^{235}U (C^*); the ^{238}U-to-^{235}U fission ratio (δ_{28}); the intracellular distribution of the activity of bare and cadmium-covered gold foils; and the axial and radial activity distributions of bare and cadmium-covered gold foils. Corrections derived from theory had to be applied to account for the presence of source neutrons and boundary effects. The age-diffusion model developed by Peak was improved and corrections were obtained to extrapolate the miniature lattice data to exponential, critical, and infinite assemblies. To test the validity of the extrapolation methods, the results obtained by extrapolating the miniature lattice data to exponential assemblies were compared with the results of measurements made in the exponential assembly at M.I.T. The extrapolated and measured results agreed generally within the experimental error. It is shown that to extrapolate the values of ρ_{28}, δ_{25}, and C^* measured in the miniature lattice to larger assemblies, it is only necessary to describe theoretically the measured spatial distribution of the cadmium ratio of gold. The experimental determination of the material buckling in miniature lattices was investigated. It is apparent that the inclusion of transport effects may be necessary, first, to define the material buckling and, second, to obtain its value. The correction factors for ρ_{28}, δ_{25}, and C^* were shown to depend on k_∞ so that k_∞ cannot be determined directly from measurements in the miniature lattice. An iterative procedure was developed to determine k_∞ which converges rapidly and, for the lattices investigated, led to results that were in agreement with the values of k_∞ obtained from measurements in the exponential assembly at M.I.T. (auth)

38717 MEASUREMENT OF REACTOR PARAMETERS IN SLIGHTLY ENRICHED URANIUM, HEAVY-WATER-MODERATED MINIATURE LATTICES. Sefchovich, E.; Kaplan, I.; Thompson, T. J. (Massachusetts Inst. of Tech., Cambridge). Trans. Amer. Nucl. Soc., 10: 194-5(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11–15, 1967. See CONF-670602.

38716 MEASUREMENTS AND ANALYSES OF SOME $^{233}UO_2$–ThO_2 WATER LATTICE EXPERIMENTS. Sehgal, Bal Raj; Windsor, H. H.; Tunney, W. J. (Brookhaven National Lab., Upton, N. Y.). Trans. Amer. Nucl. Soc., 10: 193-4(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11–15, 1967. See CONF-670602.

38668 (ORNL-CDC-1) CRITICALITY OF LARGE SYSTEMS OF SUBCRITICAL U(93) COMPONENTS. Thomas, J. T. (Oak Ridge National Lab., Tenn.). Aug. 1967. Contract W-7405-eng-26. 37p. Dep. CFSTI.

Methods for estimating the number of components required for criticality of unreflected and paraffin reflected systems of subcritical units are described. A neutron nonleakage fraction parameter is defined and leads to a correlation confirmed to within 5% of the number of units by comparison with experimental data for three dimensional cuboidal arrays. A density analogue representation of the arrays is readily derivable and is shown to approximate the results from the above method, but is less precise. Factors by which the number of units in an unreflected critical array is reduced by adding a paraffin reflector are found to range from about six to greater than 30 depending on the material and on the average uranium density considered. The methods are supported by Monte Carlo calculations demonstrated to be reliable by comparison with the results of critical experiments. (auth)

30138 (GA-7934) LATTICE PHYSICS STUDIES. Quarterly Progress Report for the Period Ending March 31, 1967. Trimble, G. D.; Gozani, T.; Neill, J. M.; Orphan, V.; Preskitt, C. A. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Apr. 11, 1967. Contract AT(04-3)-167. 85p. (EURAEC-1840). Dep. CFSTI.

Work Performed under United States-Euratom Joint Research and Development Program.

A facility for measurement of neutron spectra in tightly packed lattices was performance-tested. Foil activation analyses and time-of-flight spectral measurements were made in erbium nitrate. k_{eff} and dieaway time for the first experimental lattice was calculated. The possible use of a pressure vessel for elevated temperature measurements was evaluated. (M.L.S.)

38722 EXPERIMENTS AND CALCULATIONS FOR H_2O-MODERATED ASSEMBLIES CONTAINING UO_2-2 WT PERCENT PuO_2 FUEL RODS. Uotinen, V. O.; Williams, L. D. (Battelle-Pacific Northwest Lab., Richland, Wash.). Trans. Amer. Nucl. Soc., 10: 186-7(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

30257 (MLM-1395) NEUTRON MULTIPLICATION EXPERIMENT WITH ^{238}Pu DIOXIDE DOUBLE SEALED IN CALORIMETER PRESSURE CONTAINERS. Wolfe, R. A.; Edling, D. A.; Giessing, D. F.; Kahle, J. B.; Stubbins, W. F. (Mound Lab., Miamisburg, Ohio). Jan. 9, 1967. Contract AT(33-1)-Gen-53. 16p. Dep. CFSTI.

The neutron multiplication of high isotopic analysis PuO_2 assembled in various planar arrays in both water and air media was determined. The isotopic distribution of the Pu was approximately 80% ^{238}Pu, 16% ^{239}Pu, 3% ^{240}Pu, and 1% ^{241}Pu. Twenty-one double-sealed calorimeter pressure containers, each containing approximately 130 grams of the ^{238}Pu isotope, were assembled in three planar arrays consisting of 3-in., 1-in., and $\frac{1}{16}$-in. edge-to-edge spacing. No significant neutron multiplication was detectable in the fully assembled 1-in. planar array in the water medium or in the 3-in. planar array in the air medium. A low multiplying region ($M \approx 1.025$) did exist in the $\frac{1}{16}$-in. planar array but was not large enough to be used in reliably extrapolating to the critical mass of ^{238}Pu. (auth)

30334 SUB-CRITICAL MEASUREMENTS OF BUCKLING AND MIGRATION AREA USING SPONTANEOUS FISSION AS THE PRIMARY NEUTRON SOURCE. Ahmad, M.; Harris, M. J. (Univ. of Manchester, Eng.). J. Phys., D, 1: 645-51(May 1968).

Using spontaneous fission as the sole primary neutron source, measurements were made of the lattice constants of a sub-critical assembly fuelled with natural uranium rods of 1.2 in. diameter and moderated by water (V_m/V_u = 1.43). The experimental procedure is described and a simple two-group analysis is developed for interpreting the measurements. The thermal neutron fluxes were low, being of the order of 25 neutrons/cm^2 s; nevertheless the buckling was determined as $-(6.80 \pm 0.42)\ 10^{-4}$ cm^{-2}, a value in good agreement with that obtained from conventional exponential experiments. (auth)

50831 EXPERIMENTAL REACTOR RB-1. CALIBRATION AND MEASUREMENT OF K FOR A NATURAL URANIUM LATTICE MODERATED BY GRAPHITE. Aiello, P.; Azzoni, P.; Casali, F.; (and others) (CNEN, Bologna). pp 341-60 of Fisica del Reattore. Rome, Consiglio Nazionale delle Ricerche, 1966. (In Italian).

From Conference on Physics of Reactors, Milan. See CONF-469.

Reactivity measurements in RB-1 using the supercritical period method are described; calibration of the control rods and control system is discussed. Neutron flux measurements using activation detectors is described. Measurement of infinite neutron multiplication is summarized; data are tabulated. (M.L.S.)

35641 VARIATION OF k-inf WITH COOLANT DENSITY FOR A PLUTONIUM-FUELLED STEAM-COOLED FAST REACTOR LATTICE. COMPARISON OF EXPERIMENT WITH PREDICTION. Arnold, M. J.; Fox, W. N.; George, C. F.; Richmond, R. (United Kingdom Atomic Energy Authority, Winfrith, Eng.). pp 429-50 of Fast Reactor Physics. Vol. II. Vienna, International Atomic Energy Agency, 1968.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.2); CONF-671043-(Vol.2).

The design of a steam-cooled fast reactor with satisfactory safety and control characteristics and an acceptable breeding performance requires an accurate knowledge of the neutron balance and its variation with coolant density. When an assessment of such a system was undertaken the basic nuclear data available for use in the lattice calculation was of insufficient accuracy and little integral information on lattices of the required composition was available. A program of experimental measurements of the neutron balance and its variation with coolant density in lattices of this type was therefore undertaken. Measurements of important neutron reaction rates and of k-infinity were carried out in the reactor DIMPLE at A.E.E., Winfrith, in a small central fast reactor zone driven critical by a surrounding thermal reactor zone. This system produced at its center the correct fast spectrum while requiring only 50 kg of Pu in the central fast test zone. Three lattices were studied in which appropriate polypropylene plates were inserted in a regular array to simulate the flooded condition, and the operating condition (0.1 g/cm^3 equivalent steam density); the voided condition was studied by removing the plates. In all three lattices studied, the most important reaction rates influencing the coolant coefficient of k-infinity were fission in ^{239}Pu, capture in ^{238}U and to a lesser extent capture in ^{239}Pu. Particular attention was therefore paid to the direct measurement of these quantities which, together with other relatively unimportant fission rates, accounted for 80 to 90% of the total lattice absorption. In one lattice these measurements were supplemented by a direct measurement of k-infinity to provide a check on the calculation of unmeasured reaction rates. (auth)

30338 ZERO-REACTIVITY METHOD IN NUCLEAR PARAMETER EVALUATION OF HEAVY-WATER POWER REACTORS. Baccolini, A.; Casali, F.; Casini, G.; Giacobbi, V.; Hage, W.; Rustichelli, F.; Sturm, B. (CNEN, Bologna. EURATOM, Ispra, Italy). pp 643-53 of Heavy-Water Power Reactors. Vienna, International Atomic Energy Agency, 1968.

From Symposium on Heavy-Water Power Reactors, Vienna, Austria. See STI/PUB-163; CONF-670917.

A series of k_∞ measurements by the zero-reactivity method and by detailed parameter determinations has been carried out jointly by CNEN and EURATOM at the RB-1 reactor of Montecuccolino (Bologna), Italy. The aim of the experiments was to supply data for the development of the ORGEL-type reactor, to investigate the feasibility of the zero-reactivity method and to test the the reliability of the nuclear codes used. The experimental data were compared with the same quantities inferred from theoretical evaluations and with the results of substitution measurements performed at the ECO critical facility of Ispra. The fuel elements used were 19-rod uranium metal and 7-rod uranium carbide clusters, organic cooled. From the experimental and theoretical analysis it appeared that the results from the zero-reactivity method were consistent with the results obtained by the other experimental techniques and the theoretical calculations. It is to be outlined that, in the zero-reactivity experiments, the buffer region was constituted by only 8 ORGEL elements which represents an attractive economical advantage compared with other techniques. Moreover, it seems that when small changes in the test elements are involved, the measurement can be performed leaving the same buffer region without seriously affecting the accuracy of the results. (auth)

25008 DEVELOPMENT OF PLUTONIUM FUELS AND STUDIES OF PLUTONIUM RECYCLING IN LIGHT WATER REACTORS. Bairiot, H. (Belgo Nucleaire, Brussels); Debrue, J.; de Waegh, F.; Fossoul, E.; Motte, F.; Vanden Bemden, E.; Van Lierde, W. pp 335-78 of Development of Light Water Reactors in the Community. Brussels, Euratom, 1967. (In French).

See EUR-3561; CONF-661134.

Use of plutonium recycle in thermal water-cooled power reactors is discussed; economics are considered. The development of the fuel is outlined; fabrication, nondestructive testing and radiation testing are discussed. Thermal studies and neutronics calculational methods are described. Criticality studies using plutonium fuel elements in the VENUS critical assembly, fission ratio measurements, and neutron spectra determinations are outlined. All data are tabulated. 35 references. (M.L.S.)

53454 (BNWL-828) PLUTONIUM UTILIZATION PROGRAM. Technical Activities Quarterly Report, March-May 1968. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Sept. 1968. 67p. Dep. CFSTI.

CRITICALITY STUDIES—measurements of critical number of aluminum-plutonium fuel rods in water moderator, effect of lattice pitch and boron concentration on; microscopic lattice parameter measurements on water-moderated plutonium oxide (PuO_2)-uranium oxide (UO_2) rods; measurements of boron worth and concentration for cold and xenon-free PRTR Batch Core

NEUTRONS—capture-to-fission ratio measurements in plutonium-239; multiplication factor determination in water-moderated plutonium oxide (PuO_2)-uranium oxide (UO_2) lattices; resonance escape probability determination for uranium-238

33[illegible] MEASUREMENTS ON LATTICES OF UO_2-PuO_2 ROD CLUSTERS IN D_2O WITH D_2O, AIR, AND ORGANIC COOLANTS. Baumann, N. P.; Pellarin, D. J.; Olson, R. L.; O'Neill, G. F. (E. I. duPont de Nemours and Co., Aiken, S. C.). Trans. Amer. Nucl. Soc., 11: 254-5(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

48727 (BNWL-834) TUBULAR FUEL ELEMENT WITH INTEGRAL TARGET: PCTR EXPERIMENT. Bennett, R. A.; Newman, D. F.; Vaughn, A. D.; Davenport, L. C.; Robertson, D. E.; Bennett, C. L. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). June 1, 1968. Contract AT(45-1)-1830. 72p. Dep. CFSTI.

Two tubular fuel elements with integral target lattices have been investigated in the Physical Constants Test Reactor (PCTR). Multiplication constants, neutron utilization ratios, fast fission factors, and initial conversion ratios have been inferred from null reactivity experiments and measurements of spatial distributions of neutron reaction rates. Both wet and dry experiments were carried out in a three-by-three array of lattice cells 37.5" long placed in the cavity of the PCTR. (11 references) (auth)

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50815 COMPARISON OF CRITICAL EXPERIMENTS WITH CALCULATIONS FOR ORGANIC LATTICES. Bitelli, G.; Grifoni, S.; Martinelli, R.; Orestano, F. V.; Santandrea, E. (CNEN, Rome). pp 485-92 of Fisica del Reattore. Rome, Consiglio Nazionale delle Ricerche, 1966. (In Italian).

From Conference on Physics of Reactors, Milan. See CONF-469.

The core for ROSPO is described; nominal dimensions are given. Methods for calculating nuclear constants and reactivity for organic moderated lattices are described. Reactivity measurements are summarised; measured and calculated values are compared graphically. (M.L.S.)

50814 THEORETICAL ANALYSIS OF CRITICAL MEASUREMENTS IN A NATURAL URANIUM METAL-HEAVY WATER LATTICE. Bonalumi, Riccardo; Zorzoli, Giovanni Battista (CISE, Milan). pp 479-84 of Fisica del Reattore. Rome, Consiglio Nazionale delle Ricerche, 1966. (In Italian).

From Conference on Physics of Reactors, Milan. See CONF-469.

A summary description of the physical model used for criticality measurements in natural U-heavy water lattices is given. Results of the experiments are discussed; methods of data analysis are described. (M.L.S.)

35631 EXPERIMENTAL STUDY OF THE NEUTRON CHARACTERISTICS OF FAST CORES IN THE ERMINE THERMAL-FAST CRITICAL ASSEMBLY. Bouchard, J. (CEA, Fontenay-aux-Roses, France); Vidal, R.; Mougniot, J. C. pp 309-25 of Fast Reactor Physics. Vol. I. Vienna, International Atomic Energy Agency, 1968. (In French).

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.1); CONF-671043-(Vol.1).

A section of fast-reactor lattice was placed in the central hole of the MINERVE reactor, thus forming a coupled "thermal-fast" critical assembly. The construction of the first core, composed of the MASURCA 1-B lattice of 30% enriched uranium diluted in graphite, together with the experimental techniques used is described. Most measurements were carried out by the oscillation method, using an automatic regulating rod, to compensate for reactivity effects. The experiment was conducted with special care, so as to avoid electronic, mechanical and neutron perturbations as much as possible. The results of measurements of spectral indices and of reactivity effects of fissile, fertile and structural materials are presented. The results of experiments carried out to determine heterogeneity effects, the Doppler effect of ^{238}U and the importance functions are also presented. These experimental values are compared with values calculated from transport theory. (auth)

37543 MATERIAL BUCKLING MEASUREMENTS IN THE ANNA CRITICAL ASSEMBLY. Bouzyk, J.; Kubowski, J.; Latek, S. (Inst. of Nuclear Research, Swierk, Poland). At. Energ. (USSR), 24: 425-9(May 1968). (In Russian).

The differential reactivity method has been applied to determine the material buckling as well as the reflector savings in the reflected, heterogeneous, water-graphite moderated system with enriched uranium. The measuring technique and the results obtained are described. The results are discussed with respect to the application of the water height experiment to a complex reactor system. (auth)

18471 FURTHER REACTOR PHYSICS STUDIES FOR STEAM GENERATING HEAVY WATER REACTORS. PART I. UNIFORM CLUSTER LATTICES CONTAINING UO_2 OR PuO_2/UO_2 FUEL. Briggs, A. J.; Johnstone, I.; Kemshell, P. B.; Newmarch, D. A. (Atomic Energy Establishment, Winfrith, Eng.). J. Brit. Nucl. Energy Soc., 7: 61-90(Jan. 1968).

As part of the study of SGHW lattices, a wide range of uniform cluster arrays was studied. Both enriched UO_2 and PuO_2/UO_2 fuels were used, and the range included pin diameters from 0.3 in. to 0.6 in in clusters which contained from 37 to 90 pins each. Measurements of material buckling, detailed reaction rates and void coefficient are compared with theoretical predictions using METHUSELAH II, which is an improved version of the five-group diffusion theory code, METHUSELAH I, originally developed for SGHW assessment and design studies, and the 60-group, transport theory code, WIMS, which has superseded THULE. (auth)

3885 (AEEW-R-502) MEASUREMENTS OF MATERIAL BUCKLING AND DETAILED REACTION RATES IN A SERIES OF LOW ENRICHMENT UO_2 FUELLED CORES MODERATED BY LIGHT WATER. Brown, W. A. V.; Fox, W. N.; Skillings, D. J.; George, C. F.; Burholt, G. D. (Atomic Energy Establishment, Winfrith (England)). Sept. 1967. 129p. Dep. CFSTI. UK 19s. 0d.

Measurements were made in the DIMPLE reactor on a number of regular, 3% enriched, UO_2-light water lattices. Detailed reaction rate measurements were made in addition to the material bucklings, using moderator to fuel volume ratios from 3.16 to 0.78. One assembly was heated to 80°C; in addition, the coolant density change on heating to 250°C was simulated by inserting aluminium void pins into the lattice. (UK)

42368 EXPERIMENTS ON NATURAL URANIUM GRAPHITE LATTICES BY THE NULL REACTIVITY METHOD. Casali, F. (CNEN, Bologna); Ghilardotti, G.; Montagnini, B. J. Nucl. Energy, 22: 337-54(June 1968).

The measurement of the infinite multiplication factor of natural uranium-graphite lattices in the critical assembly RB-1, Bologna, by the null reactivity method, is described. The procedure which was set up for the execution of the measurements and for their interpretation is given in detail. The error on $k_\infty - 1$ was estimated to be of the order of 2 to 3%, in good experimental conditions. The lattices had been previously tested in the critical assembly Marius, by the substitution method, thus making possible a direct comparison between the two methods. It appears that a fair agreement exists between the two sets of results. (auth)

18577 CRITICALITY OF ARRAYS OF ^{233}U SOLUTION. Lloyd, R. C. (Battelle Memorial Inst., Richland, Wash.); Clayton, E. D.; Chalmers, J. H. Contract AT(45-1)-1830. Nucl. Appl., 4: 136-41(Mar. 1968). (BNWL-SA-1078).

The results of neutron multiplication measurements performed with arrays of ^{233}U solution apply to criticality safety considerations in handling solutions at a concentration of ≈ 330 g ^{233}U/liter and are useful in checking computational methods. The measurements were made with ≤17.3 kg ^{233}U in both reflected and unreflected arrays. Critical numbers of bottles were determined as a function of spacing, and the effect of adding moderating material between the bottles comprising an array was also examined. Monte Carlo calculations were found to reproduce the experimental data reasonably well, with k_{eff} being computed to within about 0.03 of unity for those cases compared. (auth)

22832 URANIUM AND URANIUM-PLUTONIUM FUELLED LATTICES WITH GRAPHITE AND HEAVY WATER MODERATOR. A COMPARISON OF EXPERIMENT WITH PREDICTION. Cogne, F.; Meyer-Heine, A. (CEA, Saclay, France). pp 48-58 of The Physics Problems in Thermal Reactor Design. London, British Nuclear Energy Society, 1967.

From International Conference on Physics Problems of Thermal Reactor Design, London. See CONF-670607.

Experiments on graphite-uranium lattices are compared to the COREGRAF code predictions. The agreement is satisfactory over a wide range of lattices, but a discrepancy remains as far as the temperature coefficient is concerned. Results on uranium-plutonium fuel experiments carried out in the heavy-water and graphite facilities Aquilon and César are also compared with calculations. (auth)

33006 REACTOR PHYSICS PARAMETERS OF 1.03 PERCENT ENRICHED URANIUM METAL, D_2O MODERATED LATTICES. D'Ardenne, Walter H. (Pennsylvania State Univ., University Park); Bliss, Henry E.; Lanning, David D.; Kaplan, Irving; Thompson, Theos J. Nucl. Sci. Eng., 32: 283-91(1968).

Reactor physics parameters were measured in three heavy water lattices consisting of 0.250-in.-diam, 1.03 wt % ^{235}U metal fuel rods in triangular arrays spaced at 1.25, 1.75, and 2.50 in. The following quantities were measured in each lattice: the ratio of epicadmium to subcadmium radiative captures in ^{238}U; the ratio of epicadmium to subcadmium fissions in ^{235}U; the ratio of radiative captures in ^{238}U to fissions in ^{235}U; and the fissions in ^{238}U to fissions in ^{235}U. These experimental results were used to calculate the following reactor physics parameters for each lattice: the resonance escape probability, the fast fission factor, the multiplication factor for an infinite system and the initial conversion ratio. Analytical results obtained by using THERMOS and GAM-I are in fair agreement with the experimental results. 11 references. (auth)

35593 (EURAEC-1882) RESULTATS D'EXPERIENCES SOUS-CRITIQUES REALISEES SUR DES RESAUX H_2O–UO_2 ENRICHIS A 5 PERCENT. (Results of Subcritical Experiments Carried Out on 5 Percent Enriched H_2O–UO_2 Lattices). Debrue, J.; Mewissen, L.; Motte, F. (Centre d'Etude de l'Energie Nucleaire, Brussels (Belgium)); Basselier, J.; Delrue, R.; Fossoul, E.; Haubert, N.; Lamotte, H.; Stievenart, M.; Van Deyck, D. (Societe Belge pour l'Industrie Nucleaire, Brussels). July 17, 1967. 199p. (In French). (EUR-3378). Dep. CFSTI.

Work performed under United States–Euratom Joint Research and Development Program.

The results of all the different types of measurements carried out on subcritical assemblies using UO_2 with a 5% ^{235}U enrichment in the THETIS reactor at the Centre d'Etude de l'Energie Nucléaire (CEN) at Mol are presented. The lattices studied were of the square pitch type and the H_2O/UO_2 ratio had successively the five following values: 1.51, 2.63, 4.13, 6.33 and 9.35. The maximum number of fuel elements available was 500, these having a useful length of 376.5 mm and a useful diameter of 7.6 mm. The cladding is steel (304) with a thickness of 0.2 mm. The following measuring techniques were used: the Inverse Multiplication Technique (IMT), the Source Ejection Technique (SET) and the Pulsed Neutrons Technique (PNT). A certain number of measurement results were selected in order to interpret the variation in the characteristics of the assemblies as a function of a determined variable (charging, "buckling," poisoning, etc.). (auth)

35594 (EURAEC-1882(Annex)) RESULTATS D'EXPERIENCES SOUS-CRITIQUES REALISEES SUR DES RESAUX H_2O–UO_2 ENRICHIS A 5 PERCENT. (Results of Subcritical Experiments Carried Out on 5 Percent Enriched H_2O–UO_2 Lattices). Debrue, J.; Mewissen, L.; Motte, F. (Centre d'Etude de l'Energie Nucleaire, Brussels (Belgium)); Basselier, J.; Delrue, R.; Fossoul, E.; Naubert, N.; Lamotte, H.; Stievenart, M.; Van Deyck, D. (Societe Belge pour l'Industrie Nucleaire, Brussels). 1967. 133p. (In French). (EUR-3378(Annex)). Dep. CFSTI.

Work performed under United States–Euratom Joint Research and Development Program.

Results of physics measurements carried out on subcritical assemblies using UO_2 with a 5% ^{235}U enrichment in the THETIS reactor are given. The lattices studied were of the square pitch type and the H_2O/UO_2 ratio had successively the five following values: 1.51, 2.63, 4.13, 6.33, and 9.35. The maximum number of fuel elements available was 500, these having a useful length of 376.5 mm and a useful diameter of 7.6 mm. The cladding is steel (304) with a thickness of 0.2 mm. The following measuring techniques were used: the Inverse Multiplication Technique (IMT), the Source Ejection Technique (SET), and the Pulsed Neutrons Technique (PNT). (D.C.C.)

6121 ANALYSIS OF FTR CONTROL-ROD EXPERIMENTS PERFORMED IN ZPR-3 ASSEMBLY 48 AND 48A. Engstrom, S. L.; Bennett, R. A. (Battelle-Pacific Northwest Lab., Richland, Wash.). Trans. Amer. Nucl. Soc., 10: 528-9(Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5–9, 1967. See CONF-671102.

33061 MEASURED CRITICAL SPACINGS OF PLUTONIUM ARRAYS. PART II. Finn, H. F.; Pierce, G. A. (Univ. of California, Livermore). Trans. Amer. Nucl. Soc., 11: 379-80 (June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

25086 (IEA-139) PROGRESSIVE SUBSTITUTION EXPERIMENTS IN UO_2 LATTICES MODERATED BY D_2O–H_2O MIXTURES. Franzen, H. R. (Instituto de Energia Atomica, Sao Paulo (Brazil)). Apr. 1967. 30p. Dep.

Buckling measurements for cores of uranium oxide (3.00% enriched in ^{235}U) 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 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 regions, 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 regions, one group theory. All the results were found to agree satisfactorily with the results from critical experiments. (auth)

33046 EXPERIMENTAL BUCKLINGS OF SIMULATED BURNED-UP NATURAL URANIUM CLUSTERS IN HEAVY WATER MODERATOR. French, P. M.; Solheim, R. (Atomic Energy of Canada, Ltd., Chalk River, Ont.). Trans. Amer. Nucl. Soc., 11: 251-2(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

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32999 EXPERIMENTAL MEASUREMENT OF K_∞ AND η IN SUBCRITICAL LATTICES. Harrington, J.; Lanning, D. D.; Thompson, T. J.; Kaplan, I. (Massachusetts Inst. of Tech., Cambridge). Trans. Amer. Nucl. Soc., 11: 48(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

22838 MEASUREMENT OF MATERIAL BUCKLING AND DIFFUSION COEFFICIENT IN HEAVY WATER MODERATED LATTICES CONTAINING NATURAL UO_2 FUEL CLUSTERS. Heer, W. (Eidgenoessisches Institut fuer Reaktorforschung, Wuerenlingen, Switzerland); Lutz, H. R.; Richmond, R.; Persson, R. pp 111-16 of The Physics Problems in Thermal Reactor Design. London, British Nuclear Energy Society, 1967.

From International Conference on Physics Problems of Thermal Reactor Design, London. See CONF-670607.

A description is given of a series of experiments carried out in support of Swiss reactor assessment studies on heavy water moderated lattices containing natural UO_2 cluster fuel elements. The experiments, which involved measurements of material buckling and diffusion coefficients, were designed to give a comparison of the results of measurements on single-zone lattices in the Swiss subcritical assembly MINOR with those given by substitution measurements in the Swedish reactor RO. A comparison of the two sets of experimental data confirmed that accurate results can be obtained by the substitution method using very small numbers of fuel clusters, but showed that, in some cases, the method may be significantly in error, and emphasized the need to investigate the conditions in which these errors may occur. The results of the buckling measurements were compared with the predictions of the UKAEA assessment code METHUSELAH and the Swedish code REBUS. Both codes were shown to give satisfactory results over the range of lattices examined. The measured coefficients were well predicted by the theory of Benoist. (auth)

12032 (NAA-SR-Memo-12472) NUCLEAR SAFETY SRE CORE III FUEL ELEMENT STORAGE. Ketzlach, Norman (Atomics International, Canoga Park, Calif.). May 17, 1967. 7p. Dep. CFSTI.

The nuclear safety of a single-plane array of SRE Core III fuel elements, spaced on 12-inch centers, has been evaluated. The array is less than 45% of critical, based on the mass/unit area principle for the fuel rod sizes of interest. This is independent of the degree of water flooding. A fully-flooded array of the assembly of elements spaced as stated above would have a k_{eff} of less than 0.6. An unmoderated assembly of fuel of the given enrichment having a neutron-reflection equivalent to water would require at least three times as much fuel for criticality, independent of spacing between elements or the thickness of water reflector. (auth)

5746 TEMPERATURE COEFFICIENTS OF PuO_2–UO_2–H_2O LATTICES. Kobayashi, S. (Hitachi, Ltd., Tokyo); Uotinen, V. O. Trans. Amer. Nucl. Soc., 10: 564-5(Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

1865 (JPRS-42322, pp 102-57) CRITICALITY OF SYSTEMS OF INTERACTING SUBCRITICAL ASSEMBLIES OF FISSIONABLE MATERIALS. Translated from pp 169-201 of Kriticheskie Parametry Sistem s Delyashchimisya Veshchestavami i Yadernaya Bezopasnost'.

Semi-empirical methods for evaluating the nuclear safety of a system of interacting subcritical assemblies are examined. The methods discussed are: (1) the equivalent dimensions method, (2) the method of the safe solid angle, (3) the interaction parameter method, and (4) the homogenization method. A summary of published experimental data is also presented. (H.D.R.)

37551 MEASUREMENTS OF FAST FISSION RATIO IN NATURAL URANIUM CARBIDE HEAVY WATER LATTICES. Maracci, G.; Rustichelli, F. (CCR-EURATOM, Ispra, Italy). Energ. Nucl. (Milan), 15: 330-4(May 1968).

The measurements of the fast fission ratio performed at the ECO reactor in natural uranium carbide heavy water lattices, by the integral gamma counting technique, are presented. The results of the measurements carried out on three different types of elements are compared with theoretical values calculated by the PINOCCHIO code; the agreement found is satisfactory in all the cases investigated. (auth)

16288 (NASA-TN-D-4270) ANALYSIS OF URANYL FLUORIDE SOLUTION REACTORS CONTAINING VOIDED TUBES. Mayo, Wendell (National Aeronautics and Space Administration, Cleveland, Ohio. Lewis Research Center). Feb. 1968. 24p. CFSTI.

Critical experiments with fully enriched (93.2% ^{235}U) uranyl fluoride–water solution reactors that contain arrays of large-diameter void tubes were analyzed satisfactorily. A calculational method that involves the direct application of widely used multigroup computer programs and techniques to cases of extreme heterogeneous voids is evaluated. Experimental critical solution heights for cores that contain no void tubes and for 19, 31, and 37 void tubes with a 7.658-cm diameter were obtained by using the NASA Zero Power Reactor-II facility. Both unreflected cores and cores radially reflected with 15.24 cm of water were considered. The void arrays with triangular lattice pitches of either 9.652 or 10.922 cm were centrally located in the 76.2-cm-diameter core tank. The critical heights of the voided reactors ranged from 21 to about 84 cm. The calculational method consists of first computing axial leakage rates from axially finite cylindrical cells that contain the void tube and a proportional amount of fuel solution. The cell dimensions and fuel are obtained from the corresponding critical reactors. Two-dimensional (r-z) S_4P_1 transport calculations with five energy groups of finite height cylindrical cells are used. The axial leakage rates per source neutron, obtained from the cell calculations, are incorporated into one-dimensional radially finite reactor calculations by defining an axial leakage cross section for each energy group to account for axial neutron streaming out of the voided region of the reactor. The advantage factors, which are also obtained from the two-dimensional cells, are important especially for the shorter height reactors that contain the more concentrated fuel solutions. The calculational method is satisfactory for the reactors examined and is readily adapted for use with other reactor configurations provided that two-dimensional (r-z) cells can be defined appropriately. (auth)

5103 ^{233}U–Th SEED-BLANKET CRITICAL EXPERIMENTS. Milani, S. (Bettis Atomic Power Lab., West Mifflin, Pa.). Trans. Amer. Nucl. Soc., 10: 595(Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

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14262 (WAPD-TM-614) SMALL ^{233}U FUELED SEED-AND-BLANKET CRITICAL EXPERIMENTS. Milani, S.; Weiss, S. H. (Bettis Atomic Power Lab., Pittsburgh, Pa.). Nov. 1967. Contract AT(11-1)-Gen-14. 167p. Dep. CFSTI.

A series of eight small seed-and-blanket critical assemblies were studied at the Bettis Atomic Power Laboratory. Rod-type seed fuel elements which contained either ^{233}U or ^{235}U were utilized so that a direct comparison could be made between the lattice characteristics of the two fuels. Also, blanket regions which contained rod type elements with either natural ThO_2 or 1 w/o $^{233}UO_2$–ThO_2 were compared. The eight assemblies were of two principal types. The first type was a rectangular array having a central seed region surrounded by a wet blanket with a metal-to-water ratio of about one, and the second was a hexagonal array having a central seed region surrounded by a tightly packed dry blanket with a metal-to-water ratio of about 9.2. All experiments on these eight assemblies were conducted at 20°C temperature. In addition, two of the wet blanket assemblies were studied in the High Temperature Test Facility at 180°F and 650 psi. (auth)

22844 INTERPRETATION OF CRITICAL EXPERIMENTS WITH URANIUM-PLUTONIUM FUEL ELEMENTS. Naudet, R. (Centre d'Etudes Nucleaires, Saclay, France). pp 109-25 of Fuel Burn-up Predictions in Thermal Reactors. Vienna, International Atomic Energy Agency, 1968. (In French).

From IAEA Meeting on Fuel Burnup Predictions in Thermal Reactors, Vienna, Austria. See STI/PUB-172; CONF-670418.

A set of critical experiments performed successively in the heavy-water pile AQUILON II and the graphite pile CESAR was designed to compare the neutron properties of the two lattices, which are geometrically identical but in which the fuel composition is slightly different. The same fuel elements were used in the two piles, i.e., uranium metal rods of uniform diameter. The following compositions were compared: Control set: natural uranium; Set used as an intercomparison standard: uranium very slightly depleted or enriched in ^{235}U (0.69%–0.83%–0.86%); Three sets to be studied containing plutonium: 1P: natural uranium + 0.04% plutonium; 2P: highly depleted uranium + 0.30% plutonium; 3P: similar to 1P but with plutonium containing a higher proportion of the isotope ^{240}Pu. In AQUILON the experiments were performed over a whole series of lattice spacings so as to vary the spectrum over a wide range, but all the measurements, with one exception, were made at environmental temperature. In CESAR, the experiments involved only one or two spacings, but a wide range of temperature. Up to the present time measurements have been made at 20, 100 and 200°C, and they will be continued at 300 and 400°C. The purpose of these experiments is to obtain data on which to evaluate the equivalence between ^{235}U and plutonium in conditions similar to those encountered in natural uranium reactors during irradiation, and to test the calculation methods. The experiments performed in AQUILON are described in greater detail, but the results already obtained in CESAR are also indicated by way of comparison. (auth)

18545 (BNWL-622) PCTR MEASUREMENT OF k_∞ FOR URANIUM LITHIUM SUPERCELL. Newman, D. F. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Dec. 20, 1967. Contract AT(45-1)-1830. 70p. Dep. CFSTI.

Experimental values for infinite medium neutron multiplication factor, k_∞, have been determined for a graphite moderated supercell utilizing 2.1% enriched U fuel and 2.93 wt % Li-in-Al target in separate process tubes, in a three-to-one ratio. Determinations of k_∞ were made with the coolant channels wet and dry to obtain an estimate of the reactivity coefficients due to water loss. Both a 6 × 6 and 5 × 5 array of test cells were inserted in the PCTR test cavity for these measurements. Results of experimental measurements are listed. (auth)

Critical Experiments: Lattices

7890 (AECL-2778) LATTICE MEASUREMENTS WITH 19-ELEMENT RODS OF ThO_2–$^{235}UO_2$ IN HEAVY WATER MODERATOR. PART I. BUCKLING, FINE STRUCTURE, AND NEUTRON SPECTRUM PARAMETERS. Okazaki, A.; Durrani, S. A. (Atomic Energy of Canada Ltd., Chalk River (Ontario)). Oct. 1967. 47p. Dep. CFSTI. CAN $1.50.

Lattice parameters were measured for heavy-water-moderated lattices of 19-element clusters of ThO_2 containing 1.5 wt % enriched UO_2 (93 at. % ^{235}U). Bucklings were determined from critical substitution measurements in the ZED-2 reactor using MICRETE, a two-group heterogeneous reactor code. Neutron density distributions and Westcott spectrum parameters r and T were obtained from measured ^{56}Mn, ^{116}In, and ^{177}Lu activities in foil detectors. Measurements were made for D_2O, air, H_2O and HB-40 coolants and for 22, 24, 28, and 32 cm triangular lattice pitches. (auth)

22836 CHALK RIVER STUDIES OF D_2O MODERATED NATURAL URANIUM AND ^{235}U–THORIUM LATTICES. Okazaki, A.; Towhes, B. M.; Durrani, S. A. (Atomic Energy of Canada Ltd., Ontario). pp 95-102 of The Physics Problems in Thermal Reactor Design. London, British Nuclear Energy Society, 1967.

From International Conference on Physics Problems of Thermal Reactor Design, London. See CONF-670607.

Progress is described on an investigation into the adequacy of the four-factor two-group description of the neutron cycle, by comparison with experimental measurements. Maximum use is made of experimental information, but some theory and basic nuclear data are involved. The calculated k_{eff} values show systematic trends with pitch and type of fuel cluster, and with coolant. Explanations for these trends are being sought. In addition, lattice parameter measurements on D_2O-moderated lattices of 19-element rods of ThO_2 containing 1.5 wt. % $^{235}UO_2$ are reported. The measurements were made for four coolants over a range of lattice pitches. (auth)

44882 (AECL-2779) LATTICE MEASUREMENTS WITH 19-ELEMENT RODS OF ThO_2–$^{235}UO_2$ IN HEAVY WATER MODERATOR. PART II. RELATIVE FISSION RATES, FAST FISSION RATIO, CONVERSION RATIO, AND COMPARISON WITH CALCULATIONS. Okazaki, Albert (Atomic Energy of Canada Ltd., Chalk River (Ontario)). June 1968. 41p. Dep. CFSTI. CAN $1.50.

Lattice parameter measurements of heavy water moderated lattices of 19-element clusters of ThO_2 containing 1.5 w/o UO_2 (93 a/o ^{235}U) are presented. Relative ^{233}U, ^{235}U and ^{239}Pu fission rates, fast fission ratio, and conversion ratio measurements are described. D_2O, air, H_2O and organic HB40 coolants were studied at 22 and 28 cm triangular lattice pitches. Additional analyses of the critical substitution measurements have been made and revised bucklings are given. Calculations made with the LATREP and HAMMER codes are compared with measured lattice parameters. Both codes underestimate the buckling by up to 1 m^{-2} with the largest discrepancy at the tightest lattice pitch. The calculations overestimate the neutron disadvantage factor and conversion ratio. (auth)

33026 BUCKLING AND PARAMETER MEASUREMENTS IN THE Th-D_2O ORGANIC SYSTEM. Pellarin, D. J.; Baumann, N. P.; Satterfield, R. M.; Vandervelde, V. D. (E. I. duPont de Nemours and Co., Aiken, S. C.). Trans. Amer. Nucl. Soc., 11: 253-4(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

33060 MEASURED CRITICAL SPACINGS OF PLUTONIUM ARRAYS. PART I. Pierce, G. A.; Morton III, J. R. (Univ. of California, Livermore). Trans. Amer. Nucl. Soc., 11: 378-9 (June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

53432 A REVIEW OF THE EXPERIMENTAL AND ANALYTICAL RESULTS OF THE HIGH-CONVERSION CRITICAL EXPERIMENT. Plumlee, Karl E.; Pennington, Edwin M. Reactor Fuel-Process. Technol., 11: 117-20(Summer 1968).

Criticality calculations and experiments for enriched UO_2-fueled water-moderated core are presented. The fuel elements consist of Type 304 stainless steel- and Al-clad pellet-shaped UO_2. Calculations for various geometry BORAX-5 and HI-C cores are presented. (D.C.C.)

53433 REVIEW OF RESULTS OF THE DIMPLE CRITICAL EXPERIMENT. Plumlee, Karl E. Reactor Fuel-Process. Technol., 11: 121-2(Summer 1968).

Physics measurements for four different cores in the DIMPLE critical assembly are presented. The cores consist of HI-C pellet-type enriched UO_2 fuel elements with Type 304 stainless steel and aluminum cladding. The assemblies are water-moderated. (D.C.C.)

6377 CRITICALITY OF HETEROGENEOUS ARRAYS UNDERGOING DISSOLUTION. Richey, C. R. (Battelle Memorial Inst., Richland, Wash.). Nucl. Sci. Eng., 31: 40-8(1968).

Experimental data to establish criticality control specifications for enriched uranium rods undergoing dissolution are extremely limited. A principal difficulty in treating the problem theoretically is that the resonance absorbing ^{238}U is admixed in the aqueous solution in which the rods are immersed. The "narrow resonance" and "infinite mass" approximations are applied; and from this application, expressions are developed for treating resonance capture by an absorbing lump embedded in a moderator admixed with the absorber. The computed change in the critical buckling of a heterogeneous array on replacing the water moderator by a uranyl nitrate solution is in good agreement with experiment. Results from survey calculations for 3 and 5 wt % ^{235}U rods latticed in uranium-water mixtures are given. It was concluded that for enrichments up to 5 wt % ^{235}U, dissolver vessels designed geometrically safe for water-moderated arrays of uranium rods will remain safe during the dissolution process. (auth)

6105 ANALYSES OF BROOKHAVEN ^{233}U-^{232}Th OXIDE, LIGHT- AND HEAVY-WATER-MODERATED LATTICE EXPERIMENTS. Sehgal, Bal Raj (Brookhaven National Lab., Upton, N. Y.). Trans. Amer. Nucl. Soc., 10: 596-7 (Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

1875 AN EXPERIMENTAL COMPARISON OF THE FAST AND THERMAL NEUTRON BUCKLINGS IN A URANIUM DIOXIDE LIGHT-WATER MODERATED CRITICAL ASSEMBLY. Shaw, Robert Alexander. Ithaca, N. Y., Cornell Univ., 1967. 64p. Thesis.

A comparison of the spatially dependent fast and thermal neutron flux profiles has been made in the Cornell University Zero Power Reactor, a light-water moderated and reflected, 2.1% enriched, UO_2 fueled, aluminum clad, triangular pitched critical assembly, using the core which has a nominal water-to-fuel ratio of 1.5 : 1. The radial buckling, defined by a least-squares fit of the radial flux profile to a J_0 Bessel function, is chosen as a convenient single-parameter characterization of the profile for comparison of flux profiles. The radial fast flux profile was determined by using a semiconductor detector to detect fission fragments from the fast fissioning of Th-232. The radial thermal flux was determined by the irradiation of manganese foils. The bucklings are found as a function of the radial distance (r) from the core center by performing the least-squares fit on data taken from the core center out to r. The thermal neutron reflector hump causes the thermal flux to deviate significantly from the J_0 dependence for r greater than about two-thirds of the core radius. Considering the region in which the thermal flux generally follows a J_0 dependence, the ratio of the thermal buckling to the fast buckling is found to be 1.131 ± .052 and 1.076 ± .052 for the two largest values of r in this region for which the bucklings are calculated. An analysis of theoretical fast and thermal flux profiles for this core which were calculated independently of this work supports the conclusion that the ratio of the thermal buckling to the fast buckling is greater than one. However, a precise comparison is not valid because the fast flux was theoretically calculated for neutrons in the energy range between 0.821 MeV and 10.0 MeV whereas the effective fission threshold for Th-232, used to experimentally determine the fast flux, is about 1.35 MeV. (Disser. Abstr.)

6115 CRITICAL EXPERIMENTS WITH PHOENIX FUEL IN AN MTR MOCKUP. Smith, R. I.; Davis, E. C. Jr.; Williams, L. D. (Battelle-Pacific Northwest Lab., Richland, Wash.). Trans. Amer. Nucl. Soc., 10: 537 (Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

33047 BUCKLING MEASUREMENTS IN A HEAVY-WATER SUPERLATTICE. Tanner, C. J.; Andrews, D. G. (Univ. of Toronto). Trans. Amer. Nucl. Soc., 11: 252-3(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

1565 (ORNL-CDC-4) EFFECT OF UNIT SHAPE ON THE CRITICALITY OF ARRAYS. Thomas, J. T. (Oak Ridge National Lab., Tenn.). Oct. 1967. Contract W-7405-eng-26. 19p. Dep. CFSTI.

The known criticality of uranium metal enriched to 93 wt % in ^{235}U as subcritical cylindrical components in unreflected arrays is used to explore component geometry effects in such arrays. Three basic geometries are chosen; the cube, the sphere, and the cylinder with equal height and diameter. It is shown that the multiplication factor of a unit defines sizes of the three shapes useful in establishing critical arrays equivalent to a given reference array. The number of components and the average uranium density of the equivalent arrays are the same as the reference array. The effects of changes in average uranium density and component mass on the array multiplication factor are investigated. (auth)

32976 (MIT-2344-12) HEAVY WATER LATTICE PROJECT FINAL REPORT. Thompson, T. J.; Kaplan, I.; Driscoll, M. J. (eds.) (Massachusetts Inst. of Tech., Cambridge. Dept. of Nuclear Engineering). Sept. 30, 1967. Contract AT(30-1)-2344. 208p. (MITNE-86). Dep. CFSTI.

CRITICALITY STUDIES — physics of heavy water-moderated slightly enriched uranium-fueled lattices, (E/T)
NEUTRONS — buckling in heavy water-moderated uranium oxide (UO_2)-fueled lattices, (E/T); multiplication constant measurements in heavy water-moderated uranium oxide (UO_2)-fueled lattices, (E/T)
URANIUM-235 — neutron fission ratio to uranium-238, measurement by γ spectrometry

50788 (AE-330) STUDIES OF THE EFFECT OF HEAVY WATER IN THE FAST REACTOR FRO. Tiren, L. I.; Haekansson, R.; Karmhag, B. (Aktiebolaget Atomenergi, Stockholm (Sweden)). Aug. 1968. 25p. Dep.

Core 9 of the FRO fast critical assembly was diluted with heavy water to 24 vol. per cent, contained in thinwalled Cu cans. Measurements of the critical mass and the reactivity coefficient of heavy water in this core are presented. The effect of the heterogeneous core composition on these items is also discussed. The results are compared with theoretical predictions using several computer codes. Criticality is accurately predicted, but the measured reactivity coefficient of heavy water is about 20% lower than the value obtained with the best available methods, involving the SPENG and DTF-4 programmes. The result of bunching measurements, in which the degree of heterogeneity of core composition was changed, is compared with theoretical estimates of the resonance shielding, flux advantage and leakage components of the heterogeneity effect. (auth)

44885 (BNWL-692) PCTR MEASUREMENT OF k_∞ FOR URANIUM BISMUTH SUPERCELL. Vaughn, A. D. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Feb. 1, 1968. Contract AT(45-1)-1830. 48p. Dep. CFSTI.

A measurement is made of the neutronic parameters for a low absorption target in a supercell consisting of a three-to-two fuel-to-target ratio with separate tubes of uranium fuel and bismuth metal target. Each tube is water cooled and surrounded by a graphite moderator on an 8.0″ lattice pitch. The bismuth target elements are relatively larger (3.750″ O.D.) than the uranium fuel elements (1.508″ O.D.) in order to irradiate as much of the target material as possible because of the significantly low bismuth cross section. Four inch square hole bars are removed from each target cell to contain a round aluminum process tube with the target elements. The graphite hole bars for the fuel cells contained 2.0″ O.D. holes and zirconium process tubes. The experiment is conducted in the Physical Constants Testing Reactor (PCTR) for a zero exposure case at room temperature. The reactivity effect of coolant loss is included in these measurements. (auth)

50800 (BNWL-792) PCTR MEASUREMENT OF k_∞ FOR URANIUM THORIA SUPERCELL. Vaughn, A. D.; Fleischman, R. M. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). July 31, 1968. Contract AT(45-1)-1830. 58p. Dep. CFSTI.

Experimental measurements were made on a test lattice of separable tubes of fuel and target material in a graphite-moderated lattice in the PCTR. Measurements were made of the infinite medium neutron multiplication factor on a defined supercell of three uranium fuel cells and one ThO_2 target cell. The measured parameters include data for two distinct fuel types in lattices with and without coolant consisting of 1.25 wt % enriched uranium fuel and ThO_2 target at 77% theoretical density in separate process tubes. Determinations of k_∞ were made with the coolant channels wet and dry and with two differing fuel sizes to obtain an estimate of the change in reactivity due to water loss as a function of coolant and fuel geometry. (auth)

22760 EXPERIMENTAL STUDIES OF UO_2–H_2O LATTICES OF UNIFORMLY ARRAYED RODS AND CLUSTERED ELEMENTS. Wajima, J. T.; Kobayashi, S.; Yamamoto, H.; Kikuchi, H.; Ohnishi, T.; Matsuoka, K.; Yamamoto, K. (Hitachi, Ltd., Tokyo). pp 67-72 of The Physics Problems in Thermal Reactor Design. London, British Nuclear Energy Society, 1967.

From International Conference on Physics Problems of Thermal Reactor Design, London. See CONF-670607.

Lattice studies of uniformly arrayed UO_2 rods and of clustered elements of a nuclear superheat reactor were performed and analyzed. Microparameters were measured and multiplication factors for the four-factor formula with ^{235}U epithermal fissions were obtained. Empirical formulae for microparameters were derived, and a method of measuring the nonleakage probability is presented. The analysis of the lattice of the clustered fuel element is discussed. (auth)

6116 ACTIVATION SHAPES IN ^{233}U- AND ^{235}U-THORIUM SEED-BLANKET CRITICAL ASSEMBLIES. Weiss, S. H.; Lehmann, P. H.; Mitchell, J. A.; Johnson, P. J.; Russell, C. D.; Wheat, L. L. (Bettis Atomic Power Lab., West Mifflin, Pa.). Trans. Amer. Nucl. Soc., 10: 537-8 (Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

22837 STUDIES OF SINGLE-ROD LATTICES OF UO_2 (NATURAL AND ENRICHED), PuO_2 OR ThO_2 IN HEAVY WATER. Wikdahl, C.-E.; Sokolowski, E.; Persson, R.; Jonsson, A.; Andersson, A. J. W. (Aktiebolaget Atomenergi, Stockholm). pp 103-10 of The Physics Problems in Thermal Reactor Design. London, British Nuclear Energy Society, 1967.

From International Conference on Physics Problems of Thermal Reactor Design, London. See CONF-670607.

Extensive studies were carried out on single-rod lattices in heavy water. Fuel rod composition, lattice pitch and (in some cases) temperature were the parameters varied. Bucklings, spectral indices and conversion ratios were measured. The experimental results were compared with calculated values obtained with the four-group program CAROL, and the more advanced cell program FLEF, which is based on multigroup integral transport theory. (auth)

6104 REACTOR PHYSICS PARAMETERS OF $^{233}UO_2$-ThO_2 LATTICES IN D_2O. Windsor, Henry H.; Tunney, William J. (Brookhaven National Lab., Upton, N. Y.). Trans. Amer. Nucl. Soc., 10: 595-6 (Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

III. REACTIVITY MEASUREMENT

1967

42725 (ANL-7357) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JULY 1967. (Argonne National Lab., Ill.). Aug. 31, 1967. Contract W-31-109-eng-38. 148p. Dep. CFSTI. CRITICAL ASSEMBLIES—perturbation measurements in ZPR-3, use of fissionable materials for; perturbation measurements in ZPR-3, use of boron and tantalum for; flux measurements in ZPR-6, adjoint; Doppler effect measurements in ZPR-9

38711 EXPERIMENTAL AND ANALYTICAL DETERMINATION OF THE REACTIVITY WORTHS OF ^{233}U, ^{235}U, ^{236}U, AND ^{237}Np SAMPLES IN A SERIES OF ^{235}U-GRAPHITE CRITICAL ASSEMBLIES. Bardes, R. G.; Gillette, E. M.; Nirschl, R. J.; Traylor, R. C. (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Trans. Amer. Nucl. Soc., 10: 198-200(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

8061 SURFACE-TO-MASS AND HETEROGENEITY EFFECTS ON THE SPECIFIC REACTIVITY OF SELECTED MATERIALS IN A FAST-NEUTRON SPECTRUM. Carpenter, S. G.; Mountford, L. A.; Springer, T. H.; Tuttle, R. J. (Atomics International, Canoga Park, Calif.). Trans. Amer. Nucl. Soc., 9: 504(Oct.-Nov. 1966).

35671 (ANL-7320, pp 107-15) COMPARISON BETWEEN EXPERIMENTAL AND THEORETICAL INTEGRAL DATA ON FAST CRITICAL FACILITIES. CALI, A PROGRAM FOR GENERATING "EFFECTIVE" NUCLEAR GROUP CONSTANTS BY A CORRELATION METHOD. Cecchini, G.; Gandini, A. (Comitato Nazionale per l'Energia Nucleare, Casaccia (Italy). Centro di Studi Nucleari); Dal Bono, I.; Faleschini, B. (Comitato Nazionale per l'Energia Nucleare, Bologna (Italy). Centro di Calcolo).

In order to treat simultaneously a large number of experimental data relative to a given material, a program (CALI) has been written which determines a new set of "effective" multigroup constants consistent with the experiments considered and as close as possible to a reference set obtained by best-value criteria. Proper allowance is given to the uncertainties of the reference cross sections and to the errors associated with the integral data themselves. This This allowance removes most of the difficulties encountered when correlating data obtained in assemblies with spectral characteristics not sufficiently different from each other with respect to the precision of the experimental techniques. A systematic comparison between experimental and theoretical data relative to the critical facilities ZPR-III, ZPR-VI, and ZEBRA is then performed. The quantities so far considered are the reactivity and fission ratios of Pu-239 and U-233 with respect to U-235. The theoretical predictions are obtained using the APDA 24-energy-group cross-section set and the Russian 26-group one. The new White data of Pu are also considered. The experimental results are corrected for sample size and shape effect. The corrected APDA Pu values and the Russian ones are more consistent with the experimental data than are the uncorrected values of the APDA set, when proper corrections for sample size are introduced. A list of 23 references is included. (auth)

14000 MEASUREMENTS OF VOID EFFECTS ON THE REACTIVITY. Chrysochoides, N. G. (Democritus Nuclear Research Center, Athens). Chim. Chron. (Athens, Greece), 30A: 137-42 (1965).

Two experimental methods were used to simulate voids in the reactor moderator: (a) Statical method, in which air gaps were introduced in the reactor moderator, (b) dynamic method, in which steam bubbles were created in the reactor moderator. The effects of the voids on the reactor reactivity were measured and the void coefficients were evaluated. The first method, which simulates better the case of uniformly distributed air in the moderator, gives accurate results. The proposed second method, which simulates better the case of real steam bubble formation in the moderator, is simple and gives satisfactory results for fair approximation. (auth)

30202 DETERMINATION OF THE EFFECTIVE MULTIPLICATION COEFFICIENT OF NEUTRONS BY THE MEASUREMENT OF THE DIFFERENTIAL REACTIVITY. Didelkin, T. S.; Shishin, B. P. At. Energ. (USSR), 22: 113-17(Feb. 1967). (In Russian).

Relations between the effective multiplication coefficients of neutrons in reactors and the experimental values of reactivity coefficients, determined by measurements of differential reactivities, were established. Correction terms were determined in integral form. (tr-auth)

19469 (KFK-473) INTERPRETATION OF DOPPLER COEFFICIENT MEASUREMENTS IN FAST CRITICAL ASSEMBLIES. Fischer, E. A. (Kernforschungszentrum, Karlsruhe (West Germany). Institut Angewandte Reaktorphysik). Oct. 1966. 22p. Dep. mn.

Analytical results are presented on Doppler experiments in which the reactivity change due to heating samples in fast critical assemblies are measured. A formalism is developed which allows calculation of reactivity changes due to sample heating. (J.R.D.)

40684 MEASUREMENT OF THE REACTIVITY COUPLING COEFFICIENT IN THE IOWA STATE UTR-10. Hendrickson, Richard A.; Danofsky, Richard A. (Iowa State Univ., Ames). pp 506-20 of Coupled Reactor Kinetics. Chezem, C. G.; Koehler, W. H. (eds.). College Station, Tex., Texas A and M Press, 1967.

From American Nuclear Society, Coupled Reactor Kinetics Conference, College Station, Tex., Jan. 23-24, 1967. See CONF-670107.

A cross-spectral density method for obtaining the reactivity coupling coefficient of a coupled core reactor is developed. An experimental measurement of the ratio of the reactivity coupling coefficient to the mean generation time of neutrons in the core is described. 6 references. (M.L.S.)

7923 MEASURED SODIUM-VOID COEFFICIENT AND FISSION RATIOS IN A LARGE ZONED FAST REACTOR. Karam, R. A.; Kato, W. Y.; Main, G.; Rusch, G. K. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 9: 487-8(Oct.-Nov. 1966).

25730 CENTRAL REACTIVITY MEASUREMENTS ON ASSEMBLIES 1 AND 3 OF THE FAST REACTOR FRO. Londen, Stig-Olof (Aktiebolaget Atomenergi, Studsvik, Sweden). Nukleonik, 9: 18-25(Jan. 1967).

The reactivity effects of small samples of various materials have been measured by the period method at the core centre of Assemblies 1 and 3 of the fast zero power reactor FRO. For some materials the reactivity change as a function of sample size has also been determined experimentally. The core of Assembly 1 consisted only of U enriched to 20% whereas the core of Assembly 3 was diluted with 30% graphite. The results have been 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 has generally been found, although discrepancies still exist. The spectrum perturbation caused by the experimental arrangement has been analyzed and found to be rather important. (auth)

38689 FFTF CRITICAL EXPERIMENTS: CONTROL-ROD STUDIES ON ZPR-3. Long, J. K. (Argonne National Lab., Ill.); Hess, A. L; McVean, R. L.; Persiani, P. J.; Ulrich, A. J.; Baird, Q. L.; Bennett, R. A.; Engstrom, S. L. Trans. Amer. Nucl. Soc., 10: 269-70(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

7872 (GAMD-7230) NULL REACTIVITY MEASUREMENTS OF MULTIPLICATION CHARACTERISTICS FOR HIGH-TEMPERATURE, GAS-COOLED REACTOR CORES IN THE MODIFIED HTGR CRITICAL FACILITY. Nichols, P. F. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Aug 23, 1966. Contract AT(04-3)-167. 36p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

Measurements of the multiplication characteristics for HTGR were made in the modified HTGR critical facility. The method of measurement involves experimental determination of the physical constituents of a unit cell with a multiplication factor of one when it is in a spectral environment characteristic of an infinite array of cells of the same type. The unit cell then has required poison added and the calculations of the poisoned cell compared with unity. These measurements are valuable for core samples with multiplication factors at nearly unity in the absence of poison. (J.C.W.)

27733 MEASUREMENT OF THE DOPPLER COEFFICIENT OF FAST REACTORS USING HEATED SAMPLES. Pucker, Norbert (Univ., Graz). Atomkernenergie, 12: 189-92(May-June 1967).

The measurement of the Doppler coefficient of a fast reactor due to heating a sample is discussed. The suitability of multigroup diffusion methods for this problem is investigated and results are presented within first order perturbation theory. (auth)

34078 (BNWL-442) EFFECTS OF UO_2 CRYSTALLINE BINDING ON DOPPLER COEFFICIENTS CALCULATIONS FOR FAST REACTOR SYSTEMS. Schenter, R. E. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). June 1967. Contract AT(45-1)-1830. 12p. Dep. CFSTI.

The change in the Doppler coefficient due to crystalline binding effects was studied for ^{238}U in a UO_2 lattice for fast reactor systems. The "weak binding," or the short compound nucleus lifetime (SCNL) approximation was used, where the ^{238}U atoms in the UO_2 lattice are treated as a free gas with an effective temperature. Calculations were made to determine the dependence of effective temperature versus temperature using the experimental phonon distribution of Dolling, et al. Comparisons were also made with effective temperatures obtained assuming a Debye model for the crystal. It was found that for temperatures above 300°K the effective temperature dependence was well represented by the Debye model (θ_D ~620°K). The temperature dependence of the reactivity was calculated for the ANL critical assembly, ZPR-III, 47, with the result that incorporation of the crystalline binding effects caused the calculated Doppler coefficient (dk/dT) to diverge from the "1/T" rule in the low temperature region, in qualitative agreement with the experimental results of Reynolds and Stewart. (auth)

1967

27739 INVESTIGATIONS ON THE ROSSI α EXPERIMENT. Schulze, Ekhard (Technische Hochschule, Darmstadt, Ger.). Nukleonik, 9: 85-97(Feb. 1967). (In German).

The Rossi α measurements which are in opposition to theory were described. Results of Monte Carlo calculations show that the basis for the discrepancies is not to be found in reactor kinetics. An extensive analysis shows that the physical bases of the Rossi α experiment have not been formulated rigidly enough up to the present. The mean time pattern of a neutron chain can be described through a decaying exponential function. Since only such neutron chains can be used experimentally which lead at least to an event, such chains were preferentially recorded which increase by an above average number of fissions. In this way, one of the exponential functions of average deviating time behavior was produced. This is not considered in the common theory. By the counting process, a new model representation was given and was developed formally. The theoretical considerations therefore lead to Rossi α experiments with special conditions which give a higher ratio and correlation term and background. As an application of the Rossi α measurements, a method for the determination of β/L was given in which no reactivity measurements were necessary. The theory of the Feynman α experiment was not used in the proposed new interpretation of the Rossi α experiment. (tr-auth)

21660 THE EFFECT OF PLUTONIUM ISOTOPIC COMPOSITION ON THE DOPPLER COEFFICIENT IN FAST REACTORS. Shaviv, G.; Yiftah, S. (Soreq Nuclear Research Center, Yavne, Israel). Nucl. Appl., 3: 213-16(Apr. 1967).

The effect of Pu isotopic composition on the Doppler coefficient is examined in fast reactors having different chemical compositions of the fuel and different core volumes. It is found that for a given core volume and chemical composition the absolute value of the Doppler coefficient increases with increase of the amount of high Pu isotopes (^{240}Pu, ^{241}Pu, and ^{242}Pu). (auth)

1968

12006 (ANL-7399) REACTOR DEVELOPMENT PROGRAM. Progress Report, November 1967. (Argonne National Lab., Ill.). Dec. 28, 1967. Contract W-31-109-eng-38. 172p. Dep. CFSTI.

CRITICAL ASSEMBLIES—reactivity measurements for small/sample perturbations in ZPR-3 fast; core material samples for ZPR-3 Assembly 50 fast, specifications for, mockup of FFTF in ZPR-3, composition of core for; atomic concentrations of ZPR-9 assemblies 19-21, reactivity worths of FFTF safety rods in ZPR-3, calculation of

24926 (APDA-214) QUARTERLY TECHNICAL PROGRESS REPORT ON AEC-SPONSORED ACTIVITIES, SEPTEMBER 15-DECEMBER 15, 1967. (Atomic Power Development Associates, Inc., Detroit, Mich.). Contract AT(11-1)-865. 122p. Dep. CFSTI.

CRITICAL ASSEMBLIES—reactivity worths and reaction rate ratios calculated for ZPR-3 and -4

32969 (GA-8468) RESULTS OF HTGR CRITICAL EXPERIMENTS DESIGNED TO MAKE INTEGRAL CHECKS ON THE CROSS SECTIONS IN USE AT GULF GENERAL ATOMIC. Bardes, R. G.; Gillette, E. M.; Nirschl, R. J.; Traylor, R. C. (Gulf General Atomic, Inc., San Diego, Calif.). Feb. 12, 1968. Contract AT(04-3)-167. 117p. Dep. CFSTI.

Cross-section data are presented for ^{233}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{232}Th, ^{239}Pu, ^{240}Pu, and boron, the latter being used as a standard. The neutron spectra in the five critical assemblies can be characterized by their mean fission energy. This ranged from 0.074 eV in the C/U-5000 assembly to 12.7 eV in the C/U-432 assembly. The softer spectra permit the study of cross sections in the thermal energy range while the harder spectra emphasize events in the epithermal range. The comparison between the calculated and measured results for the above materials in the five core assemblies shows the percent deviation of the calculated value for a given material from the measured values. The percent deviation represents the average for the different material loadings of each material investigated and includes an allowance for the estimated experimental uncertainties. (auth)

35636 CENTRAL REACTIVITY CONTRIBUTIONS OF ^{244}Cm, ^{239}Pu, AND ^{235}U IN A BARE CRITICAL ASSEMBLY OF PLUTONIUM METAL. Barton, David Maxwell (Los Alamos Scientific Lab., N. Mex.). Contract W-7405-eng-36. Nucl. Sci. Eng., 33: 51-6(1968). (LA-DC-9342).

Central reactivity contributions of gram-sized samples of ^{244}Cm, ^{239}Pu, and ^{235}U have been obtained in a fast critical assembly of bare ^{239}Pu in a spherical geometry. Resulting values are: ^{244}Cm = (1276 ± 5%) cents/g at.; ^{239}Pu = (1393 ± 3%) cents/g at.; ^{235}U = (701 ± 2%) cents/g at. From these data, the critical mass of a bare sphere of ^{244}Cm is estimated to be (27.7 ± 2.5) kg at a density of 13.5 g/cm^3. (auth)

33036 THE REACTIVITY WORTH OF SODIUM IN THE ZPR-3 PLUTONIUM ASSEMBLIES 48, 48B, AND 49. Broomfield, A. M. (Atomic Energy Establishment, Winfrith, Eng.); Matlock, R. G.; McVean, R. L. Trans. Amer. Nucl. Soc., 11: 240(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

22835 STUDIES OF BERYLLIUM OXIDE-MODERATED REACTORS USING MINIMAL QUANTITIES OF MATERIAL. Connolly, J. W.; Gemmell, W.; Marks, A. P.; Tattersall, R. B. (Australian Atomic Energy Commission Research Establishment, Lucas Heights, New South Wales). pp 83-90 of The Physics Problems in Thermal Reactor Design. London, British Nuclear Energy Society, 1967.

From International Conference on Physics Problems of Thermal Reactor Design, London. See CONF-670607.

Small BeO-moderated subcritical assemblies fueled with ^{235}U were studied in the internal reflector of an Argonaut-type reactor. Thoria was added in some cases. Measurements were compared with multigroup calculations, and the agreement was satisfactory. (auth)

32970 (GEAP-5584) SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. Twenty Fifth Quarterly Report, November 1967-January 1968. (General Electric Co., Sunnyvale, Calif. Advanced Products Operation). Mar. 1968. Contract AT(04-3)-189. 112p. Dep. CFSTI.

URANIUM-238—capture cross section of, error analysis for
PLUTONIUM-239—capture cross section of, error analysis for; fission cross section of, error analysis for

18563 FIRST EXPERIMENTS WITH THE ZERO POWER FACILITY ECO. Hage, W.; Hettinger, H.; Hohmann, H.; Metzdorf, H. J.; Toselli, F. Atomkernenergie, 13: 133-8(Mar.-Apr. 1968).

A short description is given of the zero power facility ECO, its reference fuel elements U/19/12, and a summary of the results obtained during the initial start-up experiments. Results of reactivity measurements on control elements and on certain perturbed core configurations are given, together with a list of the measured bucklings of the reference core. (auth)

25105 MEASURING THE GEOMETRICAL DIMENSIONS OF HOMOGENEOUS URANIUM-WATER REACTOR CORES HAVING SMALL-SCALE HETEROGENEITIES. Kontrovskii, S. N.; Levadnyi, V. A.; Yaroshevich, O. I. (Inst. of Nuclear Power Engineering, Minsk). Vestsi Akad. Navuk BSSR, Ser. Fiz.-Tekh. Navuk, No. 3, 32-7(1967). (In Russian).

The critical dimensions of a reactor, and the effective addition of reflector can be determined by making reactivity measurements on subcritical assemblies, and by noting a number of critical states of the active core at various settings of the control rods. These methods were found to be accurate ~2% for relatively small homogeneous cores. The effective addition of reflector and geometric size as measured by the first method was less accurate than that measured by the second method. The values obtained by both methods coincided within the limits of the experimental errors. The method of measuring neutron flux distribution was not applicable in this case. (TTT)

22821 (LA-DC-9216) CRITICALITY AND CENTRAL REACTIVITY CALCULATIONS USING ENDF/B DATA. LaBauve, R. J.; Battat, M. E. (Los Alamos Scientific Lab., N. Mex.). [1968]. Contract W-7405-eng-36. 7p. (CONF-680307-23). Dep. CFSTI.

From 2nd Conference on Neutron Cross Sections and Technology, Washington, D. C.

A series of criticality calculations has been performed for selected experimental assemblies to test the Category I, ENDF/B neutron data. These assemblies include JEZEBEL (plutonium core), TOPSY (enriched uranium core with natural uranium reflector), and ZPR-3 Assembly 48 (plutonium fuel, soft spectrum). Central reactivity worths were also computed for several materials of interest in fast reactor design. In the course of obtaining multigroup constants for input to the Los Alamos Scientific Laboratory codes used in these calculations, several ENDF/B retrieval and processing codes were employed. These include DAMMET, a code for rearranging and altering the mode of the standard BCD ENDF/B library tape; ETOE, a code for preparing an MC^2 library tape; and MC^2, a code for generating multigroup constants from microscopic neutron data. Calculational results have been compared with experiment as well as results obtained using other nuclear data libraries. (auth)

15925 (ORNL-TM-1736) PROMPT NEUTRON DECAY AND REACTIVITY MEASUREMENTS IN SUBCRITICAL URANIUM METAL CYLINDERS. Mihalczo, J. T. (Oak Ridge National Lab., Tenn.). Feb. 1968. Contract W-7405-eng-26. 35p. Dep. CFSTI.

Prompt neutron decay constants have been determined for unreflected and unmoderated subcritical cylinders of enriched uranium (93.15% ^{235}U) by the Rossi-α technique. The cylinder diameters were 17.77, 27.94, and 38.09 cm and the heights, at these diameters, varied from 10.184 to 2.548, 8.431 to 5.399, and 7.502 to 4.780 cm, respectively. The decay constants agreed to within 4% with those measured by the pulsed neutron method; the comparison with the results of S_n transport theory calculations showed disagreements as large as 20%. Reactivities as much as 33 dollars subcritical were determined from the prompt neutron decay constant at delayed criticality and changes in the prompt neutron lifetime with cylinder height calculated by S_n transport theory. These reactivities agreed favorably with values determined by an analog computer whose input was the response of an ionization chamber to power changes when an assembly was disassembled from delayed criticality to a given reactivity. (auth)

33004 PROMPT NEUTRON DECAY AND REACTIVITY MEASUREMENTS IN SUBCRITICAL URANIUM METAL CYLINDERS. Mihalczo, J. T. (Oak Ridge National Lab., Tenn.). Contract W-7405-eng-26. Nucl. Sci. Eng., 32: 292-301(1968). (ORNL-P-3620).

Prompt-neutron decay constants have been determined for unreflected and unmoderated subcritical cylinders of enriched uranium (93.15 wt % ^{235}U) by the Rossi-α technique. The cylinder diameters were 17.77, 27.93, and 38.09 cm and the heights varied from 10.184 to 2.548, 8.431 to 5.399, and 7.502 to 4.780 cm, respectively. The decay constants agreed to within 4% with those measured by the pulsed-neutron method; the comparison with the results of S_n transport theory calculations showed disagreements as large as 20%. The ratio of the prompt-neutron decay constant of a cylinder at delayed criticality to that of a subcritical cylinder and the ratio of the corresponding prompt-neutron lifetimes were used to obtain subcritical reactivities as great as 33 dollars. The lifetimes were calculated using neutron fluxes from S_8 transport theory. These reactivities agreed favorably with values determined by an analog computer whose input was the response of an ionization chamber to power changes when an assembly was disassembled from delayed criticality to a given reactivity. 11 references. (auth)

27466 (LA-DC-9404) REACTIVITY VALUES FROM THE LASL CROSS-SECTION TAPE LIBRARY. Mills, C. B. (Los Alamos Scientific Lab., N. Mex.). [nd]. Contract W-7405-eng-36. 3p. (CONF-680601-12). Dep. CFSTI.

From 14th Annual Meeting of the American Nuclear Society, Toronto, Ontario.

Effects of computer approximations on reactivity determinations are discussed. Computations of central reactivities in fast spectrum critical assemblies are tabulated as a function of neutron energy group; this shows reactivity dependence on energy. Comparison of reactivities, determined in this manner, allows neutron cross section evaluation. (M.L.S.)

1967

44895 (IN-1067) ANALYSIS OF ARMF MEASUREMENTS BY ONE-DIMENSIONAL MULTIGROUP DIFFUSION CALCULATIONS. Millsap, D. A. (Idaho Nuclear Corp., Idaho Falls). July 1968. Contract AT(10-1)-1230. 56p. Dep. CFSTI.

Descriptions of the ARMF-I and ARMF-II core loadings for reactivity measurements are given. Experimental procedures are given. Calculational procedures are described; energy and lethargy group structures are listed. Normalized unperturbed real fluxes, adjoint fluxes, and absorption statistical weights are tabulated. Results of measured and calculated values are shown graphically. Group cross sections and reactivities are tabulated for water, heavy water, Be, C, Mg, Al, Zr, Pb, and Bi. 12 references. (M.L.S.)

35667 (NAA-SR-Memo-12288) SURFACE-TO-MASS AND HETEROGENEITY EFFECTS ON THE SPECIFIC REACTIVITY OF SELECTED MATERIALS IN A FAST NEUTRON SPECTRUM. Mountford, L. A.; Springer, T. H.; Carpenter, S. G.; Tuttle, R. J. (Atomics International, Canoga Park, Calif.). Mar. 7, 1967. 21p. (CONF-661001-57). Dep. CFSTI.

From 14th Conference on Remote Systems and Technology, Pittsburgh, Pa.

The reactivity worths of several samples of U (of several enrichments), Th, Ta, and W have been measured in a neutron spectrum characterized by a median fission energy of 63 keV. The results have been used to determine the dependence of the specific worths on the surface-to-mass ratio (S/M) of the samples. The dependence on enrichment has also been measured for U with heterogeneous mixtures of ^{235}U and ^{238}U showing that the effect of using two samples of different enrichment was little different from that of using a corresponding uniform enrichment. The reactivity effect of thermal expansion was calculated from the (S/M) dependence and used to correct the total temperature coefficient of reactivity to obtain the effect of Doppler broadening. (auth)

32977 (NAA-SR-Memo-12289) THE CONTRIBUTION OF THERMAL EXPANSION TO THE TEMPERATURE COEFFICIENT OF REACTIVITY OF ^{235}U METAL SAMPLES. Mountford, L. A.; Springer, T. H.; Carpenter, S. G.; Tuttle, R. J. (Atomics International, Canoga Park, Calif.). Mar. 27, 1967. 23p. (CONF-650602-89). Dep. CFSTI.

From American Nuclear Society 11th Annual Meeting, Gatlinburg, Tenn.

The reactivity effect of thermal expansion of ^{235}U metal has been measured and calculated in a series of neutron energy spectra. The correction of the measured temperature coefficient of reactivity for this effect to obtain the Doppler effect is shown for one of these spectra. The results of temperature coefficient measurements with ^{238}U and Th in these spectra are given. (auth)

26767 SPATIAL DEPENDENCE IN THE MEASUREMENT OF THE REACTIVITY IN A SUBCRITICAL SYSTEM. Ortiz, G. Lopez; Olarte, F. J. (Junta de Energia Nuclear, Madrid). Nukleonik, 10: 329-30(Jan. 1968).

The reactivity of a heavy water–natural uranium carbide cylindrical system was measured using a 150-keV Cockcroft–Walton accelerator as the neutron pulsed source. Calculative techniques of Garelis and of Gozani were used in which system response to a neutron pulse is determined. (S.F.L.)

1968

47004 (APDA-216(Vol.1)) ANALYSIS OF SODIUM REACTIVITY MEASUREMENTS. VOLUME I. CROSS SECTION EVALUATION AND DATA TESTING. Pitterle, T. A.; Page, E. M.; Yamamoto, M. (Atomic Power Development Associates, Inc., Detroit, Mich.). June 1968. Contract AT(11-1)-865. 104p. Dep. CFSTI.

An analysis of sodium reactivity measurements in fast reactor critical assemblies is presented. Volume II presents the sodium-void analysis. In Vol. I, emphasis is placed on cross section evaluation and data testing conducted to establish the accuracy of the cross section data used for the calculations of Vol. II. Volume I describes the evaluation of the cross section data, testing of the data by comparison of calculations with integral experiment measurements, and an examination of methods used for critical assembly calculations. Neutron cross sections important for the sodium analysis have been evaluated as modifications to the ENDF/B data file. Calculations of criticality, reaction-rate ratios, and material worths using both the ENDF/B and modified ENDF/B data have been made for ZPR-III Assemblies 48, 48B, 49, and ZPR-VI Assemblies 2 and 3. The modified ENDF/B data are found to be over-reactive for these assemblies by 0.2% to 0.4% Δk while the ENDF/B data are under-reactive by 1.3% and 0.7% for the ^{239}Pu and ^{235}U-fueled assemblies, respectively. In general, the modified data yield better agreement with experiment than the ENDF/B data. Methods examined include resonance self-shielding techniques, variations in number of groups and geometrical representation, an investigation of absolute central worth discrepancies, and the use of cell-homogenized cross sections. The cell-homogenized cross sections are obtained as averages over transport-theory calculations of the spatial distributions of the flux in the plates forming a cell. These calculations indicate that the plate heterogeneities may have significant effects on the real and adjoint flux spectra. (auth)

47005 (APDA-216(Vol.2)) ANALYSIS OF SODIUM REACTIVITY MEASUREMENTS. VOLUME II. SODIUM VOID CALCULATIONS. Pitterle, T. A.; Page, E. M.; Yamamoto, M. (Atomic Power Development Associates, Inc., Detroit, Mich.). June 1968. Contract AT(11-1)-865. 104p. Dep. CFSTI.

The second of a two-volume report on an analysis of sodium reactivity measurements in fast reactor critical assemblies is presented. Volume I describes the cross section evaluation and data testing conducted in support of the sodium void analyses. Volume II describes the methods of calculation and the calculated results for an analysis of sodium void measurements performed in ZPR-III Assemblies 48 and 48B and ZPR-VI Assemblies 2 and 3. The detailed void measurements in the assemblies have been calculated using the MENDF/B cross section data described in Volume I, and additional calculations have been with the ENDF/B data. The basic method of calculation is perturbation theory in one- and two-dimensional diffusion theory. Methods examined include variations in resonance self-shielding techniques, number of groups, variations of perturbation theory including exact perturbation theory, transport theory, and heterogeneity considerations. For the heterogeneity analysis, cell-homogenized cross sections are obtained by flux-weighting over the cell using transport theory calculations of the spatial flux distributions. Overall calculations of the void reactivities with the MENDF/B data show better agreement with experiment, particularly for the ^{239}Pu-fueled assemblies. For centrally voided regions, the MENDF/B data yield the following qualitative agreement with experiment: 25% less positive than experiment for Assemblies 48 and 48B, 15% less negative than experiment for Assembly 2, and 25% more negative than experiment for Assembly 3. Regions with large leakage contributions tend to be consistently underestimated by about 15% with maximum discrepancies of 25% for all assemblies analyzed. Heterogeneity and transport theory effects on the void reactivities are found to be notably large for Assembly 48. (auth)

53292 EXPERIMENTAL AND THEORETICAL STUDIES OF REACTIVITY COEFFICIENTS OF IMPORTANCE FOR FAST REACTOR SAFETY. Tiren, Ingmar. Goteborg, Chalmers Tekniska Hoegskola, 1968. 23p.

Thesis.

A summary is made of 6 published papers by the author and coauthors on the following subjects: measurement and analysis of reactivity effects in empty channels in a fast reactor; tables related to the mean square chord length in right parallelepipeds; studies of the reactivity of polyethylene in the fast reactor FR-O; studies of the effect of heavy water in the fast reactor FR-O; activation Doppler measurements on ^{238}U and ^{235}U in some fast reactor spectra; and comparison of theoretical and experimental values of the activation Doppler effect in some fast reactor spectra. (Swed)

1873 STUDIES OF THE RATIO OF ^{238}U CAPTURE AND ^{235}U FISSION CROSS SECTIONS IN THE FAST REACTOR FR-O. Tiren, L. I. (Aktiebolaget Atomenergi, Studsvik, Sweden). Nukleonik, 10: 141-8(1967).

Measurements of the ratio of ^{238}U capture and ^{235}U fission cross sections have been made in five cores of the fast zero energy reactor FR-O, corresponding to three substantially different neutron spectra. The experimental results were calibrated by measurements in a thermal spectrum, for which the cross sections involved are accurately known. The capture rate in ^{238}U was detected by counting the ^{239}Np γ-activity of irradiated foils using the γ-X-ray coincidence technique, and the fission ratio in ^{235}U was obtained from the counting rate of a small fission chamber. The experimental results were reproducible to within about 1%. Systematic errors due to the heterogeneous core loadings and other effects add another 1 to 2% to the net uncertainties. The measured values obtained at the centers of the cores are in good agreement with results of multigroup calculations. (auth)

IV. NEUTRON FLUX SPECTRA

1967

12247 NEUTRON SPECTRUM MEASUREMENT IN A FAST CRITICAL ASSEMBLY. Bennett, E. F. (Argonne National Lab., Ill.). Nucl. Sci. Eng., 27: 28-33(Jan. 1967).

The neutron spectrum at the center of a large, dilute fast reactor was measured over the energy interval from 1 keV to 1 MeV. Resolution of the measurement was about 20% (FWHM) except at the lower energies. Errors in the measurement are described and a comparison made of the measured result with a multienergy-group calculation. There exists fair agreement between the measured spectrum and the multigroup calculation. (auth)

35686 (ANL-7320, pp 477-80) SPECTRUM MEASUREMENTS IN A LARGE DILUTE PLUTONIUM-FUELED FAST REACTOR. Bennett, E. F.; Gold, R.; Huber, R. J. (Argonne National Lab., Ill.).

Spectrum measurements have been made at the center of the ZPR-3 Assembly 48, a large, dilute, Pu-fueled fast reactor. The energy distribution of fragments from the $^{6}Li(n,\alpha)t$ reaction and of protons recoiling in a H proportional counter can be interpreted in terms of the neutron-energy spectrum. The results of measurements with the two techniques are compared, and the agreement is within estimated errors. (auth)

38687 EXPERIMENTAL NEUTRON-SPECTRUM COMPARISON FOR A ZONED AND A HOMOGENEOUS FAST CRITICAL ASSEMBLY. Bennett, E. F. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 10: 271-2(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

27798 (GEAP-5340) NEUTRON SPECTRUM MEASUREMENTS IN A FAST PLUTONIUM ASSEMBLY, ZPR-III ASSEMBLY 48. Brown, P. S.; Weitzberg, A. (General Electric Co., Sunnyvale, Calif. Advanced Products Operation). Jan. 1967. Contract AT(04-3)-189. 28p. Dep. CFSTI.

The neutron spectrum has been measured in ZPR-III Assembly 48, a dilute, Pu-fueled fast reactor. At the core center, a proton recoil spectrometer was used to cover the energy range from 14 keV to MeV and resonance sandwich detectors were used to cover the range from 18.8 eV to 2.95 keV. In the U-238 blanket, a proton recoil spectrometer was used to cover the energy range from 4 keV to 1 MeV. (auth)

35688 (ANL-7320, pp 486-9) COMMENT ON SPECTRUM MEASUREMENTS IN A LARGE, DILUTE PLUTONIUM-FUELED FAST REACTOR. Brown, P. S. (General Electric Co., Pleasanton, Calif. Nuclear Technology Dept.).

A generalized description of a proton-recoil spectrometer for neutron spectrum measurements is given. Experiments utilizing this spectrometer are briefly discussed. Comparisons of the spectrum at core center of ZPR-3 and in the blanket of ZPR-3 are shown graphically. (M.L.S.)

14027 FAST-NEUTRON SPECTRA IN WATER AND GRAPHITE. Harris, L. Jr.; Sherwood, G.; King, J. S. (Univ. of Michigan, Ann Arbor). Nucl. Sci. Eng., 26: 571-3(Dec. 1966).

Measurements of space-dependent fast neutron spectra in water and graphite at 2.0 and 12.0 MeV for fluxes directed normally to the Ford reactor core face with penetrations up to 60 Cm are reported. Comparisons of the measurement values were made with those calculated using the shielding code NIOBE. A solid-state proton-recoil telescope was used to measure neutron energies. (J.R.D.)

32365 (ANS-SD-2, Paper 3) IN-CORE EXPERIMENTS WITH A ^{6}Li "SANDWICH" FAST NEUTRON SPECTROMETER. Huber, R. J. (Argonne National Lab., Idaho Falls, Idaho). 18p.

The use of surface barrier detector—lithium-6 sandwiched fast neutron spectrometers for in-core measurements in the Zero Power Reactor III Assembly 45, a zoned fast critical having a soft (for fast reactors) neutron energy spectrum, and the Argonne Fast Source Reactor is described and the data obtained are presented. (B.G.D.)

38211 FAST-NEUTRON SPECTRA IN A DEPLETED-URANIUM SPHERE. Moore, R. A.; Neill, J. M.; Gozani, T.; Russell, J. L. Jr. (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Trans. Amer. Nucl. Soc., 10: 267 (June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

1967

40346 (RPI-328-87, pp 22-59) FAST NEUTRON SPECTRUM PROGRAM. (Rensselaer Polytechnic Inst., Troy, N. Y.).

Information is given on the design and development of a spherical lead electron target for use as the neutron source in fast assemblies. Measurements were made of the neutron spectrum from a 40 × 40 × 38 in.^{3} iron assembly pulsed with the lead target. The spectrum of neutrons from several re-entrant holes in the iron assembly were measured by time-of-flight using a $^{10}B_4C$–Na2 detector. An effort was made to formulate the adjoint fast reactor spectrum, and an application is given for a representative oxide core. A variational principle is developed for the determination of decay constants in pulsed fast neutron assemblies. A group constants averaging procedure for few-group importance-function and reactivity calculations in fast reactors is presented. A six-group importance function calculation is given in graphical form for GODIVA using importance-averaged and flux-averaged group constants. (S.F.L.)

30247 (AERE-R-5347) THE DETERMINATION OF THE FAST-NEUTRON SPECTRUM IN POSITIONS IN THE DAPHNE REFLECTOR. Silk, M. G. (Atomic Energy Research Establishment, Harwell (England)). Mar. 1967. 30p. Dep. CFSTI. UK 4s. 6d.

The difficulties inherent in spectrum determinations in reactors at positions remote from the core center are considered, and the suitability of ^{6}Li and ^{3}He semiconductor sandwich spectrometers for this work is discussed. Spectra obtained in many positions in the DAPHNE reflector are compared with calculated spectra. (auth)

35687 (ANL-7320, pp 481-5) A COMMENT ON THE COMPARISON OF THEORETICAL WITH EXPERIMENTAL NEUTRON SPECTRA IN FAST CRITICAL ASSEMBLIES. Travelli, A. (Argonne National Lab., Ill.).

The experimental data obtained by Bennett in the measurement of central neutron spectra in fast critical assemblies have a resolution of 15 to 20% in energy over the energy range, which extends approximately from 1 keV to 1 MeV. The high resolution, combined with the small statistical error of the data, makes it possible to measure, in addition to the macroscopic behavior of the neutron spectrum, some of the microscopic variations of the spectrum at those energies for which resonances of the light elements cause sharp variations of the transport cross section. The greater resolution of the experimental spectra raises the question as to how accurately the present analytical methods can predict the microscopic variations of the spectrum, and adds considerable interest to the comparison of experiment with theory. The results of two such comparisons are shown in which some detailed experimental spectra were matched to corresponding high-resolution calculations. A list of 14 references is included. (auth)

1968

4433 NEUTRON SPECTRUM MEASUREMENTS IN ZPR-3, -6, AND -9 USING HYDROGEN-RECOIL PROPORTIONAL COUNTERS. Bennett, E. F.; Gold, R. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 10: 577(Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5–9, 1967. See CONF-671102.

37276 (AERE-R-5364) MEASUREMENTS OF FAST NEUTRON SPECTRA IN REACTOR MATERIALS. Coates, M. S.; Gayther, D. B.; Goode, P. D. (Atomic Energy Research Establishment, Harwell (England)). Feb. 1968. 38p. Dep. CFSTI. UK 8s. 0d.

A series of neutron energy spectra emerging from spherical shells of natural uranium, polyethylene, graphite and sodium were measured in the energy region 300 eV to 6.5 MeV. The time-of-flight method was used with a 60-m path and a pulse source of fast neutrons provided by the 45-MeV linear accelerator. The source was located at the centre of the shell and spectra leaving the surface at 0 and 45° to a radius vector were determined. In addition a few measurements were made of spectra in the 0° direction from different penetration depths into the shell wall. The source was designed to emit neutrons isotropically, and the aim of the measurements was to provide spectra in simple one-dimensional systems in order to test the nuclear data sets used in reactor calculations. The experimental method is described in detail, and some comparisons based on discrete ordinate solutions of the Boltzmann equation are presented. (auth)

53425 NEUTRON PROPAGATION IN URANIUM DIOXIDE. PART I. SPACE-ENERGETIC DISTRIBUTIONS OF NEUTRONS. Golubev, V. I.; Golyaev, N. D.; Zvonarev, A. V.; Zizin, M. N.; Koleganov, Yu. F.; Nikolaev, M. N.; Orlov, M. Yu. At. Energ. (USSR), 25: 292-7(1968). (In Russian).

The experimental results of the neutron induced reaction velocities in a large block of UO_2 (dioxide reflector of BR-I reactor) are presented. The neutron spectrum in the large UO_2 block is investigated by the time-of-flight method and sandwich technique. The results of the measurements are compared with the calculations of the neutron propagation in UO_2 by 26-group set of constants. The results give the possibility to estimate the accuracy of the set of constants and calculation methods. (auth)

7619 MEASUREMENT OF REACTOR NEUTRON ENERGY DISTRIBUTION TO ~20 MeV. Kukhtevich, V. I.; Trykov, L. A.; Trykov, O. A. At. Energ. (USSR), 23: 191-5(Sept. 1967). (In Russian).

The spectra of the reactor investigated were measured at energies up to 19.5 ± 0.6 MeV using a scintillation spectrometer with a stilbene crystal and discrimination of the γ background during the excitation using filters of lithium hydride (21.4 g/cm^2) under conditions of good geometry. Some deviations of the measured energy distribution from the calculated and experimental fission spectra can be explained on account of the interaction of neutrons with the materials of the active zone. With transmission coefficients up to 10^{-3} to 10^{-4}, the total cross section was measured for water, carbon, and lead in the energy interval from 1 to 9 MeV with errors of 2.5 to 3% at $1.5 < E_n < 6$ MeV and 3 to 6% at $E_n > 6$ MeV. The total cross section for carbon and water at $E_n > 2$ MeV satisfactorily agrees with published data. For lead and water below 2 MeV, the sizes of the cross sections are lower (in comparison with the data for thick samples), which is explained by the passage of neutrons through the minimum value of the cross section. For lead in the energy range above ~3 MeV, a tendency to irregularities was observed. The dependence of the luminescence of the stilbene crystal on the energy made it possible to determine the scattering from carbon samples. (tr-auth)

50620 EXPERIMENTAL DETERMINATION OF THE AMOUNT OF FAST NEUTRONS IN THE FISSION SPECTRUM. Nasyrov, F.; Stalborskii, B. D. Dokl. Akad. Nauk SSSR, 180: 836-8(June 1, 1968). (In Russian).

If a small amount of moderator is contained in the reactor core, the reactor neutron spectrum may be used to study the spectral properties of fission neutrons. This problem was investigated within the core of a fast reactor operating with highly enriched ^{235}U; the core also contained 20% Fe, 10% Mo, and 10% ^{238}U. The neutron spectra were investigated in the 0.6 to 24 MeV region by means of threshold detectors. Fission chamber determinations and activation studies were also made. It was found that below 3 MeV, the reactor spectrum differs from the fission spectrum; this is due to the low energy transitions of neutrons caused by inelastic processes. Above 3 MeV, the spectra are fairly similar. The evaporation model characterizes the fission neutron spectrum up to 24 MeV; it remains to be determined how far this range extends. (TTT)

29829 (GA-8551) FAST NEUTRON SPECTRA IN MULTIPLYING AND NON-MULTIPLYING MEDIA. Neill, J. M.; Russell, J. L., Jr.; Moore, R. A.; Preskitt, C. A. (Gulf General Atomic, Inc., San Diego, Calif.). Feb. 23, 1968. Contract AT(04-3)-167. 17p. (CONF-680307-33). Dep. CFSTI.

From 2nd Conference on Neutron Cross Sections and Technology, Washington, D. C.

Fast neutron spectra were measured at various positions in spheres of depleted uranium and 93.2% enriched uranium, and these data were used to provide integral checks on the accuracy of neutron cross sections and computational methods. The data cover the energy range between about 10 keV and 15 MeV and were obtained using three flight path lengths, 45, 50, and 210 meters. The detectors used consisted of a 5-in. diameter NE-213 proton-recoil detector for fast neutrons and a 5-in. diameter NE-908 lithium glass detector for intermediate energy neutrons. The pulsed source for the measurements was obtained by impinging the beam from the Gulf General Atomic Linear Electron Accelerator onto tungsten or uranium targets. Several different types of calculation have been compared with the measurements, including multigroup transport theory, and two different sets of cross sections have been used. The measured spectra in the ^{235}U sphere are consistently softer than the calculated values. The measured spectra in the ^{238}U sphere are accurate enough to permit one to choose the better of the two cross section sets. (auth)

33031 FAST-NEUTRON SPECTRUM MEASUREMENTS IN THE CORE AND REFLECTOR OF ZPR-3 ASSEMBLY-51. Powell, J. E. (Argonne National Lab., Idaho Falls, Idaho). Trans. Amer. Nucl. Soc., 11: 217-18(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

30302 (GA-8377) FAST REACTOR SPECTRUM MEASUREMENTS. Quarterly Progress Report for the Period Ending October 31, 1967. Preskitt, C. A. (Gulf General Atomic, Inc., San Diego, Calif.). Nov. 21, 1967. Contract AT(04-3)-177. 16p. Dep. CFSTI.

Construction of the Subcritical Time-of-Flight Spectrum Facility (STSF) is described; problems encountered are discussed. An inventory of core materials present for loading the STSF is tabulated. Core and reflector composition for STSF-1 is given; reactivities of various loadings are tabulated. Neutron spectra are shown for various locations with respect to core content. Reactivity transients resulting from unintentional criticality while closing the beds are shown as a function of time. (M.L.S.)

32452 INTERMEDIATE-ENERGY FAST REACTOR SPECTRA BY TIME-OF-FLIGHT. Preskitt, C. A.; Neill, J. M.; Stevens, C. A. (Gulf General Atomic Inc., San Diego, Calif.). Trans. Amer. Nucl. Soc., 11: 216-17(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

47008 (GA-8715) FAST REACTOR SPECTRUM MEASUREMENTS. Annual Progress Report for the Period Ending April 30, 1968. Preskitt, C. A.; Neill, J. M.; Young, J. C. (Gulf General Atomic, Inc., San Diego, Calif.). June 3, 1968. Contract AT(04-3)-167. 48p. Dep. CFSTI.

A description of the STSF split-table assembly is presented. The first core loading for the assembly, designated STSF-1, is described. Description of the experimental electronics is presented. Data reduction procedures are analyzed. Spectra measurements for intermediate and fast neutrons are presented. (D.C.C.)

39548 (RPI-328-123, pp 18-25) FAST REACTOR PHYSICS EXPERIMENTAL. (Rensselaer Polytechnic Inst., Troy, N. Y.).

The analysis of time-of-flight measurements of fast neutron spectra in depleted uranium was continued, and computer programs were developed for analyzing time-of-flight data on fast neutron spectra. Preliminary results of measurements of position-dependent fast neutron spectra in iron are also reported. (D.C.W.)

48424 (RPI-328-133, pp 35-55) FAST REACTOR PHYSICS: EXPERIMENTAL. (Rensselaer Polytechnic Inst., Troy, N. Y.).

A series of position-dependent fast neutron spectrum measurements in rectangular assemblies of Fe and depleted U was performed; comparisons were made with theoretical predictions. Preliminary studies of an ^{6}Li glass detector for measuring intermediate neutron energy spectra were initiated. Some developments in detectors and associated electronic equipment for measuring fast neutron spectra are also summarized. (D.C.W.)

V. NEUTRON CROSS SECTIONS

1. Sources of Data

1967

19151 (NAA-SR-Memo-12315) THE ATOMICS INTERNATIONAL EVALUATED NUCLEAR DATA FILES AND ASSOCIATED COMPUTER PROGRAMS FOR THE AUTOMATED PREPARATION OF MULTIGROUP CONSTANTS. Alter, H. (Atomics International, Canoga Park, Calif.). Feb. 24, 1967. Contract AT(04-3)-701. 29p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Input data from the Atomics International Evaluated Nuclear Data File (AIENDF) tape are processed to give spectrum-weighted group-averaged neutron cross sections and other quantities required for the solution of the neutron transport equation by multigroup diffusion and transport theory methods. The calculation of multigroup Doppler-broadened effective resonance integrals and cross sections in the resonance region for a heterogeneous or homogeneous resonance absorber is based on the single-level Breit-Wigner theory with no overlapping of neighboring resonances. Inelastic and elastic scattering transfer matrices are calculated using differential scattering cross section data stored on the AIENDF Angular Distribution Data Tape (ADDT). Multigroup libraries are punched in any of the formats required by the one-dimensional multigroup diffusion theory codes, ULCER, FAIM, FAIM-CELL, CAESAR, and CAESAR IV; the one-dimensional multigroup S_n transport theory code, DTF; and the spectrum codes, FORM and AILMOE. (auth)

33852 (EANDC-58(U)) THIRD BIENNIAL REPORT OF THE ACTIVITIES OF THE EUROPEAN AMERICAN NUCLEAR DATA COMMITTEE. Bretcher, E.; Batchelor, R. (European-American Nuclear Data Committee). May 1966. 10p. Dep.

The activities of the European-American Nuclear Data Committee between January, 1964, and January, 1966, are summarized. (D.C.W.)

38149 (BNL-50061A) SIGMA CENTER, NEUTRON CROSS SECTION COMPILATION CENTER SCISRS NEWSLETTER. PART A. REFERENCE AND BIBLIOGRAPHY. (Brookhaven National Lab., Upton, N. Y.). June 1967. 139p. Dep. CFSTI.

A listing by reference of the literature incorporated into the SCISRS bibliographic system is presented together with a bibliographic index that is listed by accession number (reference abbreviation.) (D.C.W.)

42520 (BNL-50061B) SIGMA CENTER, NEUTRON CROSS SECTION COMPILATION CENTER SCISRS NEWSLETTER. PART B. ISOTOPE LISTING. (Brookhaven National Lab., Upton, N. Y.). June 1967. 107p. Dep. CFSTI.

A listing, arranged by isotope, is given of references to published and other material on various neutron cross sections. The cross-section types, dates of publication, and neutron energy ranges are included. The listing comprises the major part of the data contained in the Sigma Center Storage and Retrieval System (SCISRS) to date. (S.F.L.)

31736 (NAA-SR-11980(Vol.4)) COMPILATION, EVALUATION, AND REDUCTION OF NEUTRON DIFFERENTIAL SCATTERING DATA. Campbell, R. W.; Davis, S. K.; Alter, H.; Dunford, C. L., Berland, R. F. (Atomics International, Canoga Park, Calif.). Apr. 3, 1967. Contract AT(04-3)-701. 288p. Dep. CFSTI.

A compilation is given of the angular distributions of elastically scattered neutrons, based on available experimental data, for isotopes in the mass range $2 \leq A \leq 244$. These data were analyzed and reduced into a form convenient for use in digital computer calculations. The elastic differential scattering data are represented by expansions in a finite series of Legendre polynomials with energy-dependent coefficients in the center-of-mass coordinate system. A listing of the Legendre coefficients for all angular distributions, as well as plots of selected data, are included. The primary source of experimental data was report BNL-400, second edition. (S.F.L.)

1009 (BNL-10585) ACTIVITIES OF THE CROSS-SECTION COMPILATION AND EVALUATION CENTERS AT THE BROOKHAVEN NATIONAL LABORATORY. Chernick, Jack (Brookhaven National Lab., Upton, N. Y.). [1966]. Contract AT(30-2)-Gen-16. 7p. (CONF-661014-2). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From IAEA Conference on Nuclear Data, Paris, France.

A summary of activities in the compilation and evaluation of neutron cross sections is presented. (D.C.W.)

11989 (NYO-72-107) CINDA: AN INDEX TO THE LITERATURE ON MICROSCOPIC NEUTRON DATA. (Columbia Univ., New York. ENEA Neutron Data Compilation Centre, Gif-sur-Yvette (France)). July 1, 1966. Contract AT-30-1-GEN-72. 138p. (EANDC-60(U); CCDN-CI-11). DTI Extension.

A cumulative bibliography is given of the literature on microscopic neutron cross sections and allied data. The material is arranged in order of the target nucleus and by the type of data referenced. Coverage is generally limited to scattering and reactions induced by neutrons of energy <20 MeV, for specific elements and isotopes. Information on (γ,n) and (γ,f) reactions has been included for cases in which the γ-ray energy is less than ~15 MeV and the (γ,n) cross section greater than 0.1 mb. While most of the references covered report on measurements, CINDA also includes theoretical calculations, cross section evaluations, and compilations insofar as they refer to specific target nuclei. (S.F.L.)

35546 (ANL-7320, pp 3-14) STATUS OF BASIC NUCLEAR DATA REQUIRED FOR FAST BREEDER REACTOR DEVELOPMENT. Cox, S. A. (Argonne National Lab., Ill.).

The present precision and availability of nuclear data are discussed. Fission cross sections are examined in the light of recent ^{235}U measurement at A.W.R.E., Aldermaston. Since the ^{235}U fission cross section is used as a normalization standard for many cross-section measurements, its status is examined in detail. The normalization of neutron capture cross sections depends not only on the the ^{235}U fission cross section, but also on the ^{10}B(n,α) cross section and measurements of spherical shell transmission. Each normalization procedure is discussed in detail. It is concluded that much of the existing disparity in results for fission and capture cross sections would be greatly reduced by a critical renormalization of all of the data to currently acceptable values of the ^{235}U fission cross section and the ^{10}B(n,α) cross section. Recent re-evaluations of measurements of spherical shell transmission also tend to reduce the disparity. Neutron scattering cross sections from 0.3–1.5 MeV are in good shape. Because of the recent observation of intermediate structure in elastic neutron-scattering cross sections, measurements must be made in great detail especially below ~3–5 MeV. More data are needed from 1.5–5 MeV. Above 5 MeV the optical model can be used to interpolate between measurements rather widely spaced in energy. Recent $\bar{\nu}$ measurements confirm the nonlinearity of $\bar{\nu}$ vs E_n and also suggest the presence of an anomaly at ~0.4 MeV. Requests for nuclear data are examined according to the feasibility of the measurement and the man-year requirement necessary to achieve the requested precision. A list of 31 references is included. (auth)

21526 INELASTIC SCATTERING MEASUREMENTS IN A FAST REACTOR BY THE SPHERICAL SHELL METHOD. Davey, William G.; Amundson, Paul I. (Argonne National Lab., Ill.). Nucl. Sci. Eng., 28: 111-23(Apr. 1967).

The spherical shell method for investigating inelastic scattering cross sections was used in a fast-reactor core environment. The changes in ^{238}U/^{235}U, ^{236}U/^{235}U, and ^{234}U/^{235}U fission ratios caused by placing shells of graphite, sodium, aluminum, ion, stainless steel, lead, and depleted uranium around the fission chambers were measured. The studies show that reasonably accurate measurements can be made in a fast-reactor core. When comparisons can be made, our results are in excellent agreement with the fission spectrum results of Bethe, Beyster, and Carter. Comparisons of the experimental data with values calculated using two multigroup cross-section sets show clearly where these data sets are accurate and where they are in error. (auth)

11932 (TID-23357(Suppl.1)) CINDA: AN INDEX TO THE LITERATURE ON MICROSCOPIC NEUTRON DATA. (Division of Technical Information Extension (AEC), Oak Ridge, Tenn. ENEA Neutron Data Compilation Centre, Gif-sur-Yvette (France)). Oct. 15, 1966. 327p. (EANDC-66(U); CCDN-CI-13). Dep. mn. CFSTI $3.00 cy, $0.65 mn.

A compilation of all the additions entered into the CINDA master library tape between July 1, 1966 and Oct. 15, 1966 is presented. Corrections and improvements to previous entries are included. (D.C.W.)

23541 (NAA-SR-Memo-12314) EVALUATION OF HEAVY EVEN-EVEN NUCLIDE ELASTIC AND INELASTIC CROSS SECTIONS BY MEANS OF A NONSPHERICAL OPTICAL MODEL. Dunford, Charles L. (Atomics International, Canoga Park, Calif.). Mar. 15, 1967. Contract AT(04-3)-701. 23p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

The heavy even-even isotopes were investigated with a combined nonspherical potential optical model and compound-nucleus theory. Compound-nucleus theory provides a method for treating all the reaction cross sections [fission, capture, (n,n′), (n,2n)]. An analysis of ^{238}U and ^{232}Th for which considerable data is available has justified the usefulness of this approach to evaluation. The nonspherical optical model permits the evaluator to separate the elastic and inelastic components of a measured angular distribution for these isotopes. Combined with an appropriate compound-nucleus model, it is possible to differentiate between conflicting sets of ^{238}U capture data within the range of 10 to 100 keV. (auth)

35437 (EANDC(E)-76(U)) NUCLEAR DATA RESEARCH IN THE EURATOM COMMUNITY. Progress Report, January 1–December 31, 1966. (European-American Nuclear Data Committee). Jan. 1967. 203p. Dep.

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NEUTRONS, FAST—reactions (n,f) with plutonium-239 and -240 at 5 to 150 keV, cross sections for (E)
URANIUM-235—neutron fission cross section at 30 and 64 keV (E)
NEUTRONS, FAST—reactions (n,f) with uranium-235 at 30 and 64 keV, cross sections for (E)

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NEUTRONS, FAST—reactions (n,f) with uranium-235 at 7 to 14 MeV, neutron emission in (E)
URANIUM-235—neutron fission at 7 to 14 MeV, neutron emission in (E)

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1967

35670 (ANL-7320, pp 80-7) EVALUATION OF BASIC CROSS-SECTION DATA BY ANALYSIS OF FAST CRITICAL ASSEMBLIES. Fillmore, F. L.; Specht, E. R.; Vernon, A. R.; Ottewitte, E. H. (Atomics International, Canoga Park, Calif.).

Calculations of the properties of a number of fast critical assemblies are being carried out as part of a continuing program for testing and evaluating multigroup cross sections for fast calculations. The experiments included in the present analysis have been chosen to cover a wide range of fuel materials and reactor spectra. This increases the ability to identify errors according to the material and energy range in which they occur. The assemblies studied include a number of ^{235}U-fueled ZPR-III assemblies; ZPR-III-45 and 45A, driven Pu assemblies which were mainly used for Doppler measurements; the SEFOR mockup, ZPR-III-47; the Pu-fueled assembly, ZPR-III-48; ZEBRA Core 5; the small fast assemblies POPSY and JEZEBEL (fueled with metallic Pu), and GODIVA (fueled with metallic U). The properties examined include critical mass, Doppler effect, fission ratios, and the reactivity worths of various material replacements. The properties of the small fast assemblies depend only on the high-energy data; they are practically independent of cross sections below 100 keV. However, the larger assemblies require accurate data down to about 100 eV. All of the multigroup cross sections have been constructed using basic microscopic cross-section data and have not been adjusted to force agreement with the critical-assembly data. For the large, soft-spectrum assemblies, the cross sections were weighted by a fine-group spectrum calculated for each assembly. A list of 25 references is included. (auth)

21536 THE MEASUREMENT OF ^{239}Pu CAPTURE TO FISSION RATIOS IN FAST REACTOR LATTICES. Fox, W. N.; Richmond, R.; Skillings, D. J.; Wheeler, R. C. (Atomic Energy Establishment, Winfrith, Eng.). J. Brit. Nucl. Energy Soc., 6: 63-79(Jan. 1967).

The breeding gain of plutonium fuelled fast reactors is strongly influenced by the capture-to-fission ratio α_9 of ^{239}Pu. In the softer spectra associated with a large dilute fast reactor, the uncertainty in α_9 is of the order of ±25%. To reduce this uncertainty, two new techniques are being developed for use in zero-energy fast reactor lattices. In the first method, measurements are made in a lattice which is arranged to have an infinite multiplication constant k_∞ near to unity, so that k_∞ can be determined by a null reactivity technique without introducing significant systematic errors. All the important neutron fission and capture rates, except for the capture rate in ^{239}Pu, are then measured in this modified lattice; and α_9 is inferred from the known neutron balance. The second method, which is at an earlier stage of development than the first, involves the direct observation of capture and fission γ rays from a ^{239}Pu sample placed in a neutron beam taken from the zero-energy fast reactor core. A coincidence technique is used to distinguish between capture and fission γ rays, and the apparatus is calibrated by repeating the measurement in a thermal neutron beam for which α_9 is known. Some preliminary results obtained by the first technique indicate that current nuclear data sets underestimate α_9 significantly in dilute fast reactor lattices. (auth)

33958 (BNL-325(2nd Ed.)(Suppl.2)(Vol.2C)) NEUTRON CROSS SECTIONS. VOLUME IIC. Z equals 61 TO 87. Goldberg, Murrey D.; Mughabghab, Said F.; Purohit, Surendra N.; Magurno, Benjamin A.; May, Victoria A. (Brookhaven National Lab., Upton, N. Y.). Aug. 1966. 439p. Dep. CFSTI.

An updated compilation of neutron cross sections and resonance parameters is presented for Z = 61 to 87 nuclei. The reference sources for the data are also listed. (D.C.W.)

23536 (JAERI-1126) PROCEEDINGS OF THE 2ND SEMINAR ON FAST-NEUTRON CROSS SECTIONS [HELD IN TOKYO, 18-20 AUGUST 1966]. (Japan Atomic Energy Research Inst., Tokyo). Aug. 1966. 116p. (In Japanese). (CONF-660828). Dep. mn.

The 2nd Seminar on Fast-Neutron Cross Sections was held at the Tokai Research Establishment of the Japan Atomic Energy Research Institute on 18-20 August, 1966. About 70 scientists in the fields of the nuclear and reactor physics participated. The main topics were optical-model analyses, resonance analyses, and problems on fission cross sections. Some original papers presented at this Seminar, in addition to review papers on the above topics, are contained in this Proceedings. (auth)

33854 (GA-7169) GAF: A COMPUTER PROGRAM FOR CALCULATION OF NEUTRON SPECTRA AND AVERAGE CROSS SECTIONS IN THE HIGH ENERGY REGION. Mathews, D. R.; Archibald, R. J. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Jan. 20, 1967. Contract AT(04-3)-167. 75p. Dep. CFSTI.

A computer program, GAF, was written to compute neutron fluxes and currents from the B_1 equations for a maximum of 1740 energy groups above the thermal energy region. The calculated fluxes may be used to prepare average cross sections for up to 99 broad energy groups. Special data handling techniques are used to allow the practical utilization of such a large number of energy groups. The program is written in the FORTRAN-IV language for the UNIVAC-1108 computer. (auth)

25518 (WASH-1071) THE AEC NUCLEAR CROSS SECTIONS ADVISORY GROUP MEETING AT ARGONNE, ILLINOIS, NOVEMBER 10-11, 1966. Motz, H. T. (comp.) (Los Alamos Scientific Lab., N. Mex.). Contract W-7405-eng-36. 234p. (EANDC(US)-91-U; INDC(US)-5-U). Dep. mn. CFSTI $3.00 cy, $0.65 mn.

The status of activities in the cross-section measurement program is summarized. Preliminary data are included. (D.C.W.)

42780 SOME PROBLEMS FOR THE PRODUCTION OF REACTOR CONSTANTS. Nippon Genshiryoku Gakkaishi, 9: 386-95 (July 1967). (In Japanese).

An evaluation of available cross section data is given. The use of computer codes for generation of cross section libraries to be used with reactor calculations is discussed; the cross section accuracy, as calculated by these computer programs, is discussed. (M.L.S.)

1967

35382 (ANL-7320, pp 27-30) THE AUTOMATED PREPARATION OF MULTIGROUP CROSS SECTIONS FOR FAST REACTOR ANALYSIS USING THE MC^2 CODE. O'Shea, D. M. (Combustion Engineering, Inc., Windsor, Conn.); Toppel, B. J.; Rago, A. L. (Argonne National Lab., Ill.).

The fully automated multigroup-cross-section preparation program MC^2 makes use of an evaluated nuclear data file (ENDF) as its basic input. The degree to which the ENDF data is initially subdivided preparatory to averaging is variable in MC^2. Cross sections in the resolved resonance region are calculated through use of Doppler-broadened line shapes and include effects due to interference scattering and the influence of overlapping resonances in various isotopes in the mixture. Unresolved resonance cross sections are obtained by means of averages over suitable Porter-Thomas distributions of the neutron and fission widths. Quantities which are smoothly varying with respect to energy are represented in the library by coordinates of end points of linear segments taken from ln E vs ln σ, ln E vs σ, or E vs σ graphs. Inelastic scattering and $n,2n$ matrices are computed from excitation functions for individual levels and by use of a nuclear evaporation model above the resolved region. Elastic-scattering cross sections are computed from Legendre coefficients for the expansions of the angular distribution data for scattering. A fundamental-mode-weighting spectrum for the problem composition may be obtained for either the P1 or the consistent B1 or P1 approximations. Iteration on buckling to criticality is optional. The generated spectrum is used to contract the fine-group cross sections to a specified set of broad-group multigroup cross sections. (auth)

33966 (PAEC(A)IN-661) REACTOR CROSS SECTIONS. A Literature Search. (Philippine Atomic Energy Commission, Manila). Nov. 1966. 131p. Dep.

A bibliography of 805 entries is presented on nuclear reaction cross sections. The references cited were abstracted in Nuclear Science Abstracts during the period September 8, 1948, to October 1965. (S.F.L.)

35383 (ANL-7320, pp 47-53) PRESENTATION OF THE MULTIGROUP CROSS-SECTION SET PREPARED AT CADARACHE. Ravier, J.; Chaumont, J. M. (Commissariat a l'Energie Atomique, Cadarache (France). Centre d'Etudes Nucleaires).

A set of multigroup cross sections has been developed at Cadarache. After a short review of the sources of basic nuclear data, the report explains how the multigroup set was developed and how it is currently used; then a brief comparison between experimental and calculated results is given. A list of 22 references is included. (auth)

33853 (EANDC(E)-78(U)) INTRODUCTION TO KFK 120 "NEUTRON CROSS SECTIONS FOR FAST REACTOR MATERIALS." PART I. "EVALUATION." Schmidt, J. J. (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Neutronenphysik und Reaktortechnik). Mar. 1967. 12p. Dep.

The neutron data evaluation that has been performed at Karlsruhe since the beginning of the fast breeder reactor project in 1960 is outlined. (D.C.W.)

46683 (ANL-7325, pp 197-200) NUCLEAR CONSTANTS. Stupegia, D. C.; Tevebaugh, A. D.; Bingle, J. D. (Argonne National Lab., Ill.).

Developments are reported for studies on: neutron capture cross sections of reactor, structural, and control materials; and capture-to-fission cross-section ratios of fissile and fertile species. (P.C.H.)

1968

27086 (ANL-7375, pp 176-7) NUCLEAR CONSTANTS. (Argonne National Lab., Ill.).

Preliminary results in the program to measure capture cross sections for fast reactor materials and to measure and calculate capture-to-fission ratios for ^{239}Pu, ^{240}Pu, ^{242}Pu, ^{233}U, ^{235}U, and ^{238}U in EBR-II as a function of position are reported. (D.C.W.)

53165 (ANL-7425, pp 190-2) NUCLEAR CONSTANTS. (Argonne National Lab., Ill.).

Current progress is reported for neutron capture cross sections of reactor materials, capture-to-fission cross-section ratios of fissile and fertile materials irradiated in EBR-II, and preliminary investigations of tritium yields produced in fast neutron fission. (D.C.W.)

18515 THE U.S. EXPERIMENTAL PROGRAMME FOR FAST REACTOR PHYSICS. Avery, R.; Dickerman, C. E.; Kato, W. Y.; Long, J. K.; Smith, A. B. (Argonne National Lab., Ill.). pp 403-20 of Fast Breeder Reactors. Evans, P. V. (ed.). Oxford, Pergamon Press, 1967.

From British Nuclear Energy Society Conference on Fast Breeder Reactors, London. See CONF-660502.

Progress in basic cross-section measurements in the resonance, intermediate, and continuum regions is described. Critical experiments discussed relate to Doppler effects, Na void effects, and zoned core studies. Other test programmes reviewed include the SEFOR project, and the out-of-pile and TREAT experiments related to safety. (UK)

48414 (GA-8773) INTEGRAL NEUTRON THERMALIZATION. Quarterly Progress Report for the Period Ending June 30, 1968. Beyster, J. R.; Borgonovi, G. M.; Houston, D. H.; Koppel, J. U.; Slaggie, E. L.; Sprevak, D.; Young, J. A. (Gulf General Atomic, Inc., San Diego, Calif.). July 10, 1968. Contract AT-(04-3)-167. 50p. Dep. CFSTI.

A number of theoretical studies completed during this period are discussed. The final conclusions resulting from the first principles calculation for beryllium are summarized. A theoretical scattering law for UO_2 was completed; however, some additional numerical studies remain to be done before this work can be incorporated in the ENDF. A new model for polyethylene is also described, and comparisons between results of this model and experiment are presented. The lattice dynamical model for beryllium oxide was used to calculate a frequency spectrum and a scattering law in the incoherent approximation. A calculated total cross section for BeO, including coherent elastic scattering, is presented as part of this work. Some recently completed work on multiple scattering in double differential experiments is also described. This work relates to problems involved in the use of specially constructed samples designed to reduce multiple scattering. Reports on work in progress include an outline of efforts being made to improve capabilities for computing coherent inelastic scattering. Also some preliminary work is reported concerning efforts to broaden significantly the scope of the theoretical analyses underlying ENDF scattering laws by relating the temperature dependence of the frequency spectra to anharmonic effects. A report of the UC_2 total cross section experiment and the analysis of the data constitutes the section on experimental studies. (auth)

1968

35285 (AERE-PR/NP-13) NUCLEAR PHYSICS DIVISION PROGRESS REPORT FOR THE PERIOD MAY 1-OCTOBER 31, 1967. Coleman, C. F. (ed.) (Atomic Energy Research Establishment, Harwell (England)). Feb. 1968. 69p. Dep. CFSTI. UK.

Research dealing with nuclear data for reactors, nuclear structure and dynamics, radiation detectors, accelerator technology, Moessbauer applications, and astrophysics is summarized. (D.C.W.)

46698 (TID-24489) CINDA 68: AN INDEX TO THE LITERATURE ON MICROSCOPIC NEUTRON DATA. (Division of Technical Information Extension (AEC), Oak Ridge, Tenn. Gosudarstvennyi Komitet po Ispol'zovaniyu Atomnoi Energii SSSR, Obninsk. Fiziko-Energeticheskii Institut. ENEA Neutron Data Compilation Centre, Gif-Sur-Yvette (France). International Atomic Energy Agency, Vienna (Austria). Nuclear Data Unit). June 1968. 1026p. DTIE Free.

A bibliographic guide to experimental and theoretical information on neutron cross sections, resonance parameters, thermal scattering laws, fission parameters, and other related quantities is presented. A one-line format includes the element, isotope, or compound studied; the quantity studied; the type of investigation; the type of reference; the complete reference for the document; the incident neutron energy range; the laboratory at which the work was performed; and brief comments on the methods used and the results obtained in the investigation. (D.C.W.)

32422 (CCDN-NW-7) NEWSLETTER NO. 7. (ENEA Neutron Data Compilation Centre, Gif-sur-Yvette (France)). Mar. 1968. 73p. Dep.

A listing of the neutron cross-section evaluations that were available from CCDN in March, 1968, is presented. (D.C.W.)

37340 (EANDC(E)-89U) PROGRESS REPORT ON NUCLEAR DATA RESEARCH IN THE EURATOM COMMUNITY, JANUARY 1-DECEMBER 31, 1967. (European-American Nuclear Data Committee. European-American Committee on Reactor Physics). Feb. 1968. 239p. Dep.

Research activities and programs in the EURATOM community during the period Jan. 1 to Dec. 31, 1967, are summarized. Neutron cross section measurement programs comprise the bulk of the material included. (D.C.W.)

18078 (STI/PUB-140(Vol.1)) NUCLEAR DATA FOR REACTORS. Proceedings Series. Proceedings of a Conference held in Paris, 17-21 October 1966. (International Atomic Energy Agency, Vienna (Austria)). 1967. 589p. (CONF-661014-(Vol. 1)). IAEA: $12.00; Austrian Schillings 310,-; £ 4.4.8; F Fr 58,80; DM 48,-.

Separate abstracts were prepared for 23 of the 67 papers included. Thirteen papers are included in abstract form only. The other 31 papers have previously appeared in NSA and may be located by referring to CONF-661014 in the report number index. (D.C.W.)

For abstracts of individual papers see: 18099-18105, 18146, 18147, 18192-18195, 18281-18289, and 18347.

3738 NUCLEAR REACTIONS. Leonard, B. R. Jr. (Battelle-Northwest, Richland, Wash.). pp 9-30 of Plutonium Handbook. A Guide to the Technology. Vol. I. Wick, O. J. (ed.). New York, Gordon and Breach Science Publishers, 1967.

The more important features of neutron-induced reactions of the Pu isotopes are presented, with references to tabulations of detailed data. (S.F.L.)

20578 (WASH-1079) REPORTS TO THE AEC NUCLEAR CROSS SECTIONS ADVISORY COMMITTEE. Meeting held at Idaho Falls, Idaho, October 17-18, 1967. Moore, M. S. (comp.) (Idaho Nuclear Corp., Idaho Falls). 209p. (EANDC(US)-104-U; INDC(US)-2-U). Dep. CFSTI.

Progress in numerous investigations of neutron reactions and charged-particle reactions is summarized. Information, in varying degrees of completeness, is given on cross sections, resonance parameters, and level schemes. Some developments in instrumentation are also outlined. (D.C.W.)

39627 (BNL-50082, pp 19-24) NATIONAL NEUTRON CROSS SECTION CENTER. Pearlstein, S. (Brookhaven National Lab., Upton, N. Y.).

The activities of the center are summarized. Recommended values of the thermal neutron capture and fission cross sections and resonance integrals of ^{243}Am, ^{249}Bk, ^{250}Cf, ^{251}Cf, ^{252}Cf, ^{253}Cf, ^{254}Cf, ^{244}Cm, ^{245}Cm, ^{246}Cm, ^{247}Cm, ^{248}Cm, and ^{242}Pu, which are based on a systematic treatment of available data, are presented. The random matrix theory of nuclear cross-section fluctuations was extended to include the possibility of time reversal violation in nuclear forces. Optical-model parameters that provide a good fit to the cross sections of ^{206}Pb, ^{207}Pb, and ^{208}Pb at 1 MeV were obtained. Resonance parameters for 46 resonances of Ho below 235 eV were used in fitting total cross section data on Ho to a Breit-Wigner multilevel scattering and single-level absorption formula. (D.C.W.)

18194 MICROSCOPIC NEUTRON SCATTERING CROSS SECTIONS FOR REACTOR DESIGN. Smith, A. B.; Lister, D. (Argonne National Lab., Ill.). pp 399-408 of Nuclear Data for Reactors. Vienna, International Atomic Energy Agency, 1967. From IAEA Conference on Nuclear Data, Paris. See STI/PUB-140(Vol.1); CONF-661014-(Vol.1).

The results obtained from a comprehensive experimental study of elastic and inelastic neutron scattering are reported. The incident neutron energy interval was 0.3 to 1.5 MeV and scattering from 50 elements extending from Be to U was investigated. Fast neutron time-of-flight techniques including a multi-angle detector system and fully automated computer control were utilized to achieve a good scattered neutron resolution. Differential elastic and inelastic scattering cross sections were determined at eight or more angles at incident neutron energy intervals of 50 keV or less. The elastic angular distributions are expressed as Legendre expansions of up to six terms. The observed differential inelastic cross-sections are integrated to obtain the respective inelastic excitation cross-sections. The experimental results are compared with optical-model Hauser-Feshbach calculations; and it is shown that interpolations of experimental values, based on the model, are valid. Experimental evidence for intermediate resonance structure, width fluctuation effects, and nuclear deformation is presented. The influence of each on calculation is illustrated. (auth)

1968

20576 (ORNL-TM-2159) SOME RECENT NUCLEAR CROSS SECTION MEASUREMENTS AT ORNL, OCTOBER 1, 1967–FEBRUARY 29, 1968. Stelson, P. H. (Oak Ridge National Lab., Tenn.). Mar. 1968. Contract W-7405-eng-26. 20p. Dep. CFSTI.

Some recent measurements of differential cross sections and γ spectra for the reactions $^{14}N(n,n')^{14}N$, $^{14}N(n,p)^{14}C$, and $^{14}N(n,\alpha)^{11}B$; the neutron total cross section of ^{16}O; γ spectra from resonance and thermal neutron capture by ^{140}Ce, ^{39}K, ^{41}K, ^{185}Re, ^{187}Re, ^{203}Tl, ^{205}Tl, ^{112}Sn, ^{114}Sn, and ^{115}Sn; and capture and fission cross sections for ^{239}Pu from thermal to 30 keV and for ^{233}U from thermal to 1 eV are summarized. Calculations of the elastic and inelastic scattering cross sections for ^{56}Fe are also reported, as is the status of the electron linear accelerator. (D.O.W.)

35347 BASIC NUCLEAR DATA FOR THE HIGHER PLUTONIUM ISOTOPES. Yiftah, S.; Schmidt, J. J.; Caner, M.; Segev, M. (Soreq Nuclear Research Centre, Yavne, Israel). pp 123-49 of Fast Reactor Physics. Vol. I. Vienna, International Atomic Energy Agency, 1968.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.1); CONF-671043-(Vol.1).

Fast reactor fuel may have an appreciable content of high Pu isotopes, the amount varying according to the source. Therefore, in calculations of static and dynamic reactor characteristics, reliable basic nuclear data are needed not only for ^{239}Pu but for the higher isotopes as well. Reactor computations depend on cross sections as functions of energy, dilution and temperature. Basic nuclear data for a fuel isotope consists therefore of average cross sections and resonance parameters. Experiment alone does not furnish such data in a final complete form. In fact, the experimental information needs interpretation, weighting, evaluation and often also interpolation on theoretical grounds. Within the framework of contracted research with the Association EURATOM-Karlsruhe on fast reactors, an evaluation was made of basic nuclear data for the high Pu isotopes. Some pertinent parts and aspects of the evaluation are summarized. Resonance parameters and cross sections are presented for ^{240}Pu, ^{241}Pu, and ^{242}Pu in the form of experimental and recommended data. Complete sets of parameters include the first 43, 61, and 20 resonances of these isotopes, respectively. Average parameters are derived from these sets, to be used at higher energies where either the parametrization is incomplete or the resonances unresolved. Although the samples are, as a rule, too poor for a direct derivation of statistical distributions, there is enough general knowledge on the subject today to fix these distributions within narrow limits. (auth)

53379 MODERN FAST REACTOR CROSS SECTION SYSTEMS. Yiftah, S.; Segev, M.; Lemanska, M.; Caner, M. (Israel Atomic Energy Commission, Yavne). pp IIa9.1-8 of Proceedings of the International Conference on the Safety of Fast Reactors, Aix-en-Provence, September 19–22, 1967. Denielou, G. (ed.). Paris, Commissariat a l'Energie Atomique, 1967.

See CONF-670916.

Several classes of problems must be solved in the preparation of a group cross section set for fast reactor calculations. The first step is the evaluation of the basic nuclear data, including compilation of all available experimental information, calculations based on nuclear models to fill gaps in the data, clarification of inconsistencies and conflicting experimental information using systematics or computing weighted averages, in order to establish a complete almost point-wise scheme of the energy dependence of cross sections and other nuclear data in the energy range of interest. The second class of problems relates to the proper definition of group cross sections. When using existing group cross sections or constructing new ones, one must have a clear idea what type of group parameters are involved and exactly how their definitions as averages or integrals over products of basic nuclear data and weighting functions are given. The third problem concerns the weighting functions, which determine the group-averaging technique. Different group table types stem from different structures in the weighting functions, namely the gross structure, intermediate structure and fine structure. With the above consideration in mind, three modern group cross section sets are compared. These are the Russian 26-group set ABN, the Argonne 22-group set ANL, and the recently constructed Israeli 30-group set YSL. In the comparison, attention is paid to group structure, type of cross sections, representation and magnitude of self-shielding factors and temperature-dependence. Some typical fast reactor problems are run with the ABN and YSL sets and the results compared. (auth)

V. NEUTRON CROSS SECTIONS

2. ENDF/B Tapes and Other Evaluated Lists

1967

23404 (ORNL-P-2914) NEUTRON CROSS SECTIONS AND REACTION PRODUCTS FOR H, C, N, AND O FOR THE ENERGY RANGE FROM THERMAL TO 15 MeV. Auxier, J. A.; Brown, M. D. (Oak Ridge National Lab., Tenn.). [1966]. Contract W-7405-eng-26. 16p. (CONF-660920-15). Dep. mn. CFSTI $3.00 cy, $0.65 mn.

From 1st International Congress of the International Radiation Protection Assn., Rome, Italy.

The accurate calculation of neutron dose must be based on definitive cross sections and a precise knowledge of the reaction products in tissue. Although there are still several uncertainties in these parameters, a compilation has been made of the most detailed cross-section data available and reaction products for the four major elements in tissue (i.e., H, C, N, and O). The compilation is for neutron energies below 15 MeV, but the energy interval requiring the most study and analysis was that from 2.5 to 15 MeV. Particular attention was directed to the nonelastic reactions [e.g., the C(n,n′) 3α reaction]. Average values for the energies of the various charged particles as a function of the energy of the incident neutron have been computed. These values were compiled to provide a basis for revision of the dose-distribution functions for neutron exposures of man and of animals used in radiobiological studies. An analysis of the results of various measurements are compared with calculated values based on these cross sections and with the values listed in NBS Handbook 63. (auth)

33884 (LA-3695) ^{6}Li AND ^{7}Li DATA IN THE ENDF/B FORMAT Battat, M. E.; Dudziak, D. J.; LaBauve, R. J. (Los Alamos Scientific Lab., N. Mex.). Mar. 1967. Contract W-7405-eng-36. 39p. Dep. CFSTI.

At the Cross Section Evaluation Working Group (CSEWG) Meeting on June 9-10, 1966, at Brookhaven National Laboratory, the Los Alamos Scientific Laboratory was assigned the responsibility of preparing the data for the isotopes ^{6}Li and ^{7}Li for the first version of the Evaluated Nuclear Data File/B (ENDF/B) tape. These data were assembled in the ENDF/B format amd were sent to the Cross Section Evaluation Center (CSEC) at Brookhaven National Laboratory. Most of the data are from the AWRE data file originated by K. Parker of Aldermaston. Values for $\bar{\mu}_{LAB}$ and ξ, along with Legendre coefficients for the elastic scattering angular distributions, were received from H. Alter of Atomics International. Plots of the original AWRE cross section data converted to the ENDF/B format are presented. The ENDF/B listings for the ^{6}Li and ^{7}Li data, as they appeared on the first version (approximately February 1967) of the ENDF/B tape, are shown. (auth)

35480 FISSION AND THE SYNTHESIS OF HEAVY NUCLEI BY RAPID NEUTRON CAPTURE. Bell, George I. (Los Alamos Scientific Lab., N. Mex.). Contract W-7405-eng-36. Phys. Rev., 158: 1127-41(June 20, 1967). (LA-DC-8513).

The role of fission is examined in the synthesis of heavy nuclei by multiple capture of neutrons in thermonuclear explosions. Evidence from the recent Tweed and Cyclamen experiments indicating that neutron-induced fission is a serious source of depletion in neutron capture chains which start from targets of ^{242}Pu and ^{243}Am is reviewed. An analysis of Tweed abundances is made to obtain capture-to-fission ratios for the odd-A plutonium isotopes through A = 253. The liquid-drop model of Myers and Swiatecki plus empirical shell corrections and pairing energies is then used in order to correlate and predict spontaneous fission lifetimes and fission barriers. For nuclei having $Z \leq 101$ and $N \leq 157$, the shell correction is extrapolated, assuming it to be a function of N plus a function of Z. Thus, neutron binding energies, fission barriers, and spontaneous fission lifetimes for neutron-rich heavy nuclei are obtained. Capture-to-fission ratios are estimated for many of these nuclei, and qualitative agreement is found with laboratory and Tweed results. The extrapolation is continued out to N = 159 and Z = 104. It is concluded that by using the liquid-drop model plus semiempirical shell corrections, one can obtain capture-to-fission ratios and spontaneous fission half-lives which are usefully accurate. However, for predicting properties of nuclei having Z > 104, $N \gtrsim 159$, one needs, in this formalism, an accurate way of predicting shell corrections or nuclear masses. (auth)

25511 (LA-DC-8640) ^{233}U CROSS SECTIONS AS DETERMINED USING NEUTRONS FROM A NUCLEAR DETONATION. Bergen, D. W. (Los Alamos Scientific Lab., N. Mex.). [1966]. Contract W-7405-eng-36. 5p. (CONF-670602-1). Dep. CFSTI.

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif.

Some measurements of the capture and fission cross sections of ^{233}U in which target foils were exposed to a moderated flux from a nuclear explosion are summarized. Relation fission cross sections are presented for energies of 20 to 10^{6} eV, as well as information on the capture-to-fission ratio between 27 and 63 eV. The results of resonance analyses using both single-level and multilevel formalisms indicate that the fission of ^{233}U takes place through several channels. (D.C.W.)

1967

27568 (LA-3676) THE ^{233}U FISSION AND CAPTURE CROSS SECTIONS AND THEIR ANALYSIS AT LOW ENERGIES. Bergen, Delmar W. (Los Alamos Scientific Lab., N. Mex.). Dec. 1966. Contract W-7405-eng-36. 76p. Dep. CFSTI.

Thesis. Submitted to Univ. of New Mexico [Albuquerque].

The ^{233}U fission and capture cross sections were measured using a nuclear-device neutron source and time-of-flight techniques. Cross section data are presented from 20 to 10^{4} eV for fission and from 20 to 63 eV for fission + capture. The resonance region (20 eV to 63 eV) was fitted with both a single-level function consisting of a sum of Breit–Wigner levels and the Reich and Moore multilevel function based on R-matrix theory. The resulting resonance parameters are listed and discussed. In order to establish the validity of the resonance parameters derived from the multilevel fit, a study is presented of the cross section derived from two and three hypothetical resonances under various conditions and of the cross sections obtained from randomly generated resonances. (auth)

38306 URANIUM-233 CROSS SECTIONS AS DETERMINED USING NEUTRONS FROM A NUCLEAR DETONATION. Bergen, D. W. (Los Alamos Scientific Lab., N. Mex.). Trans. Amer. Nucl. Soc., 10: 221(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

27579 (UNC-5099) NEUTRON CROSS SECTIONS OF ^{238}U, ^{235}U, ^{237}U, ^{239}U, ^{234}U, ^{236}U, ^{239}Pu, ^{240}Pu, W, Pb, Ni, Cr, C, ^{6}Li, ^{7}Li, AND T. Bertin, M. C.; Celnik, J.; Flesher, H.; Grochowski, G.; Kalos, M. H.; Ray, J. H.; Troubetzkoy, E. S. (United Nuclear Corp., White Plains, N. Y. Development Div.). Dec. 31, 1964. Contract DA-18-108-AMC-131(A). 292p. (AD-616629).

Neutron cross-section sets were prepared for 16 elements or isotopes for neutron energies from 0.037 eV to 18 MeV. The cross sections tabulated include the total, elastic, inelastic, (n,2n) and fission cross sections, as well as cross sections for charged-particle emission. Information is also given on the angular distribution of elastically scattered neutrons and on the energy distribution of neutrons and γ rays following nonelastic scattering. (auth)

7770 (LA-DC-7864) NEUTRON CROSS-SECTION MEASUREMENTS OF RADIOACTIVE NUCLIDES USING NEUTRONS FROM AN UNDERGROUND NUCLEAR EXPLOSION. Brown, W. K.; Diven, B. C.; Seeger, P. A. (Los Alamos Scientific Lab., Univ. of California, N. Mex. [1966]. Contract W-7405-eng-36. 18p. (CONF-660817-1). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From International Symposium on Why and How Should We Investigate Nuclides Far Off the Stability Line, Lysekil, Sweden.

The method developed at the Los Alamos Scientific Laboratory of using time-of-flight techniques in combination with a neutron burst from the underground detonation of a nuclear explosive has opened new possibilities in neutron cross-section measurement. The enormously high neutron flux generated in such a burst can be used to give correspondingly large reaction rates in targets. An obvious exploitation of this circumstance lies in the measurement of the cross sections of radioactive nuclides. Such measurements are often impossible using conventional neutron sources due to the high radioactive background, the rapid disappearance of the target, or both. With large reaction rates, the radioactive background becomes relatively small, and the amount of sample does not change significantly in the several milliseconds during the measurement. Having demonstrated the efficacy of the method in general on the Petrel event in June 1965 at the Nevada Test Site, efforts are being directed specifically to measurements of the cross sections of radioactive isotopes. In the autumn of 1966 it is planned to measure the capture cross section of ^{147}Pm (2.7 y), a fission product from which, through neutron capture, the reactor poison ^{148}Pm is synthesized. Plans for future measurements include the capture cross section of ^{148}Pm (41 d), ^{233}Pa (27.4 d), and possibly the fission cross section of ^{235m}U (26.2 m) which would require fast chemistry under field conditions. Measurements on nuclides increasingly further from the stability line are planned. These measurements will provide basic nuclear structure information not otherwise presently obtainable. Measurement of these cross sections will strengthen the knowledge of the systematics among radioactive nuclides and lower the uncertainties in the extrapolation necessary for the analysis of r-process nucleosynthesis. (auth)

38307 EVALUATION OF ^{233}U CROSS SECTIONS. Drake, M. K. (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Trans. Amer. Nucl. Soc., 10: 221-2(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

29833 (NAA-SR-12271) NEUTRON CROSS SECTIONS FOR ^{238}Pu, ^{242}Pu, AND ^{244}Cm. Dunford, C. L.; Alter, H. (Atomics International, Canoga Park, Calif.). May 1, 1967. Contract AT-(04-3)-701. 142p. Dep. CFSTI.

Evaluated neutron cross section data for the nuclides ^{238}Pu, ^{242}Pu, and ^{244}Cm were prepared for ENDF/B. Because of the lack of experimental data, much of the information contained in these libraries is based on theoretical calculations. All experimental data available through December 31, 1966 was included in the evaluation. A complete set of neutron cross section data was prepared for each nuclide for incident neutron energies between 10^{-4} and 1.5×10^{7} eV. These data in the ENDF/B format are available from the Cross Section Evaluation Center at Brookhaven National Laboratory. (auth)

1967

17270 (RPI-328-71) LINEAR ACCELERATOR PROJECT. Annual Technical Report, July 1966–September 1966. Gaertner, Erwin R. (Rensselaer Polytechnic Inst., Troy, N. Y.). Contract AT(30-3)-328. 97p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

The sensitivity of a neutron-capture detector to prompt neutron scattering was investigated. The capture cross sections of Al, Fe, ^{56}Fe, ^{57}Fe, ^{58}Fe, Ni, ^{58}Ni, ^{60}Ni, ^{61}Ni, ^{64}Ni, and Na were measured at energies from about 10 keV to a few hundred keV. The total neutron cross sections of C, Ta, W, ^{182}W, ^{184}W, and ^{186}W were measured from 0.4 to 20 MeV. Gamma spectra from neutron capture by ^{198}Hg, ^{199}Hg, ^{182}W, and ^{183}W were analyzed; transition strengths were obtained. The average neutron number/fission of ^{235}U was measured over the neutron energy range from 2.5 to 9.0 eV, and the scattering cross sections of ^{239}Pu, ^{233}U, and ^{235}U were measured for neutron energies between 1 and 100 eV. Preliminary transmission measurements of the neutron total cross sections of ^{147}Pm and ^{148}Pm were made over the energy range from 1 eV to 2 keV; a graphical area analysis was performed to obtain parameters for the 15.6 eV ^{147}Pm resonance. Time-dependent neutron spectra were measured in light water, and a computer program was written to analyze the data. A facility for studying low-energy neutron inelastic scattering was investigated, and measurements were made for scattering by polyethylene, NbH, ZrH, and UO_2. Codes were also written to automate data reduction. Transient radiation effects in silicon were investigated using a pulsed 50-MeV electron beam. (D.C.W.)

11982 (COO-1573-6) NEUTRON TOTAL CROSS SECTION MEASUREMENTS USING A "WHITE" NEUTRON SOURCE. Galloway III, L. A.; Shrader, E. F. (Case Inst. of Tech., Cleveland, Ohio. Dept. of Physics). Sept. 1966. Contract AT-(11-1)-1573. 131p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

A method for measuring neutron total cross sections using a neutron intensity spectrum continuous in energy (a "white" spectrum) and a pulsed beam time-of-flight technique is used to measure neutron total cross sections in the 2 to 10 MeV region. Total cross sections for the elements Mg, Al, Ca, V, Fe, Pd, Ag and Pb were measured to 1% average uncertainty in steps of 0.08 ns/m. Energy resolution varied from about 1.5% at 2 MeV to 3% at 10 MeV. Results of these measurements are compared with measurements on the same samples with neutrons of known energy and with measurements of other workers. (auth)

2653 (BNL-325(2nd Ed.)(Suppl.2)(Vol.2A)) NEUTRON CROSS SECTIONS. VOLUME IIA. Z = 21 TO 40. Goldberg, Murrey D.; Mughabghab, Said F.; Magurno, Benjamin A.; May, Victoria M. (Brookhaven National Lab., Upton, N. Y.). Feb. 1966. Contract AT(30-2)-Gen-16. 393p. Dep. mn. CFSTI $4.00 cy, $1.75 mn.

An updated compilation of thermal cross sections, resonance parameters, and cross-section curves is presented for elements and isotopes with Z = 21 to 40. The reference sources are also included. (D.C.W.)

11919 (BNL-325(2nd Ed.)(Suppl.2)(Vol.2B)) NEUTRON CROSS SECTIONS. VOLUME IIB. Z = 41 TO 60. Goldberg, Murrey D.; Mughabghab, Said F.; Purohit, Surendra N.; Magurno, Benjamin A.; May, Victoria M. (Brookhaven National Lab., Upton, N. Y.). May 1966. Contract AT(30-2)-Gen-16. 418p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

A compilation of neutron cross sections and resonance parameters is presented for nuclei with Z = 41 to 60 for neutron energies between 0 and 200 MeV. The energy dependence of the cross sections is stressed. (D.C.W.)

13783 (GEAP-5272) EVALUATION AND COMPILATION OF ^{239}Pu CROSS-SECTION DATA FOR THE ENDF/B FILES. Greebler, P.; Aline, P. G.; Hutchins, B. A. (General Electric Co., Sunnyvale, Calif. Advanced Products Operation). Dec. 1966. Contract AT(04-3)-189. 70p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

An extensive evaluation of ^{239}Pu neutron cross-section data was made, and a set of recommended values was compiled for the Evaluated Nuclear Data Files (ENDF/B) being set up at Brookhaven National Laboratory. This work was done as part of the cooperative effort by the Cross Section Evaluation Working Group to put together an initial ENDF/B cross-section file that will include most of the materials that are important for reactor analysis. The recommended ^{239}Pu cross-section data extend in energy from thermal to 15 MeV. Recent experimental work, such as the Petrel bomb test data, and previously unreported theoretical work that provides a firm basis for the selection of resonance parameters in the unresolved resonance region are taken into account. Data sources are summarized and are followed by a complete documentation of the evaluation analysis and a listing of the selected data. (auth)

1077 (LA-DC-7813) NEW TIME-OF-FLIGHT MEASUREMENTS MADE WITH AN INTENSE SOURCE. Hemmendinger, A. (Los Alamos Scientific Lab., Univ. of California, N. Mex.). [1966]. Contract W-7405-eng-36. 51p. (CONF-661014-5). Dep. mn. CFSTI $3.00 cy, $0.50 mn.

From IAEA Conference on Nuclear Data, Paris, France.

In an experiment in Nevada in June 1965 a nuclear device with a yield equivalent of 1.2 kton of TNT provided the neutron source for time-of-flight measurements over a path of 185 m in vacuo. To exploit the combination of high flux and high energy resolution, new recording techniques were required. Because more than a million data points are acquired in any one exposure of a set of targets, the general problems of data retrieval and processing required special attention. Measurements of fission cross sections of the nuclides ^{233}U, ^{235}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am, and ^{242}Am are reported. In addition, capture-to-fission ratios of ^{233}U and ^{240}Pu are reported. The neutron energy range is 10 eV to 2 MeV. Individual resonances are resolved in the 100-eV range. Fission data in the resonance region are characterized by lower minima than are reported by most earlier investigators, indicating more favorable signal-to-background ratios. A unique feature of these experiments is the high rate of data acquisition, which allows cross-section measurements on short-lived nuclides. Even for the long-lived nuclides, these experiments provide an abundance of data required in current nuclear technology—data that could otherwise be acquired only by years of tedious measurement. (auth)

4274 (GEMP-448) EVALUATION AND COMPILATION OF ^{181}Ta, ^{182}W, ^{183}W, ^{184}W, AND ^{186}W CROSS SECTION DATA FOR THE ENDF/B FILE. Henderson, W. B.; DeCorrevont, P. A.; Zwick, J. W. (General Electric Co., Cincinnati, Ohio. Nuclear Materials and Propulsion Operation). Nov. 11, 1966. Contract AT(40-1)-2847. 91p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Evaluated neutron cross section data are given for ^{181}Ta, ^{182}W, ^{183}W, ^{184}W, and ^{186}W. The resolved resonance range is specified to extend upward from 10^{-4} eV to the point at half the level spacing above the last resolved resonance. The lower limit was picked to ensure that at room temperature, and above, errors of the order of 1% or less would be incurred by neglecting reactions below the cutoff. The single-level Breit-Wigner formula for s-wave neutrons was used in a code to compute point value and integral cross sections. No Doppler broadening or solid-state effects were included. No attempt was made to compensate for inadequacy of the single-level formula by smooth cross sections. A single negative energy level was used in each isotope to improve the fit to measured absorption and total cross sections in the thermal range. The values assigned are regarded as preliminary. (S.F.L.)

21410 NEUTRON CROSS SECTIONS OF HYDROGEN IN THE ENERGY RANGE 0.0001 eV TO 20 MeV. Horsley, A. (Atomic Weapons Research Establishment, Aldermaston, Eng.). Nucl. Data, Sect. A, 2: 243-62(Sept. 1966).

Theoretical and experimental data for the neutron cross sections of ^{1}H are surveyed and values recommended for total and partial cross sections in the energy range from 0.0001 eV to 20 MeV. Available experimental data are supplemented where necessary by estimates based on nuclear theory. Details of energy and angular dependence are given so that the data are complete for the purposes of neutronics calculations. (85 references.) (auth)

11968 (GEMP-470) BERYLLIUM NINETEEN GROUP CROSS SECTIONS. Kamphouse, J. L. (General Electric Co., Cincinnati, Ohio. Nuclear Materials and Propulsion Operation). Mar. 20, 1964. Contract AT(40-1)-2847. 33p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

The nineteen-group beryllium cross sections have been revised using the most recent data. The entire problem of the treatment of beryllium is being investigated, and the status of that work is given. Special emphasis is given to the treatment of the (n,2n) reaction. (auth)

17353 (KFK-455) CAPTURE CROSS-SECTION MEASUREMENTS FOR SOME MEDIUM AND HEAVY WEIGHT NUCLEI USING A LARGE LIQUID SCINTILLATOR. Kompe, D. (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Angewandte Kernphysik). Oct. 1966. 12p. (CONF-661014-44). Dep. mn.

From IAEA Conference on Nuclear Data, Paris, France.

Neutron capture cross sections for the nuclei Ag, Au, Cd, Cs, Hf, In, Mo, Nb, Pd, Re, Ta, and W were measured in the energy range from 10 to 150 keV using a pulsed Van de Graaff generator. A large liquid scintillator was used to detect capture events in the samples. All measurements were based on the capture cross section of Au as a standard. (S.F.L.)

11916 (ANL-6172) TABLES OF DIFFERENTIAL CROSS SECTIONS FOR SCATTERING OF NEUTRONS FROM VARIOUS NUCLEI. Lane, R. O.; Langsdorf, A. S. Jr.; Monahan, J. E.; Elwyn, A. J. (Argonne National Lab., Ill.). [ndl. Contract W-31-109-eng-38. 226p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Experimental values of the neutron differential scattering cross sections of ^{9}Be, Ca, ^{12}C, Pb, Li, ^{6}Li, ^{7}Li, ^{16}O, Si, S, and Sn are tabulated, together with values obtained by a least-squares fitting of the data, and the error matrix resulting from the least squares calculation. (D.C.W.)

10004 (LA-3586) FISSION CROSS SECTIONS FROM PETREL. (Los Alamos Scientific Lab., N. Mex.). Sept. 19, 1966. Contract W-7405-eng-36. 231p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Fission data obtained in the Petrel event of June 11, 1965, are presented graphically and in tables, including calculated standard deviations. The isotopes studied were ^{233}U, ^{235}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am, and ^{242}Am; the neutron energy range covered for most isotopes was 20 eV to 1 MeV. (auth)

31810 ^{252}Cf FISSION NEUTRON SPECTRUM FROM 0.003 TO 15.0 MeV. Meadows, J. W. (Argonne National Lab., Ill.). Phys. Rev., 157: 1076-82(May 20, 1967).

The spontaneous-fission neutron spectrum of ^{252}Cf was measured from 0.003 to 15.0 MeV by time-of-flight techniques. A hydrogenous liquid scintillator was used as a detector at the higher energies, while a ^{6}Li-loaded glass scintillator was used at the lower energies. The measured spectrum has an average energy of 2.348 MeV. A Maxwellian distribution, $N(E) \sim E^{1/2} \exp(-E/1.565)$, fits the data well for $0.5 < E < 10.0$ MeV. Below 0.1 MeV, N(E) has a $\sqrt{E}$ dependence but with values ~25% larger than those predicted by the extended Maxwellian spectrum. The results are interpreted in terms of a simplified evaporation model. (auth)

9995 (AEEW-M-504) NEUTRON CROSS-SECTIONS FOR NATURAL ZIRCONIUM IN THE ENERGY RANGE 0.0001 eV TO 6 keV. Pope, A. L. (Atomic Energy Establishment, Winfrith (England)). Sept. 1966. 26p. Dep. mn.

The available resolved resonance data for the isotopes of zirconium are evaluated, with the view to compile a file of basic tabulated neutron cross-section data. These data are also compared with integrally measured data; and from the discrepancies it is concluded that there is a need for better measurements of the radiation widths of the resonances for all zirconium isotopes, especially for the s-wave resonances of ^{91}Zr; that the measured values of the thermal neutron capture cross sections of the isotopes are not consistent with the data for natural zirconium; and that there are still considerable uncertainties in the resonance absorption integrals both for natural zirconium and the separated isotopes. (auth)

1967

23441 (KFK-120) NEUTRON CROSS SECTIONS FOR FAST REACTOR MATERIALS. PART I. EVALUATION. Schmidt, J. J. (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Neutronenphysik und Reaktortechnik). Feb. 1966. 1317p. Dep. mn.

A compilation of neutron cross sections and resonance parameters for C, Cr, He, H, Fe, Mo, Ni, O, ^{239}Pu, Na, ^{235}U, and ^{238}U is presented. The techniques that were used in the analyses of the data are described. Neutron energies in the range from 0.01 eV to 10 MeV are covered. (D.C.W.)

23487 FISSION CROSS SECTIONS OF ^{241}Am AND ^{242m}Am. Seeger, P. A.; Hemmendinger, A.; Diven, B. C. (Los Alamos Scientific Lab., N. Mex.). Contract W-7405-eng-36). Nucl. Phys., A96: 605-16(1967). (LA-DC-7624).

The acquisition and analysis of neutron cross section data from an experiment using an underground nuclear detonation are discussed with specific reference to fission cross sections measured in the Petrel event in June 1965. Results are presented for ^{242}Am and ^{242m}Am over the energy range 20 eV to 1 MeV, measured simultaneously in a single experiment covering the entire energy range, with very low background. Considerable sub-threshold fission was observed for ^{241}Am. The fission cross section of the doubly odd nucleide ^{242m}Am is about twice that of ^{239}Pu over most of the neutron energy range, but only about 20% greater at 1 MeV. (auth)

12027 (UCRL-50001-66-2) PHYSICS DEPARTMENT REPORT, JUNE-SEPTEMBER 1966. (California Univ., Livermore. Lawrence Radiation Lab.). Contract W-7405-eng-48. 63p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Theoretical studies were made of N-N scattering, electron scattering by polyatomic molecules, A-N interaction, $_{\Lambda\Lambda}He^6$, the four-nucleon system, Σ-N interaction, and π-π scattering amplitudes. The fission cross sections of ^{242m}Am and ^{232}U were measured from 0.02 eV to 6 MeV and from 0.004 to 2 keV, respectively. Threshold photoneutron cross sections for Fe and Be were also measured, as were the photoneutron cross sections of ^{89}Y, ^{90}Zr, ^{91}Zr, ^{92}Zr, and ^{94}Zr up to photon energies of 30 MeV. Cross sections for the reaction $^{14}N(n,p_0)^{14}C$ were obtained from measurements of the cross section of the reaction $^{14}C(p,n_0)^{14}N$. The reaction $^{14}C(p,n)^{14}N$ was investigated at proton energies between 7 and 14 MeV; the angular distributions at the highest energies were analyzed using a finite-range DWBA formalism. Angular distributions for the reaction $^{19}F(\alpha,t)^{20}Ne$ were analyzed using a zero-range DWBA formalism. Differential cross sections for (α,n) reactions on ^{17}O and ^{18}O were measured at 9.8, 11.6, and 12.2 MeV; and evidence was also obtained for the $(n,\gamma f)$ reaction on ^{239}Pu for slow neutrons. The neutron scattering cross sections of ^{241}Pu and ^{233}U from 2 to 32 eV and from 2 to 22 eV, respectively, were studied further. Data were also obtained on the deexcitation γ rays following (γ,n) and (γ,p) reactions on ^{16}O; and the spontaneous fission half life of ^{242m}Am was measured, as were the thermal-neutron capture cross sections of Ca, ^{42}Ca, ^{43}Ca, and ^{44}Ca. Neutron diffraction studies of the magnetic transition in NiS as a function of composition were carried out. Bounds on the fugacity and virial series of the pressure in matter were obtained, and the magnetization and conductivity of Fe were studied from 300 to 1250 kbar. Equation-of-state measurements were carried out on rare-earth metals, and elastoplastic wave structure generated in Al by a tangentially accelerated flying plate was studied. A diffusion approximation to the inertial energy transfer in isotropic turbulence was developed; and atmospheric focusing and refraction of blast waves were studied, as was plasma production using multiple laser beams. The effects of various parameters on the output energy from a Q-spoiled ruby oscillator were also studied, and the rate of energy transfer between electrons and ions in a plasma was calculated. (D.C.W.)

25399 (ANL-Trans-168) BULLETIN OF THE INFORMATION CENTER ON NUCLEAR DATA (FIRST ISSUE). Translated by Elmar K. Wilip (Argonne National Lab., Ill.), from Buylleten Informatsionnogo Tsentra po Yadernym Dannym, Atomizdat, Moscow, 1964. 442p. Dep. CFSTI.

Parameters of elementary interactions of neutrons with nuclei are presented, together with reactor constants. The calculation of neutron cross sections by the optical model, using computers is discussed, as is data processing for single-crystal fast-neutron scintillation spectrometry. (D.C.W.)

16002 (GA-8133) NEUTRON CROSS SECTIONS FOR NIOBIUM. Allen, M. S.; Drake, M. K. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Aug. 2, 1967. Contract AT(04-3)-167. 71p. Dep. CFSTI.

The neutron cross sections for Nb that have been prepared for the Evaluated Nuclear Data File (ENDF/B) as part of the cooperative effort by the Cross Section Evaluation Working Group are described. The cross sections were prepared from sets of previously evaluated data and from data that were obtained in an attempt to complete the existing data. (D.C.W.)

18230 RESONANCE ANALYSIS OF THE ^{233}U FISSION CROSS SECTION. Bergen, D. W.; Silbert, M. G. (Los Alamos Scientific Lab., N. Mex.). Phys. Rev., 166: 1178-89(Feb. 20, 1968). (LA-DC-8946).

The neutron-induced fission and capture sections of ^{233}U were measured by time of flight with a nuclear detonation as the neutron source. Cross-section data are presented from 20 to 10^{6} eV for fission and from 30 to 63 eV for the capture-to-fission ratio α. Data in the resonance region (20 to 63 eV) were fitted both by a single-level function consisting of a sum of Briet-Wigner levels and by the Reich-More multilevel function based on R-matrix theory. The resulting resonance parameters are listed and discussed. A study of cross sections derived from two and three hypothetical resonances under various conditions of interference is presented to determine the validity of the resonance parameters derived from the multilevel fit. (auth)

39652 THE ^{233}U FISSION AND CAPTURE CROSS SECTIONS AND THEIR ANALYSIS AT LOW ENERGIES. Bergen, Delmar Wesley. Albuquerque, N. Mex., Univ. of New Mexico, 1967. 76p.

Thesis.

The ^{233}U fission and capture cross sections were measured using a nuclear-device neutron source and time-of-flight techniques. Cross-section data are presented from 20 to 10^{6} eV for fission and from 20 to 63 eV for fission + capture. The resonance region (20 to 63 eV) was fitted with both a single-level function consisting of a sum of Breit-Wigner levels and the Reich and Moore multilevel function based on R-matrix theory. The resulting resonance parameters are listed and discussed. In order to establish the validity of the resonance parameters derived from the multilevel fit, a study is presented of the cross section derived from two and three hypothetical resonances under various conditions and of the cross sections obtained from randomly generated resonances. (Diss. Abstr.)

18226 NEUTRON-INDUCED FISSION CROSS SECTION OF ^{242m}Am. Bowman, C. D.; Auchampaugh, G. F.; Fultz, S. C.; Hoff, R. W. (Univ. of California, Livermore). Phys. Rev., 166: 1219-26(Feb. 20, 1968).

The neutron-induced fission cross section of ^{242m}Am was measured from 0.02 eV to 6 MeV by the time-of-flight method at the Livermore 30-MeV linear electron accelerator. The data are normalized at 0.0253 eV to a value of 6600 b measured in a reactor thermal-neutron flux. The cross section at 0.2 eV is 4700 b; at 1 eV it is 540 b; and at 4 MeV it is 2.1 b. The data are analyzed to obtain values for the neutron strength function $\langle\Gamma_n^0/D\rangle$ of 1.4×10^{-4}, the level spacing D = 1.2 eV, and the quantity $2\pi\langle\Gamma_f/D\rangle = 2.5$. All three quantities are quoted per spin state. The high cross section at low energies can be attributed to the unusually high value for $2\pi\langle\Gamma_f/D\rangle$, and to the existence of a very large resonance at 0.173 eV. The fission cross section of ^{241}Am also was measured in the MeV region and found to be 4.96 ± 0.2 b at 2.5 MeV. (auth)

24873 SELECTED FISSION CROSS SECTIONS FOR ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{237}Np, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, AND ^{242}Pu. Davey, William G. (Argonne National Lab., Idaho Falls, Idaho). Nucl. Sci. Eng., 32: 35-45(Apr. 1968).

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 published up to July, 1965, were analyzed previously to select best fission cross sections for fast-reactor analysis. Since the completion of that work, new data have been produced which necessitate reevaluation of the fission cross sections particularly in the region 1 to 5 MeV. The revised data presented here are believed to be of greater consistency and, hence, accuracy than the previous selection. (auth)

37289 (AERE-R-5224) THE AVERAGE NEUTRON TOTAL CROSS-SECTION OF ^{10}B FROM 100 eV TO 10 MeV AND ABSORPTION CROSS-SECTION UP TO 500 keV. Diment, K. M. (Atomic Energy Research Establishment, Harwell (England)). Feb. 1967. 17p. Dep. CFSTI. UK.

The transmission measurements to be described were made on the 120-m and 300-m flight paths of the Electron Linac time-of-flight spectrometer at Harwell. Analysis of the total cross section below 10 keV shows $\sigma_{n,T} = (610.3 \pm 3.1)\ E^{-1/2} + (1.95 \pm 0.10)$ barns. The deviation of $\sigma_{n,T}$ from this value above 40 keV is due to an increase in elastic scattering cross section. Subtraction of the latest scattering cross sections measured by Mooring et al. yields an absorption cross section that is proportional to $E^{-1/2}$ up to at least 250 keV. (auth)

16001 (GA-7462) NEUTRON CROSS SECTIONS FOR ^{231}Pa, ^{233}Pa, AND ^{232}U. Drake, M. K.; Nichols, P. F. (Gulf General Atomic, Inc., San Diego, Calif.). Sept. 8, 1967. Contract AT(04-3)-167. 83p. Dep. CFSTI.

A survey was made of the available experimental cross-section measurements for ^{231}Pa, ^{233}Pa, and ^{232}U. Sets of recommended neutron cross sections and resonance parameters are presented for neutron energies from 0.001 eV to 15 MeV. (D.C.W.)

16003 (GA-8135) NEUTRON CROSS SECTIONS FOR ^{234}U AND ^{236}U. Drake, M. K.; Nichols, P. F. (Gulf General Atomic, Inc., San Diego, Calif.). Sept. 1, 1967. Contract AT(04-3)-167. 76p. Dep. CFSTI.

A survey was made of the available experimental cross-section measurements for ^{234}U and ^{236}U. Sets of recommended neutron cross sections are presented for neutron energies from 0.001 eV to 15 MeV. Resonance parameters are also included. (D.C.W.)

37310 (AD-665363) NEUTRON AND GAMMA RAY PRODUCTION CROSS SECTIONS FOR SODIUM, MAGNESIUM, CHLORINE, POTASSIUM, AND CALCIUM. PART V. POTASSIUM. Drake, M. K. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Nov. 1967. Contract DA-18-035-AMC-730(A). 133p. (GA-7829(Pt.5); NDL-TR-89(Pt.5)). CFSTI.

An investigation was made of the neutron interaction probabilities with the element potassium. Sets of recommended total and partial neutron cross sections were prepared. The energy and angular distributions of the secondary neutrons are given. Also, gamma-ray production cross sections were obtained as well as energy and angular distributions of the secondary gamma rays. In general, the recommended data were based on experimentally measured data. However, where no experimental data were available, the recommended cross sections were obtained using model calculations. (auth) (USGRDR)

13904 (LA-3801) ENDF/B FORMAT REQUIREMENTS FOR SHIELDING APPLICATIONS. Dudziak, Donald J. (comp. and ed.) (Los Alamos Scientific Lab., N. Mex.). Apr. 1967. Contract W-7405-eng-36. 52p. (ENDF-111). Dep. CFSTI.

The present (April 1967) Evaluated Nuclear Data File/B (ENDF/B) format was designed primarily to satisfy the requirements of nuclear reactor core neutronics calculations. Extensions of the format specifications are proposed to include data of interest for shielding calculations for reactor and other applications. Alternate methods of presenting the necessary data are discussed, and the correspondence of ENDF/B to the United Kingdom Atomic Energy Authority Nuclear Data File (UK) is maintained wherever practical. In the case of photon interactions, detailed formats are recommended for cross sections for secondary angular, energy, and energy-angle distributions, and for incoherent and coherent scattering atomic form factors. Format recommendations for photon production data include those for photon angular distributions, photon production multiplicities, and photon energy-angle distributions. A listing of data on photon production in Na is included. (auth)

11936 NEUTRON TOTAL CROSS SECTION MEASUREMENTS USING A "WHITE" NEUTRON SOURCE. Galloway III, Louie Altheimer. Cleveland, Case Inst. of Tech., 1966. 131p. Thesis.

A method for measuring neutron total cross sections using a neutron intensity spectrum continuous in energy (a "white" spectrum) and a pulsed beam time-of-flight technique was used to measure neutron total cross sections in the 2 to 10 MeV region. Total cross sections for the elements Mg, Al, Ca, V, Fe, Pd, Ag and Pb were measured to 1% average uncertainty in steps of 0.08 ns/m. Energy resolution varied from about 1.5% at 2 MeV to 3% at 10 MeV. Results of these measurements are compared with measurements on the same samples with neutrons of known energy and with measurements of other workers. (Disser. Abstr.)

37307 (AD-665360) NEUTRON AND GAMMA RAY PRODUCTION CROSS SECTIONS FOR SODIUM, MAGNESIUM, CHLORINE, POTASSIUM, AND CALCIUM. PART II. SODIUM. Garrison, J. D.; Drake, M. K. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Nov. 1967. Contract DA-18-035-AMC-730(A). 148p. (GA-7829(Pt.2); NDL-TR-89(Pt.2)). CFSTI.

Neutron and gamma-ray production cross sections sets were prepared for the element sodium. These data sets include total and partial neutron cross sections as well as the cross sections for producing deexcitation gamma rays. Information is also given for the angular and energy distribution of the secondary neutron and gamma rays. (auth) (USGRDR)

22584 (LA-DC-9181) ^{238}U NEUTRON CAPTURE RESULTS FROM BOMB SOURCE NEUTRONS. Glass, N. W.; Schelberg, A. D.; Tatro, L. D.; Warren, J. H. (Los Alamos Scientific Lab., N. Mex.). [1967]. Contract W-7405-eng-36. 16p. (CONF-680307-20). Dep. CFSTI.

From 2nd Conference on Neutron Cross Sections and Technology, Washington, D. C.

Results on neutron capture in ^{238}U from 30 to 2050 eV neutron energy are presented. The data were obtained by neutron time-of-flight utilizing the pulse source of neutrons from the Petrel nuclear explosion. The total radiation width, Γ_γ, was determined for 62 l = 0 levels with Γ_γ = [19.1 ± 0.6 (stat.) ± 1.4 (syst.)] × 10^{-3} eV. There appears to be a significant variation in the value of Γ_γ from resonance to resonance. Approximately 200 weak resonances were found which can be ascribed to p-wave levels. Analysis of these weak resonances, assuming l = 1, gives results consistent with: an average reduced neutron width of 3.7 ± 0.7 × 10^{-3} eV; an average level spacing of 7.0 ± 0.5 eV: and a strength function of 1.8 ± 0.3 × 10^{-4}. (auth)

20561 (AHSB(S)R-141) NEUTRON INTERACTION IN THE ENERGY RANGE 1.0 E-10 MeV TO 15.0 MeV. Hart, W. (United Kingdom Atomic Energy Authority, Risley (England). Authority Health and Safety Branch). Mar. 1968. 42p. Dep. CFSTI. UK.

In the past few years, some high-resolution measurements have been performed which, when combined with theoretical predictions, afford a detailed description of neutron interactions with the nuclide tantalum. From these and other sources of information, a set of neutron cross sections and other relevant parameters have been evaluated. They cover the energy range 1 × 10^{-10} to 15 MeV and are presented as a new data file (DFN346) for the UKAEA Nuclear Data Library. (auth)

29838 CURVE FITTING AND STATISTICAL TECHNIQUES FOR USE IN THE MECHANISED EVALUATION OF NEUTRON CROSS SECTIONS. Horsley, A.; Parker, J. B.; Price, J. A. (Atomic Weapons Research Authority, Aldermaston, Eng.). Nucl. Instrum. Methods, 62: 29-42 (June 1, 1968).

A cubic spline curve fitting method and a statistical theory of unknown systematic errors are combined to give a practical computer-orientated method of evaluating neutron cross sections. Particular attention is paid to reconciling sets of discordant data. The input data and evaluated curve can be displayed on a CRT graphical display unit. Among program output examples given are evaluated curves for several cross sections of ^{10}B. (auth)

29894 (LA-3271) EVALUATED NEUTRON CROSS SECTIONS FOR DEUTERIUM. Horsley, Anthony; Stewart, Leona (Los Alamos Scientific Lab., N. Mex.). Nov. 1967. 139p. Dep. CFSTI.

A compilation of evaluated data on the neutron cross sections of deuterium is presented for incident neutron energies of 0.0001 eV to 20 MeV. The data are displayed in graphical and tabular form. Legendre coefficients are included for the scattering angular distributions. Proton spectra from neutron and proton reactions with deuterium are also included, as are proton production cross sections and neutron production cross sections. (D.C.W.)

48446 NEUTRON CROSS SECTIONS OF DEUTERIUM IN THE ENERGY RANGE 0.0001 eV TO 20 MeV. Horsley, A. (Univ. of Birmingham, Eng.). Nucl. Data, Sect. A, 4: 321-57 (July 1968).

Experimental and theoretical data for the neutron cross sections of D are surveyed, and values are adopted for total and partial cross sections. To facilitate neutronics calculations, angular distribution functions based on n + d and the conjugate reaction p + d are given. The experimental data used, with some comments, and a brief account of the evaluation procedure are presented. (auth)

3607 (ORNL-TM-1872) EVALUATION OF NEUTRON CROSS SECTIONS FOR ^{10}B. Irving, D. C. (Oak Ridge National Lab., Tenn.). Oct. 9, 1967. Contract W-7405-eng-26. 45p. Dep. CFSTI.

The neutron cross sections for ^{10}B were evaluated from 10^{-4} eV to 15 MeV. The existing experimental data are reviewed, and theoretical calculations and other reasoning are used to fill in the gaps. A complete and consistent set of cross sections is presented, and an explanation is given for the choices made in developing this cross-section set. (auth)

35627 GROUP CROSS-SECTION SET KFK-SNEAK. PREPARATION AND RESULTS. Kuesters, H.; Bachmann, H.; Huschke, H.; Kiefhaber, E.; Krieg, B.; Metzenroth, M.; Siep, I.; Wagner, K.; Woll, D. (Kernforschungszentrum, Karlsruhe, Ger.). pp 167-88 of Fast Reactor Physics. Vol. I. Vienna, International Atomic Energy Agency, 1968.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.1); CONF-671043-(Vol.1).

For the calculation of integral quantities, measured in the Fast Zero-Power Assembly SNEAK, a new group cross-section set was prepared directly from microscopic data. These data are based on the latest evaluation of cross-section and resonance parameters tabulated on the Karlsruhe Nuclear Data File.KEDAK. Special attention is paid to the calculation of the elastic moderation. (auth)

46695 (EURFNR-537) TABLES OF EVALUATED NEUTRON CROSS SECTIONS FOR FAST REACTOR MATERIALS. Langner, I.; Schmidt, J. J.; Woll, D. (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Neutronenphysik und Reaktortechnik). Jan. 1968. 663p. (KFK-750; EUR-3715; EANDC(E)-88-U).

Work Performed under United States-Euratom Fast Reactor Exchange Program.

Tables of evaluated data, as functions of incident neutron energy, are presented for the neutron cross sections and resonance parameters of materials that are of particular interest in the design of fast and intermediate reactors. The materials covered include C, Cr, He, H, Fe, Mo, Ni, O, ^{239}Pu, Na, ^{235}U, and ^{238}U. (D.C.W.)

35195 (AEEW-M-802) U.K.A.E.A. NUCLEAR DATA LIBRARY, JANUARY 1967. Norton, D. S.; Story, J. S. (Atomic Energy Establishment, Winfrith (England)). Feb. 1968. 19p. Dep. CFSTI. UK 3s. 6d.

Details are given of the UKAEA Nuclear Data Library as of January 1967. The updating of the library included the incorporation of several new data files, together with the extension of ranges of some older files. (auth)

50559 (ANL-7387) COMPILATION OF ENDF/B DATA FOR MAGNESIUM, TITANIUM, VANADIUM, MOLYBDENUM, AND GADOLINIUM Pennington, E. M.; Gajniak, J. C. (Argonne National Lab., Ill.). Mar. 1968. Contract W-31-109-eng-38. 122p. Dep. CFSTI.

The compilation of ENDF/B neutron cross-section data for the materials magnesium, titanium, vanadium, molybdenum, and gadolinium is presented. All the data in the ENDF/B format are listed, and graphs of much of the data are presented. (auth)

22615 AVERAGE CROSS SECTIONS AND RESONANCE INTEGRALS OF ^{242m}Am. Perkins, S. T.; Auchampaugh, G. F.; Hoff, R. W.; Bowman, C. D. (Univ. of California, Livermore). Nucl. Sci. Eng., 32: 131-2(Apr. 1968). (UCRL-70735).

The energy dependence of the neutron fission cross of ^{242m}Am was investigated from 0.02 eV to 6 MeV. The measurements were normalized to a value for the fission cross section of 6600 b at 0.0253 eV. Below 3.7 eV and above 300 ke eV, the cross section was corrected for ^{241}Am in the sample. The data are presented in terms of mean values and resonance integrals over the GAM group structure. The resonance integral above 0.5 eV is 1570 ± 110 b, The data below 3.5 eV were analyzed in terms of a sum-of-single-levels fit. For the six resonances below 3.5 eV, the average fission width is 0.46 eV. (D.C.W.)

46730 (APDA-217) EVALUATED NEUTRON CROSS SECTIONS OF ^{23}Na FOR THE ENDF/B FILE. Pitterle, T. A. (Atomic Power Development Associates, Inc., Detroit, Mich.). June 1968. Contract AT(11-1)-865. 43p. (ENDF-121). Dep. CFSTI.

An evaluation of ^{23}Na neutron cross section data was carried out for the ENDF/B file. Data were evaluated from 10^{-4} eV to 15 MeV for the following neutron reactions: total, elastic scattering including Legendre polynomial expansions of the angular dependence, nonelastic, inelastic including resolved levels, (n,γ), (n,p), (n,α), and (n,2n). Graphs of the evaluated data are compared with experimental data. (auth)

48508 (APDA-218) EVALUATED NEUTRON CROSS SECTIONS OF ^{240}Pu FOR THE ENDF/B FILE. Pitterle, T. A.; Yamamoto, M. (Atomic Power Development Associates, Inc., Detroit, Mich.). June 1968. Contract AT(11-1)-865. 60p. (ENDF-122) Dep. CFSTI.

Data were evaluated from 10^{-4} to 15 MeV for the following neutron reactions: total, n−γ, fission, (n,2n), (n,3n), elastic scattering including Legendre polynomial expansions of the angular dependence, nonelastic, and inelastic scattering including resolved levels. Graphs of the evaluated data are included. (auth)

16012 (KAPL-3327) EVALUATED CROSS SECTIONS FOR THE HAFNIUM ISOTOPES. Reynolds, J. T.; Lubitz, C. R. (Knolls Atomic Power Lab., Schenectady, N. Y.); Itkin, I.; Harris, D. R. (Bettis Atomic Power Lab. [Pittsburgh, Pa.]). Aug. 17, 1967. Contract W-31-109-eng-52. 22p. plus Charts. Dep. CFSTI.

Evaluated libraries of cross sections were prepared for natural hafnium and its isotopes ^{174}Hf, ^{176}Hf, ^{177}Hf, ^{178}Hf, ^{179}Hf, and ^{180}Hf. The libraries contain total, elastic, capture, inelastic, (n,p), and (n,2n) cross sections and elastic scattering Legendre moments below 15 MeV. The most recent experimental data were used in the evaluation; and, whenever data were not available, theoretical calculations were made. (auth)

48416 (KFK-120(Pt.1)) NEUTRON CROSS SECTIONS FOR FAST REACTOR MATERIALS. PART I. EVALUATION. Schmidt, J. J. (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Neutronenphysik und Reaktortechnik). Feb. 1966. 1400p. (EANDC(E)-35-U(Pt.1)).

A comprehensive documentation and critical review of the available experimental and theoretical information on microscopic neutron cross section data are presented, and the procedures that were used to reduce this information into cross section sets in graphical and tabular form are described. The neutron energies considered range from 0.01 eV to 10 MeV, with emphasis on fast and resonance neutrons. The materials covered include C, Cr, He, H, Fe, Mo, Ni, O, ^{239}Pu, Na, ^{235}U, and ^{238}U. (D.C.W.)

24861 (WANL-TME-1721) A SINGLE LEVEL ANALYSIS OF ^{233}U NEUTRON CROSS SECTIONS; AN n + ^{233}U MULTIGROUP CROSS SECTION LIBRARY. Schneider, M. J. (Westinghouse Electric Corp., Pittsburgh, Pa. Astronuclear Lab.). Dec. 1967. Subcontract NP-1. 107p. Dep. CFSTI.

Total, capture, and fission cross sections for n + ^{233}U were simultaneously least squares fit over the energy range 0.4 to 61.4 eV using the single-level Breit-Wigner formula. Sixty-three resonances were found in this range. Good agreement was obtained between integral data and this fit to differential data for the resonance integrals of ^{233}U. Resonance parameters are given, along with brief statistical analyses of them. A multigroup cross-section library from .001 eV to 10 MeV is presented. (auth)

24862 (WANL-TME-1746) CHANGES IN GAM AND TNS CROSS SECTION LIBRARIES: NEW ACTIVATION CROSS SECTIONS. Schneider, M. J. (Westinghouse Electric Corp., Pittsburgh, Pa. Astronuclear Lab.). Jan. 1968. Subcontract NP-1. 47p. Dep. CFSTI.

Some recent experimental data and evaluations of activation cross sections for several nuclei are analyzed; and new GAM, TNS, and BIT libraries based on these analyses are presented. The cross sections for the isotopes ^{55}Mn, ^{115}In, and ^{32}S, and for natural Pb and Zr are analyzed. A preliminary analysis of the ^{93}Nb(n,γ) cross section is also summarized. (D.C.W.)

7689 (BAW-316) ^{238}U NEUTRON CROSS-SECTION DATA FOR THE ENDF/B. Wittkopf, W. A.; Roy, D. H.; Livolsi, A. Z. (Babcock and Wilcox Co., Lynchburg, Va.). May 1967. 104p. Dep. CFSTI.

As part of the cooperative effort of the Cross Section Evaluation Working Group organized at Brookhaven National Laboratory in June 1966, the nuclear data on ^{238}U for use in the Evaluated Nuclear Data File B (ENDF/B) are presented. The data cover the energy range from 0.001 eV to 15 MeV. Data sources are referenced, and the theoretical methods used in evaluating certain data are described. A complete listing of the data in the ENDF/B format is provided. (auth)

22578 (IA-1152) NUCLEAR DATA FOR ^{240}Pu, ^{241}Pu, and ^{242}Pu. Yiftah, S.; Schmidt, J. J.; Caner, M.; Segev, M. (Israel Atomic Energy Commission, Yavne. Soreq Nuclear Research Center). Dec. 1967. 100p. Dep.

An evaluation of the basic nuclear data for ^{240}Pu, ^{241}Pu, and ^{242}Pu in the range 0.01 eV to 15 MeV was made. Partial cross sections are constructed for the thermal and fast neutron ranges; they are presented in graphical and tabular form. Resonance parameters and average parameters are recommended for the resonance region; they are presented in tabular form. The cross sections constructed are: total, nonelastic, elastic scattering, radiative capture, fission, total and partial inelastic scattering, (n,2n), and (n,3n). Other nuclear data considered are: the average parameters above 1 keV, the average number of prompt neutrons per neutron-induced fission, the average scattering cosine in the lab system, and the energy spectrum of secondary neutrons from fission. (auth)

V. NEUTRON CROSS SECTIONS

3. Wide Ranges in Energy

1967

29960 (IN-1026) NEUTRON TOTAL AND ABSORPTION CROSS SECTIONS OF ^{238}Pu. Young, T. E.; Simpson, F. B.; Berreth, J. R.; Coops, M. S. (Idaho Nuclear Corp., Idaho Falls). Mar. 1967. Contract AT(10-1)-1230. 81p. Dep. CFSTI.

The neutron total cross section of ^{238}Pu was measured from 0.008 to 6500 eV. These data give single-level Breit-Wigner parameters for resonances below 200 eV. The observed total cross section at 2200 m/sec is 588 barns. A value of 532 barns was calculated for the effective (equivalent 1/v) thermal absorption cross section. Parameters of individual resonances below 200 eV and average parameters at higher energies give a resonance absorption integral of 164 ± 15 barns, and a value of 1.10 ± 0.20 × 10^{-4} for the s-wave neutron strength function (Γ_n^0/D). (auth)

17341 (AERE-M-1709) ETA AND NEUTRON CROSS SECTIONS OF ^{239}Pu AND ^{233}U. Brooks, F. D.; Jolly, J. E.; Schomberg, M. G.; Sowerby, M. G. (Atomic Energy Research Establishment, Harwell (England)). Sept. 1966. 18p. Dep. mn.

Measured values of η for ^{239}Pu and ^{233}U are given for the energy range 0.04 to 11 eV, together with fission and total cross sections for ^{233}U. Average values of η were calculated and also, in the case of ^{233}U, various ratios and integrals of the cross sections and α for energy groups of interest for reactor design studies. Total cross-section measurements were also made for ^{239}Pu from 10 to 1000 eV. Comparisons are included between the results obtained in the experiment and those from other laboratories. (auth)

25734 CONFIRMATORY EXPERIMENTAL DATA ON THE HARWELL BORON PILE $\bar{\nu}$ VALUES. Colvin, D. W.; Sowerby, M. G.; MacDonald, R. I. (Atomic Energy Research Establishment, Harwell, Eng.). Vienna, International Atomic Energy Agency, 1966, Preprint No. CN-23/33. 16p. (CONF-661014-35). DTIE.

From IAEA Conference on Nuclear Data, Paris.

The $\bar{\nu}$ values obtained by the Harwell boron pile experiment, which have previously been reported (Symposium on the Physics and Chemistry of Fission, Vol. II, p. 24, 1965), are approximately 2% lower than the values obtained with large liquid scintillators and those derived from measured values of η and α. The value of pile efficiency used in these measurements was obtained by using the associated particle technique, i.e., the d(γ,n)p reaction. Two standard neutron sources, a Ra-γ-Be source and the AWRE ^{240}Pu spontaneous fission source, have now been calibrated at the National Physical Laboratory, Teddington, England, and in the boron pile. The count rates of the standard sources in the boron pile can be used to obtain a second independent value of the pile efficiency and hence give information on the correctness of the boron pile $\bar{\nu}$ values. The results of these measurements are given together with other information, which demonstrates that the correction procedures used in the boron pile experiment are valid. (auth)

46722 (NAA-SR-12271(Suppl.1)) NEUTRON CROSS SECTIONS FOR ^{238}Pu, ^{242}Pu, AND ^{244}Cm. Supplement I. Dunford, C. L.; Alter, H. (Atomics International, Canoga Park, Calif.). Sept. 15, 1967. Contract AT(04-3)-701. 5p. Dep. CFSTI.

Some corrections to the cross-section data for ^{244}Cm, ^{238}Pu, and ^{242}Pu that were presented in NAA-SR-12271 are summarized. (D.C.W.)

17373 SUBBARRIER FISSION OF ^{232}Th BY NEUTRONS. Ermagambetov, S. B.; Kuznetsov, V. F.; Smirenkin, G. N. Yadern. Fiz., 5: 257-63(Feb. 1967). (In Russian).

The dependence of the reaction cross section on neutron energy was measured in the region 0.6 to 3.0 MeV for the investigation of ^{232}Th(n,f) fission near the threshold. Some characteristics of the potential barrier in the fission are discussed, in connection with the results of the experiment. The competition of inelastic neutron scattering to levels of the ^{232}Th target in the energy region 0.75 to 1.0 MeV shows up clearly in the energy dependence of the fission cross section. The disagreement between the known thermal neutron fission cross section 0.06 ± 0.02 mb and the value extrapolated from σ_f at higher energies is discussed. (auth)

1967

29970 (WANL-TME-1586) A SINGLE LEVEL ANALYSIS OF ^{235}U BASED ON RECENT σ_t, σ_f, AND σ_c MEASUREMENTS; A n + ^{235}U MULTIGROUP CROSS SECTION LIBRARY. Gibson, Gordon (Westinghouse Electric Corp., Pittsburgh, Pa. Astronuclear Lab.). Mar. 1967. Contract SNP-1. 143p. Dep. CFSTI.

Total, capture, and fission neutron cross-section data for ^{235}U were simultaneously fit with a single-level Breit–Wigner least squares analysis. Two sets of resonance parameters for E < 63 eV are presented. One set was obtained when unit weighting was used in the least-squares analysis, and another set was obtained when weighting proportional to the inverse of the cross section was used. The resonance capture and fission integrals calculated from the parameters agree in each case to within three percent with corresponding direct numerical integration of the data. The effects of simultaneously fitting two cross sections at a time: total and fission, and capture and fission were also studied. Two n + ^{235}U multigroup cross section libraries, that differ only in the resonance range where the two sets of resonance parameters described above are used, were generated for use in reactor calculations. The ratio of the capture resonance integral to the fission resonance integral that is calculated ($\bar{\alpha}$ = 0.51) with either library agrees with the reactor integral measurements ($\bar{\alpha}$ = 0.50 ± 0.02). These libraries are presented. (auth)

17350 (KFK-450) THE FISSION CROSS-SECTIONS OF SOME PLUTONIUM ISOTOPES IN THE NEUTRON ENERGY RANGE 5 TO 150 keV. Gilboy, W. B.; Knoll, G. (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Angewandte Kernphysik). Oct. 1966. 12p. (CONF-661014-43). Dep. mn.

From IAEA Conference on Nuclear Data, Paris, France.

The neutron fission cross sections of ^{239}Pu and ^{240}Pu at 5 to 150 keV were obtained by measuring the ^{239}Pu/^{235}U and ^{240}Pu/^{235}U fission cross-section ratios. (D.C.W.)

42658 THE ENERGY DEPENDENCE IN THE RANGE 0.7 TO 4 MeV OF THE NEUTRON FISSION CROSS SECTION OF ^{239}Pu. Grama, N.; Marinescu, L.; Mihai, I.; Petrascu, M.; Sandulescu, A.; Voiculescu, G. (Inst. of Atomic Physics, Bucharest). Rev. Roum. Phys., 12: 43-51(1967).

The energy dependence of the fission cross section of ^{239}Pu in the energy range 0.7 to 4 MeV was measured. The experimental points are compared with values calculated on the basis of many-level interference theory. (auth)

25517 (RFR-191) FAST ELASTIC AND INELASTIC CROSS SECTIONS FOR ^{238}U. Haggblom, H. (Aktiebolaget Atomenergi, Stockholm (Sweden)). May 3, 1962. 28p. Dep.

Experimental and semi-empirical scattering cross sections for ^{238}U are compared with theoretical values in the energy range 0.08 to 6 MeV. The theoretical cross sections are obtained by optical-model calculations including spin-orbit interaction and by Hauser–Feshbach theory. Transport cross sections are calculated both from the experimental and the theoretical differential cross sections. Calculations of inelastic 11-group cross sections are performed using the continuum model for the compound nucleus in the region above 1.4 MeV and assuming five excitation levels below 1 MeV. Between 1.0 and 1.4 MeV, the cross-sections are obtained by taking a mean value between the results from the continuous and the discrete level theory. (auth)

35381 (ANL-7320, pp 16-21) FISSION CROSS-SECTION MEASUREMENTS OF ^{233}U, ^{235}U, ^{239}Pu, AND ^{241}Pu IN THE ENERGY RANGE FROM 1 TO 25 keV. James, G. D. (Atomic Energy Research Establishment, Harwell (England)).

Fission cross-section measurements have been carried out, by time-of-flight experiment, over the energy range from 1 eV to 25 keV for the nuclides ^{233}U, ^{235}U, ^{239}Pu, and ^{241}Pu. The data obtained from 1 keV to 25 keV are presented and compared with existing data. It is shown that in this energy range the available data for ^{235}U are in good agreement and that the data for ^{233}U and ^{239}Pu agree fairly well. A list of 19 references is included. (auth)

33953 (AAEC/E-168) A STUDY OF THE EFFECTIVE RESONANCE INTEGRAL AND DOPPLER COEFFICIENT OF ^{238}U, ^{232}Th, AND ^{240}Pu USING THE CODE COMPLEX LUBRA. Keane, A.; Kletzmayr, E. (Australian Atomic Energy Commission Research Establishment, Lucas Heights). Dec. 1966. 34p. Dep.

The effective resonance integral and Doppler coefficient of ^{238}U, ^{232}Th, and ^{240}Pu were studied in detail using the LUBRA complex of codes. Some earlier results were used for comparison to verify the validity of the LUBRA results. Discrepancies were explained, and confidence can be placed in the results given by the LUBRA codes. (auth)

40432 (WASH-1074) REPORTS TO THE AEC NUCLEAR CROSS SECTIONS ADVISORY GROUP MEETING, HELD AT BROOKHAVEN, NEW YORK, APRIL 13–14, 1967. Motz, H. T. (comp.) (European-American Nuclear Data Committee). Contract W-7405-eng-36. 153p. (EANDC(US)-99U; INDC(US)-9U). Dep. CFSTI.

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URANIUM-238— neutron capture at 0.04 to 1 90 keV, radiative widths for

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URANIUM-238— neutron fission cross section at 1 to 23 MeV
URANIUM-235— neutron fission cross section at 2 to 21 MeV

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25504 (EANDC(E)-74(U)) NEUTRON CROSS SECTIONS AND FISSION PARAMETERS OF ^{240}Pu, ^{241}Pu, AND ^{242}Pu. Prosdocimi, A. (European Atomic Energy Community, Geel (Belgium). Central Nuclear Measurements Bureau. European-American Nuclear Data Committee). Aug. 25, 1960. 18p. Dep.

A compilation of neutron resonance integrals and parameters, values for the numbers of prompt neutrons emitted in neutron and spontaneous fission, and neutron fission and total cross sections is presented for ^{240}Pu, ^{241}Pu, and ^{242}Pu. (D.C.W.)

9996 (IA-1094) BASIC NUCLEAR DATA FOR THE HIGH PLUTONIUM ISOTOPES. PART I. CROSS SECTIONS. Segev, M.; Caner, M.; Yiftah, S. (Israel Atomic Energy Commission, Yavne. Soreq Nuclear Research Center). June 1966. 27p. Dep. mn.

Energy-dependent cross sections of ^{240}Pu, ^{241}Pu, and ^{242}Pu, in the range from 1 keV to 15 MeV, are constructed from the scant available experimental data by interpolation, extrapolation, general theoretical arguments and systematics. The constructed cross sections are presented in graphical form in 14 figures. (auth)

1967

21525 TOTAL NEUTRON CROSS SECTION OF ^{233}Pa. Simpson, F. B.; Codding, J. W. Jr. (Phillips Petroleum Co., Idaho Falls, Idaho). Nucl. Sci. Eng., 28: 133-8(Apr. 1967).

Transmission measurements on ^{233}Pa were taken with the Materials Testing Reactor (MTR) fast chopper. The total cross section was calculated in the energy range from 0.01 to 10,000 eV. These measurements were made on 700 mg of chemically separated ^{233}Pa in an oxide form. The protactinium was produced by irradiating 280 g of ^{232}Th in the Engineering Test Reactor. The sample represented approximately 15,000 Ci of activity. The data were taken with a resolution of 0.08 to 2.0 μsec/m. The Breit-Wigner resonance parameters were obtained for the resonances below 18 eV. The average parameters give a value of 0.75×10^{-4} for the s-wave neutron strength function $\bar{\Gamma}_n^0/D$. Weighting the level spacings inversely as 2J + 1 gives the average observed level spacings per spin state of 1.10 and 1.84 eV. A second-order polynomial least-squares fit to the $\sigma_T\sqrt{E}$ data between 0.01 and 0.10 eV gives a 2200 m/sec total neutron cross section of 55 ± 3 b, superseding a value of 57 b given previously. The resonance-absorption integral for neutrons with energies above 0.4 eV was calculated to be 901 ± 45 b. (auth)

12014 (IN-1015) TABULATION OF THE TOTAL NEUTRON CROSS SECTION OF ^{232}U. Simpson, O. D.; Moore, M. S.; Berreth, J. R. (Idaho Nuclear Corp., Idaho Falls). Dec. 1966. Contract AT(10-1)-1230. 69p. Dep. mn. CFSTI \$3.00 cy, \$0.65 mn.

The total neutron cross section of ^{232}U was measured from 0.01 to 10,000 eV using the Materials Testing Reactor (MTR) fast chopper. A 2200 meter/sec total neutron cross section of 165 ± 10 barns was determined. Multilevel parameters are listed for resonances below 30 eV. Results of the analysis indicate that two fission channels are needed to describe the experimental data. A tabulation of the data is given. (auth)

42621 TOTAL NEUTRON CROSS SECTION OF ^{232}U. Simpson, O. D.; Moore, M. S.; Berreth, J. R. (Idaho Nuclear Corp., Idaho Falls). Nucl. Sci. Eng., 29: 415-22(Sept. 1967).

The neutron total cross section of ^{232}U was measured from 0.01 to 10,000 eV using the Materials Testing Reactor (MTR) fast chopper. A 2200 m/sec total neutron cross section of 163 ± 10 b was determined. Multilevel parameters are listed for resonances below 30 eV. Results of the analysis indicate that two fission channels are needed to describe the experimental data. (auth)

35457 (AEET-272) SELF-SHIELDED CROSS SECTIONS FOR THE MAIN FERTILE AND FISSILE NUCLEI. Singh, R. Shankar; Desai, G. A. (Atomic Energy Establishment, Trombay (India)). 1966. 18p. Dep.

Self-shielded cross-sections for ^{232}Th, ^{235}U, ^{238}U, and ^{239}Pu which exhibit resonance behavior in their reaction cross-sections with neutrons are necessary to represent the proper effective values in a multigroup analysis of reactors and to predict accurately the reactivity coefficients due to the Doppler effect, etc. These were evaluated from resonance-integral calculations under the narrow-resonance approximation using the latest available resonance parameters at four temperatures (300, 750, 1500, and 2500°K) and at σ_p (potential scattering cross section per absorber atom) values of 40 and 60 barns for ^{232}Th and ^{238}U and 126, 200, 300, and 400 barns for ^{235}U and ^{239}Pu. The status of resonance parameters for these elements is also discussed in detail. (auth)

40457 CAPTURE CROSS SECTIONS OF ^{237}Np. Stupegia, Donald C.; Schmidt, Marcia; Keedy, Curtis R. (Argonne National Lab., Ill.). Nucl. Sci. Eng., 29: 218-19(Aug. 1967).

Neutron-capture cross sections of ^{237}Np were measured at eight neutron energies between 0.15 and 1.5 MeV. The experimental method was the activation technique in which the neptunium target was irradiated with a monoenergetic neutron beam and was analyzed for the product ^{238}Np by gamma-ray spectrometry. (auth)

42636 THE FISSION CROSS SECTIONS OF ^{233}U, ^{234}U, ^{236}U, ^{238}U, ^{237}Np, ^{239}Pu, ^{240}Pu, AND ^{241}Pu RELATIVE TO THAT OF ^{235}U FOR NEUTRONS IN THE ENERGY RANGE 1 TO 14 MeV. White, P. H.; Warner, G. P. (Atomic Weapons Research Establishment, Aldermaston, Eng.). J. Nucl. Energy, 21: 671-9(Aug. 1967).

The fission cross sections were measured relative to the fission cross section of ^{235}U to an accuracy of approximately ±2% at neutron energies of 1.0, 2.25, 5.4, and 14.1 MeV. Combining these ratios with the known values of the fission cross section of ^{235}U leads to fission cross sections having an estimated uncertainty of ±3.5% and which are mostly in agreement with other recent measurements. (auth) (UK)

1968

48423 (RPI-328-133, pp 1-34) NEUTRON CROSS SECTIONS. (Rensselaer Polytechnic Inst., Troy, N. Y.).

The energy variation of $\bar{\nu}$ for ^{239}Pu was measured in 4 overlapping energy ranges from 0.01 eV to 10 keV; spin assignments for resonances of 22 to 100 eV were made. The absorption and fission cross sections of ^{239}Pu were measured from 0.01 eV to 30 keV; a preliminary evaluation of the neutron capture-to-fission ratio was made for the energy range from 1 to 30 keV. Simultaneous capture and transmission measurements were made for separated Hf isotopes. Gamma pulse-height spectra from experiments with the 1.25-m liquid scintillation capture detector are presented for ^{197}Au and ^{186}W. The angular distributions of neutrons scattered from resonances in Al were measured in order to determine the feasibility of this approach to assigning spins to $l > 0$ resonances. The technique of using the large liquid scintillation detector as an anticoincidence mantle was extended to fissile elements; preliminary scattering data for ^{239}Pu are presented. (D.C.W.)

7892 (DPST-67-83-10) USAEC-AECL COOPERATIVE PROGRAM MONTHLY PROGRESS REPORT, OCTOBER 1967. Rusche, B. C. (Du Pont de Nemours (E.I.) and Co., Aiken, S. C. Savannah River Lab.). Nov. 30, 1967. Contract AT(07-2)-1. 5p.

Comparisons of values of η obtained from HAMMER calculations for ^{235}U and ^{238}U and experimentally obtained values are made. Physics parameters for $(1/\eta)$ $(d\eta/dT)$ lattices are tabulated. Effects of temperature variation are outlined. (M.L.S.)

42011 (BNL-tr 219) MEASUREMENT OF THE RADIATIVE CAPTURE AND FISSION CROSS SECTION RATIOS FOR ^{235}U AND ^{239}Pu IN THE RESONANCE-NEUTRON ENERGY REGION. Ryabov, Yu. V.; Donssik, So; Chikov, N.; Yaneva, N. Translated by Stephen Amoretty (Brookhaven National Lab., Upton, N. Y.), from At. Energ. (USSR), 24: 351-62(1968). 20p. Dep. CFSTI.

Time-of-flight measurements of the neutron fission cross sections and the neutron capture-to-fission ratios of ^{239}Pu and ^{235}U were made over energy ranges of 5 eV to 23 keV and of 0.15 eV to 30 keV, respectively. Capture and fission resonance integrals were obtained from the data. (D.C.W.)

5811 (EURFNR-400) ENERGY AND TEMPERATURE DEPENDENT CAPTURE MEASUREMENTS BELOW 30 keV SUPPORTING DOPPLER EFFECT CALCULATIONS. Seufert, H.; Stegemann, D. (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Neutronenphysik und Reaktortechnik). Oct. 1967. 25p. (EUR-3675e; KFK-631; CONF-671043-10). Dep. CFSTI.

Work performed under United States-Euratom Fast Reactor Exchange Program.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany.

Energy- and temperature-dependent capture measurements below 30 keV neutron energy were performed in natural uranium, tungsten, and tantalum using the slowing down-time spectrometer technique. The experimental setup used for the experiments consists of a lead block of 1.3-m side length containing two experimental channels of 10 × 10 cm^2 cross section. Into the first channel the target of a 14-MeV neutron generator is introduced, whereas the second channel is used for insertion of the heated samples. Pulses of 14-MeV neutrons having a pulse width of about 1/μsec are used. The neutron energy is degraded first by inelastic collisions; afterwards only elastic collisions take place so that a specific relationship holds between mean neutron energy in the lead pile and the time after occurrence of the neutron pulse. Due to this time energy relation, a time analysis procedure for the detector counts is applied. Because the energy range below 30 keV neutron energy is most interesting for Doppler-effect investigations the slowing-down-time spectrometer is used to measure the capture ratios of hot-to-cold samples of natural uranium, tungsten, and tantalum. Thin samples were heated to different temperatures for this purpose, and the capture γ rays were detected by proportional counters. Because hot-to-cold capture ratios are measured the knowledge of the neutron flux is not necessary; therefore, a direct comparison of calculated and measured temperature-dependent cross sections is possible. A theoretical analysis of the experimental data for uranium is given. (auth)

20572 (LA-DC-9205) RELATIVE FISSION CROSS SECTIONS OF ^{236}U, ^{238}U, ^{237}Np, AND ^{235}U. Stein, W. E.; Smith, R. K.; Smith, H. L. (Los Alamos Scientific Lab., N. Mex.). [1966]. Contract W-7405-eng-36. 12p. (CONF-680307-24). Dep. CFSTI.

From 2nd Conference on Neutron Cross Sections and Technology, Washington, D. C.

The fission cross-section ratios ^{236}U/^{235}U, ^{238}U/^{235}U, and ^{237}Np/^{235}U were measured with pulsed, monoenergetic neutrons in the energy range 1.0 to 5.0 MeV with time-of-flight background discrimination. Vacuum evaporated fissile deposits (~0.5 mg/cm^2) were placed back to back between two 9.5 cm^2 surface barrier detectors. Slow and fast output signals were obtained simultaneously from each detector by means of separate electronic systems. Slow, linear pulses which exceeded a lower bound set to reject alpha particles were identified as fission events. The fraction of fragment pulses below this bias, determined from an extrapolation of the pulse-height spectrum, was ~1.2%. Only those fission events which occurred during and a few nanoseconds after the neutron burst were recorded. This time interval, determined by the measured time resolution (1.3 nsec FWHM) and the time walk of the smallest pulses (~2 nsec), was typically 6 nsec. Fission events induced by scattered neutrons which occur at later times were excluded. Characteristics of this detector system and preliminary data were reported earlier. Present results include additional data on ^{238}U/^{235}U and ^{237}Np/^{235}U and new data on ^{236}U/^{235}U. (auth)

18283 NEUTRON STRENGTH FUNCTION MEASUREMENTS IN THE MEDIUM AND HEAVY NUCLEI. Uttley, C. A.; Newstead, C. M.; Diment, K. M. (Atomic Energy Research Establishment, Harwell, Eng.). pp 165-74 of Nuclear Data for Reactors. Vienna, International Atomic Energy Agency, 1967.

From IAEA Conference on Nuclear Data, Paris. See STI/PUB-140(Vol.1); CONF-661014-(Vol.1).

Neutron total cross-section measurements were made between 100 eV and 1 MeV on nuclei near the mass-100 and mass-240 p-wave resonances using the Harwell "booster" pulsed neutron source and the 120-m and 300-m spectrometers. The s-wave strength function S_0 and distant level parameter R_0^∞ have usually been separately determined at lower energies and the corresponding p-wave parameters are obtained from a least-squares fit to the higher energy (>10 keV) total cross section using the average collision function expression from R-matrix theory. The d-wave strength function is also determined using plausible assumptions on the average parameters of the higher partial waves. The nuclei studied are ^{93}Nb, ^{98}Mo, ^{100}Mo, ^{103}Rh, ^{232}Th, ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu. (auth)

35295 (TPM-RFR-682) ^{239}Pu EVALUATIONS. Wallin, Marie (Aktiebolaget Atomenergi, [Studsvik] (Sweden)). Dec. 7, 1967. 11p. Dep.

The updating of the Speng library for ^{239}Pu to reflect recent measurements and evaluations of the cross sections and resonance parameters for ^{239}Pu is discussed. (D.C.W.)

1968

29986 (ORNL-TM-2140) MEASUREMENT OF THE NEUTRON FISSION AND CAPTURE CROSS SECTIONS FOR ^{233}U IN THE ENERGY REGION 0.4 TO 2000 eV. Weston, L. W.; Gwin, R.; deSaussure, G. (Oak Ridge National Lab., Tenn.); Fullwood, R. R.; Hockenbury, R. W. (Rensselaer Polytechnic Inst., Troy, N. Y.). Apr. 8, 1968. Contracts W-7405-eng-26; AT(30-3)-328. 64p. Dep. CFSTI.

The neutron capture cross section and fission cross section for ^{233}U were measured simultaneously in the neutron energy range 0.4 to 2000 eV. A pulsed and collimated neutron beam was passed through a ^{233}U fission chamber placed at the center of a large liquid scintillator. Capture and fission events in the ^{233}U chamber were detected in the scintillator by means of their prompt gamma rays. Coincident signals from the fission chamber and liquid scintillator distinguished fission from capture events. Comparisons with previously published data, using similar and different methods, are given. (auth)

48558 MEASUREMENT OF THE NEUTRON FISSION AND CAPTURE CROSS SECTIONS FOR ^{233}U IN THE ENERGY REGION 0.4 TO 2000 eV. Weston, L. W. (Oak Ridge National Lab., Tenn.); Gwin, R.; deSaussure, G.; Fullwood, R. R.; Hockenbury, R. W. Nucl. Sci. Eng., 34: 1-12(Oct. 1968).

The neutron capture cross section and fission cross section for ^{233}U were measured simultaneously in the neutron energy range 0.4 to 2000 eV. A pulsed and collimated neutron beam was passed through a ^{233}U fission chamber placed at the center of a large liquid scintillator. Capture and fission events in the ^{233}U chamber were detected in the scintillator by means of their prompt gamma rays. Coincident signals from the fission chamber and liquid scintillator distinguished fission from capture events. Comparisons with previously published data, using similar and different methods, are given. (auth)

3710 NEUTRON TOTAL AND ABSORPTION CROSS SECTIONS OF ^{238}Pu. Young, T. E. (Idaho Nuclear Corp., Idaho Falls); Simpson, F. B.; Berreth, J. R.; Coops, M. S. Nucl. Sci. Eng., 30: 355-61(1967).

The neutron total cross section of ^{238}Pu was measured from 0.008 to 6500 eV. These data give single-level Breit-Wigner parameters for resonances below 200 eV. The observed total cross section at 2200 m/sec is 588 b. A value of 532 b was calculated for the effective (equivalent 1/v) thermal absorption cross section. Parameters of individual resonances below 200 eV and average parameters at higher energies give a resonance absorption integral of 164 ± 15 b, and a value of $(1.10 \pm 0.20) \times 10^{-4}$ for the s-wave neutron strength function ($\bar{\Gamma}_n^0/D$). (auth)

37341 (IN-1132) NEUTRON TOTAL AND ABSORPTION CROSS SECTIONS OF ^{242}Pu. Young, T. E.; Reeder, S. D. (Idaho Nuclear Corp., Idaho Falls). June 1968. Contract AT(10-1)-1230. 85p. Dep. CFSTI.

The total neutron cross section of ^{242}Pu was measured from 0.008 to 8000 eV using PuO_2 powder samples in the Materials Testing Reactor (MTR) fast chopper. The data were analyzed to give the thermal absorption cross section and resonance parameters below 180 eV. The observed total neutron cross section at 0.0253 eV is 39 ± 1 b, and the effective (equivalent 1/v) thermal absorption cross section derived from the measurement is 22 ± 2b. Parameters of individual resonances below 180 eV and average parameters at higher energies give a resonance absorption integral of 1090 ± 60 b and a neutron s-wave strength function ($\bar{\Gamma}_n^0/D$) of $(0.85 \pm 0.10)10^{-4}(eV)^{-1/2}$. (auth)

35650 MEASUREMENT OF THE EFFECTIVE CAPTURE-TO-FISSION RATIO IN ^{239}Pu AND ^{235}U IN TWO DIFFERENT FAST REACTOR SPECTRA. Andersson, T. L.; Hellstrand, E.; Hakansson, R.; Bajbor, Z. (AB Atomenergi, Studsvik, Sweden). pp 171-85 of Fast Reactor Physics. Vol. II. Vienna, International Atomic Energy Agency, 1968.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.2); CONF-671043-(Vol.2).

Integral values of α for ^{235}U and ^{239}Pu have been deduced from experiments in two different cores (median ^{235}U fission energy 50 and 180 keV, respectively) in the FRO reactor. The method of measurement is the same as that used for instance at ZEBRA. The experiment includes reactivity measurements of the sample material and of a standard, ^{10}B, as well as an absolute determination of the fission rate in ^{239}Pu and ^{235}U and the capture rate in ^{10}B. The experimental α values agree well with the calculated ones for the hard spectrum core measurements, both for ^{235}U and ^{239}Pu. The measured value for ^{239}Pu in the soft core is slightly higher than the calculated one. The discrepancy for ^{235}U is large and hitherto unexplained. (auth)

33086 (ANL-7310, pp 431-511) REACTOR COMPUTATION METHODS AND THEORY. (Argonne National Lab., Ill.).

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URANIUM— neutron cross sections for, crystalline effects on Doppler-broadened; resonance integrals for, crystalline effects on (M.L.S.)

50605 ANALYSIS IN TERMS OF A GENERALIZED OPTICAL MODEL, OF THE CROSS SECTION OF ^{238}U IN THE ENERGY INTERVAL (0.05 TO 15) MeV. Baldoni, Bruna; Saruis, Anna Maria (CNEN, Bologna). pp 741-50 of Fisica del Reattore. Rome, Consiglio Nazionale delle Ricerche, 1966. (In Italian).

From Conference on Physics of Reactors, Milan. See CONF-469.

Use of a generalized optical model for analysis of the ^{238}U cross section is described. Differential cross sections are shown for E = 0.650 MeV, 1.1 MeV, 2.5 MeV, 4.1 MeV, 7.0 MeV, and 14.1 MeV. The equations used for the analysis are given. Total and differential inelastic cross sections are shown. Behavior of the Legendre polynomial expansion coefficient is determined as a function of energy. (M.L.S.)

35542 INACCURACIES IN THE CHARACTERISTIC PARAMETERS OF A FAST POWER REACTOR DUE TO THE PREVAILING UNCERTAINTIES IN THE BASIC NEUTRON DATA. THEIR EVALUATION AND THE EXTENT TO WHICH THEY CAN BE REDUCED BY CRITICAL EXPERIMENTS. Barre, J. Y.; Ravier, J. (CEA, Cadarache, France). pp 205-23 of Fast Reactor Physics. Vol. I. Vienna, International Atomic Energy Agency, 1968. (In French).

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.1); CONF-671043-(Vol.1).

The inaccuracies in the characteristic parameters of a fast reactor due to uncertainties in the basic neutron data, using a generalized perturbation method are described. The results of critical experiments to improve the accuracy of the calculated forecasts and thereby possibly improve the cross sections themselves are presented. (auth)

50598 (LA-3527) NEUTRON CROSS SECTIONS FOR ^{235}U AND ^{238}U IN THE ENERGY RANGE 1 keV TO 14 MeV. Berlijn, J.-J. H.; Hunter, R. E.; Cremer, C. C. (Los Alamos Scientific Lab., N. Mex.). Aug. 1968. Contract W-7405-eng-36. 33p. Dep. CFSTI.

Recommended cross sections for ^{235}U and ^{238}U are presented. Comparisons of calculated and experimental values of integral systems were used as a guide in choosing the fits to microscopic cross-section data. (auth)

35333 MEASURING THE RELATIVE INTENSITY OF FISSION REACTIONS. Bondarenko, V. V.; Bushuev, A. V.; Voropaev, A. I.; Zinov'ev, V. P.; Ukraintsev, F. I.; Yurova, L. N. At. Energ. (USSR), 24: 82-4(Jan. 1968). (In Russian).

In obtaining information on the space-energy distribution of neutrons in a reactor, the distributions of fission intensities of different isotopes are measured at various points with fission chambers or by an activation method. A series of measurements were made on a fast neutron assembly containing rods filled with enriched uranium. Small fission chambers (8 mm in diameter and 40 mm in height), and natural and 90% uranium foils were used in the measurements. The fission product activity in the foils was measurements. The fission product activity in the foils was measured on a NaI(Tl) counter. An analysis of the experimental results showed that inhomogeneities due to the heterogeneity of the lattice were observed in the space-energy distribution of neutrons. The neutron spectra in the channels and in the spaces between the channels differed considerably. The heterogeneity of the system had an effect on the ^{238}U fission reaction, but not on the ^{235}U fission reaction. Both the fission chamber and foil activation methods were free of systematic errors as shown by experiments in a homogeneous region. (TTT)

16053 INTEGRAL AND DIFFERENTIAL CROSS SECTIONS OF ^{232}Th FISSION BY NEUTRONS. Ermagambetov, S. B.; Smirenkina, L. D.; Smirenkin, G. N. 18p. (In Russian). (CONF-661014-51). DTIE.

From IAEA Conference on Nuclear Data, Paris.

The energy dependence of the fission cross section $\sigma_f(E_n)$ usually shows a complex structure in odd fissionable nuclei near the threshold. Characteristics of the lower fission channels can be derived by comparing the observed fission cross section σ_f and the angular distributions of the fission products with the theoretical values. Then the characteristics of the lower fission channels are selected in such a way as to obtain agreement between the calculated and experimental values. Such an analysis led to the following succession of lower channels of the transitory ^{233}Th nucleus, which were excited in the ^{232}Th (n,f) reaction by neutrons having an energy E_n < 1.6 MeV: $\frac{1}{2}^+$, $\frac{3}{2}^-$, and $\frac{3}{2}^+$. The new channels with K = $\frac{1}{2}$ explain the inflexion of σ_f in the region of neutron energy $E_n \approx 1.1$ MeV. (TTT)

18250 RECOMMENDED VALUES FOR THE NUMBER OF NEUTRONS PER FISSION. Fillmore, F. L. (Atomic International, Canoga Park, Calif.). J. Nucl. Energy, 22: 79-97(Feb. 1968).

The available experimental data for the value of ν for fissile and fertile isotopes are reviewed. The absolute determination of ν for ^{252}Cf is discussed, since this provides the standard for normalizing the other values. Based on weighted averages of the experimental data, recommended 2200 m/sec values are presented for ^{235}U, ^{233}U, ^{239}Pu and ^{241}Pu. The available data relating the dependence of ν on incident neutron energy are tabulated, and straightline fits to the data are made by the method of least squares. These results are of direct value in reactor calculations and the related evaluation of nuclear cross sections. Although there are instances where the data indicate structure of a more complicated nature, no attempt is made to analyze these situations. For practical purposes, a reasonably satisfactory fit to the data can be made with not more than two straight lines. (auth) (UK)

33016 THE HIGH-ENERGY INELASTIC SCATTERING AND THE EARLY NEUTRON DECAY IN ^{238}U ASSEMBLIES. Gozani, Tsahi; d'Oultremont, P. (Gulf General Atomic Inc., San Diego, Calif.). Trans. Amer. Nucl. Soc., 11: 292-3(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

50599 (LA-3528) NEUTRON CROSS SECTIONS FOR ^{239}Pu AND ^{240}Pu IN THE ENERGY RANGE 1 keV TO 14 MeV. Hunter, R. E.; Berlijn, J.-J. H.; Cremer, C. C. (Los Alamos Scientific Lab., N. Mex.). July 1968. Contract W-7405-eng-36. 27p. Dep. CFSTI.

Recommended cross sections for ^{239}Pu and ^{240}Pu are presented. Comparisons of calculated and experimental values of integral systems were used as a guide in choosing the fits to microscopic cross-section data. (auth)

32597 RESONANCE PARAMETERS OF ^{240}Pu. PART I. NEUTRON WIDTHS. Kolar, W.; Boeckhoff, K. H. (EURATOM, Geel, Belg.). J. Nucl. Energy, 22: 299-315(May 1968).

The neutron total cross section of ^{240}Pu was measured with high resolution in the energy range 20 eV to 5.7 keV. Up to this energy, 264 resonances were detected and analyzed with respect to Γ_n using the area program of Atta and Harvey. For 32 resonances between 38 and 820 eV, the full set of resonance parameters E_r, Γ_n and Γ_γ could be evaluated by combining the results of the transmission experiment with those of the capture experiment described in Part II. For the 102 resonances up to 1500 eV, a mean level spacing of $\langle D \rangle$ = 14.7 ± 0.8 eV was obtained. On the assumption that all resonances in that range are of the s-type, the strength function yields S_0 = 1.05 ± 0.16 × 10^{-4}. (auth)

37373 ABSOLUTE RADIATIVE CAPTURE CROSS SECTION FOR FAST NEUTRONS IN ^{238}U. Menlove, H. O.; Poenitz, W. P. (Kernforschungszentrum, Karlsruhe, Ger.). Nucl. Sci. Eng., 33: 24-30(1968).

The capture cross section of ^{238}U was measured absolutely at a neutron energy of 30 keV using kinematically collimated neutrons from the $^{7}Li(p,n)^{7}Be$ reaction near threshold. Activation techniques were used to determine both the number of capture events and the number of neutrons that occurred during the irradiation. The result of the ^{238}U capture cross section measurement is 479 ± 14 mb at 30 keV. In addition, the shape of the ^{238}U capture cross section was measured for neutron energies from 25 to 500 keV using neutrons from the $^{7}Li(p,n)^{7}Be$ reaction. The capture reactions in the ^{238}U target were detected using a large liquid scintillator tank and time-of-flight techniques. The relative neutron flux was measured using a flat response neutron detector. The cross-section shape measurement was normalized to the present absolute measurement at 30 keV. The present measurement was compared with several measured values, theoretical calculations, and compiled values of the ^{238}U capture cross section as given by other authors. (auth)

32621 RATIOS OF FAST NEUTRON FISSION CROSS SECTIONS OF ^{233}U, ^{235}U, AND ^{239}Pu. Nesterov, V. G.; Smirenkin, G. N. At. Energ. (USSR), 24: 185-7(Feb. 1968). (In Russian).

Relative fission cross sections of σ_3/σ_5 (^{233}U to ^{235}U) and of σ_9/σ_5 (^{239}Pu to ^{235}U) were determined over a neutron energy E of 0.3 to 2.5 MeV at a relative accuracy of 1 to 2% by measuring the number of fissions in a double ionization chamber containing layers of the isotopes which were to be compared. Since the fission cross section for the ^{235}U isotope is well known with a high accuracy of 2.5 to 3% at higher neutron energies, it becomes possible to derive more accurate data on the cross sections of ^{233}U and ^{239}Pu. The results were compared with the data compiled by Davey, and it was found that the two sets of data were in good agreement at $E_n < 0.7$ MeV, but deviated from each other by 7 to 10% at higher values of E_n. The results are in good agreement with the data of Lamphere. (TTT)

39651 INVESTIGATION OF SPACE-DEPENDENT FAST FISSION RATIOS AND MEASUREMENTS OF FISSION-FRAGMENT RECOIL RANGES IN ALUMINUM. Osias, David Justin. Ithaca, N. Y., Cornell Univ., 1967. 147p.

Thesis.

Space-dependent values of δ_{28} (^{238}U to ^{235}U fission ratio) are measured in two cores of the 2.1% enriched UO_2, light water moderated critical facility at Cornell University. Measurements are also made in an isolated 0.600-in.-diameter fuel rod in a purely thermal flux. Heterogeneous finite lattice calculations are made with different cross section sets to investigate cross section uncertainties. The ^{238}U cross sections recommended by the Atomic Weapons Research Establishment produce agreement of the calculations with the measurement of δ_{28} in a single fuel rod ($\delta_{28} = 0.0150 \pm .0003$), and because σ_{incl} has little effect on this calculated value, the agreement indicates an integral accuracy for σ_f of about 2%, which is the accuracy of the isolated rod experiment. For the finite lattice calculations, the core region is taken to be a homogenized cylinder with the space-dependent fast flux at each of 30 energy points approximated by a fourth-order even polynomial, using the neutron transport method of K. B. Cady and M. Clark, Jr. Two formulas for correcting anisotropic scattering result in high and low limits for δ_{28}. Energy-dependent disadvantage factors for the 30 energy points, used for obtaining homogenized cross sections, are obtained from a multi-energy escape probability calculations for the infinite lattice cell. The calculated intensive (infinite lattice) values of δ_{28} are substantially higher (about 15% to 20%) than the extensive values calculated for the finite lattices, showing a significant leakage effect. The two experimental points for the 1.5/1 core suggest that the average σ_f recommended by AWRE is accurate to about 2%. For the 3/1 core, the theory and experiment (about 1% experimental error) show a small unexplained discrepancy. The assumed uncertainty of σ_{inel} and other cross sections (obtained by subjectively surveying experimental data for cross sections) is not large enough to account for the 3 to 4% discrepancy. The experimental data are obtained by using a lithium-drifted germanium detector to measure the activities of the 1.6-MeV ^{140}La photopeak in highly depleted (23 ppM ^{235}U) and in 2.1% enriched UO_2 wafers. The precision of this method is believed to be better than that obtained from threshold gamma-ray counting, partly because no measurement with a double fission chamber of the time-dependent ratio of ^{238}U to ^{235}U fission product activities is needed. There is, however, a systematic uncertainty of about ±2% in the ^{140}La yield from fission of ^{238}U, and this 2% must be added to the standard errors quoted here for the δ_{28} measurements. A study is made of the gross fission product gamma-ray spectrum to discover any better means of measuring fission events. None is found, but it is discovered that gamma-ray spectra from the germanium detector can be used for quantitative measurements of many fission products with no chemical separations. A technique for computerized evaluation of complex gamma-ray spectra is developed. By using the computer program to analyze the spectra obtained with the germanium detector, measurements are made of the mean recoil ranges in aluminum of 19 high yield fission fragments from thermal fission of ^{235}U. Ranges are calculated from the relative numbers of fission fragments stopping in each of four aluminum catcher foils (1.204 mg/cm^2) placed next to a thin film of ^{235}U. Precision varies from 0.2 to 1.5%, with most measurements having standard deviations between 0.3 and 0.8%. Previous experimenters report comparable accuracy. However, the present experiment involves less effort, and provides ranges for many atomic masses which had not been measured previously. (Diss. Abstr.)

1968

18106 NEUTRON CROSS-SECTION EVALUATIONS: PAST, PRESENT, AND FUTURE. Parker, K. (Atomic Weapons Research Establishment, Aldermaston, Eng.); Goldman, D. T.; Wallin, L. pp 293-307 of Nuclear Data for Reactors. Vienna, International Atomic Energy Agency, 1967.

From IAEA Conference on Nuclear Data, Paris. See STI/PUB-140(Vol.2); CONF-661014-(Vol.2).

The need and requirements for cross-section evaluations is discussed; and the evaluations that were available on June 1, 1966, are reviewed. (D.C.W.)

3687 (EURFNR-404) SOME NEW MEASUREMENTS AND RENORMALIZATIONS OF NEUTRON CAPTURE CROSS SECTION DATA IN THE keV ENERGY RANGE. Poenitz, W. P.; Kompe, D.; Menlove, H. O.; Beckurts, K. H. (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Angewandte Kernphysik). Oct. 1967. 14p. (EUR-3679e; KFK-635; CONF-671043-9). Dep. CFSTI.

Work performed under United States-Euratom Fast Reactor Exchange Program.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany.

The absolute neutron capture cross section of Au was measured for neutron energies of 25 to 500 keV to an accuracy of about ±5%. Absolute normalization of the relative cross-section curve was performed at a neutron energy of 30 keV. The capture cross section of ^{238}U was measured over the same energy range. The capture cross sections, relative to Au, of a number of medium-weight and heavy nuclides were measured, using a time-of-flight method and a large liquid scintillator tank, in the energy range from 10 to 150 keV. Evaluated data are presented for Cs, Hf, Mo, Nb, Re, Ta, and W. The implications of the new data for fast reactor calculations were studied. (D.C.W.)

V. NEUTRON CROSS SECTIONS

4. Capture-To-Fission Ratios

1967

42610 (IN-1060) ETA RATIOS OF ^{239}Pu AND ^{241}Pu RELATIVE TO ^{235}U. Fast, E.; Aber, E. F. (Idaho Nuclear Corp., Idaho Falls). Aug. 1967. Contract AT(10-1)-1230. 42p. Dep. CFSTI.

The ratios of the neutron reproduction constants η(Pu-239)/η(U-235) and η(Pu-241)/η(U-235) were determined from reactivity measurements in ARMF-I and ARMF-II. Results for 2200 m/sec and Maxwellian average values are given. (auth)

38320 MEASUREMENT OF ^{235}U ALPHA IN AN INTERMEDIATE-ENERGY NEUTRON SPECTRUM. Fox, W. N.; Kinchin, G. H.; Sanders, J. E. (United Kingdom Atomic Energy Authority, Winfrith, Eng.). Trans. Amer. Nucl. Soc., 10: 231(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11–15, 1967. See CONF-670602.

42616 (WAPD-TM-625) MEASUREMENT OF CHANGE IN α AND FISSION RATE WITH TEMPERATURE FOR ^{235}U (LWB-LSBR DEVELOPMENT PROGRAM). Green, L. (Bettis Atomic Power Lab., Pittsburgh, Pa.). Aug. 1967. Contract AT(11-1)-Gen-14. 21p. Dep. CFSTI.

The Doppler effect of an enriched (93.2%) UO_2 target in a 1/E incident neutron spectrum was investigated in a beam geometry. The fission and capture rates of the target, enclosed in a quartz glass furnace, were followed by means of two scintillation counters operating a crossover-pickoff coincidence system. Analysis of the time data yields change in α, the capture to fission ratio, and in the fission integral, as a function of temperature. A single target 0.090-inch thick was studied in the range from room temperature to 800°C. Experimental conclusions obtained were the capture to fission ratio, α, increased with temperature, and within a 2% uncertainty no change was observed in the fission rate over the temperature interval studied. Monte Carlo calculations using recent resonance data were performed for the conditions of the experiment and agreement was obtained with these conclusions. (auth)

4434 (IDO-14678) BURNUP DETERMINATION OF NUCLEAR FUELS. Project Report for the Quarter January 1–March 31, 1966. Maeck, William J.; Lisman, Frederick L.; Rein, James E. (eds.) (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). Oct. 1966. Contract AT(10-1)-205. 23p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Measurements were made of the α (capture-to-fission ratio) for U-233 and U-235. Two capsules containing a known amount of highly enriched U-235 and four capsules containing a known amount of highly enriched U-233 were irradiated in the MTR and subsequently analyzed for total and isotopic uranium concentration. Based on these data, tentative effective $\hat{\alpha}$ values (at a Cd ratio for Co ≈ 15) and the computed 2200 m/sec values are: 0.0997 ± 0.0016 and 0.0984 ± 0,0016 for U-233, and 0.1768 ± 0.0015 and 0.1716 ± 0.0015 for U-235, respectively. (auth)

46952 ATOM RATIOS AND EFFECTIVE CROSS-SECTION RATIOS IN HIGHLY DEPLETED PLUTONIUM–ALUMINUM ALLOY FUEL. Reardon, W. A.; Christensen, D. E. (Pacific Northwest Lab., Richland, Wash.). Contract AT(45-1)-1830. Nucl. Sci. Eng., 30: 222-32(1967). (BNWL-SA-558).

The graded exposure of 4 Pu–Al alloy, 19-rod clustered fuel elements, and the subsequent destructive sampling of the elements have provided experimental data showing the variation of Pu isotopes with irradiation. Irradiations were conducted in the heavy-water-moderated and -cooled Plutonium Recycle Test Reactor. Using ^{137}Cs as a fission indicator, the depletion of the initial Pu to 50.4 ± 1.1% is determined. Reactor effective cross-section ratios for the Pu isotopes are derived from the data, and results show that the capture-to-fission cross-section ratio for ^{239}Pu(α^{239}) is 0.426 ± 0.019. (auth)

1967

12248 DIRECT DETERMINATION OF ^{235}U CAPTURE-TO-FISSION RATIO IN A ZERO-POWER REACTOR. Redman, W. C.; Bretscher, M. M. (Argonne National Lab., Ill.). Nucl. Sci. Eng., 27: 34-44(Jan. 1967).

An experimental method for the determination of the spectral average of the capture-to-fission ratio $\bar{\alpha}$ for materials inserted in a low-flux reactor is described. The procedure involves a comparison of reactor response to oscillated samples of a fissile material, an absorber, and a spontaneous fission neutron source, plus an experimental determination of fission rate for the fissile material and capture rate for the absorber. In addition, it is necessary that the neutron source be calibrated. These experimental results, combined with a knowledge of the number of neutrons per fission for the fissile material, yield a value of the quantity $1 + \bar{\alpha}$. This method has been tested in Hi-C Core 10, a critical assembly of 3%-enriched-UO_2 fuel pins, moderated and reflected by light water, in a lattice spacing which yields a H-to-^{238}U atom ratio of 2:91. The oscillator and absolute counting data yield a value of 0.217 for the central capture-to-fission ratio of ^{235}U, with a standard deviation of ±0.015. This agrees well with values derived from a combination of measured ^{235}U fission cadmium ratios and calculated thermal and epithermal values for α. (auth)

40420 (AEEW-R-526) MEASUREMENTS AND CALCULATIONS OF RATIOS OF EFFECTIVE FISSION CROSS SECTIONS IN THE ZERO-POWER FAST REACTOR, ZEBRA. Stevenson, J. M.; Broomfield, A. M. (Atomic Energy Establishment, Winfrith (England)). June 1967. 22p. Dep. CFSTI. UK 4s. 0d.

The comparison of measured and calculated central fission ratios provides a useful method for checking the accuracy of calculated spectra and fission cross-section data. A set of parallel plate fission chambers was specially made for Zebra. The design was based on that of Kirn's chambers used by the ZPR-III group at ANL, Idaho, and incorporated a number of improvements. In particular, the wall thickness was reduced to reduce the degradation of the spectrum. The fissile coatings in the earlier chambers were prepared by a painting technique and those in the later chambers by an electrodeposition method. The mass deposited was determined by low geometry α-assay. The construction of the chambers and the method of calibration and use are described. Central fission ratios in six Zebra cores measured to an accuracy of 1 to 3% are compared with values computed using the FD2 data set and the CRAM or SCRAMBLE multigroup diffusion theory programs. This comparison suggests that the calculations give too few neutrons at high energies. There is also evidence that some of the fission cross-section data used in the FD2 set is in error. (auth)

38305 SIMULTANEOUS MEASUREMENTS OF THE ^{235}U FISSION AND CAPTURE CROSS SECTIONS. Weston, L. W. (Oak Ridge National Lab., Tenn.); Gwin, R.; de Saussure, G.; Ingle, R. W.; Fullwood, R. R.; Hockenbury, R. W. Trans. Amer. Nucl. Soc., 10: 220-1(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

1968

35650 MEASUREMENT OF THE EFFECTIVE CAPTURE-TO-FISSION RATIO IN ^{239}Pu AND ^{235}U IN TWO DIFFERENT FAST REACTOR SPECTRA. Andersson, T. L.; Hellstrand, E.; Hakansson, R.; Bajbor, Z. (AB Atomenergi, Studsvik, Sweden). pp 171-85 of Fast Reactor Physics. Vol. II. Vienna, International Atomic Energy Agency, 1968.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.2); CONF-671043-(Vol.2).

Integral values of α for ^{235}U and ^{239}Pu have been deduced from experiments in two different cores (median ^{235}U fission energy 50 and 180 keV, respectively) in the FRO reactor. The method of measurement is the same as that used for instance at ZEBRA. The experiment includes reactivity measurements of the sample material and of a standard, ^{10}B, as well as an absolute determination of the fission rate in ^{239}Pu and ^{235}U and the capture rate in ^{10}B. The experimental α values agree well with the calculated ones for the hard spectrum core measurements, both for ^{235}U and ^{239}Pu. The measured value for ^{239}Pu in the soft core is slightly higher than the calculated one. The discrepancy for ^{235}U is large and hitherto unexplained. (auth)

53454 (BNWL-828) PLUTONIUM UTILIZATION PROGRAM. Technical Activities Quarterly Report, March-May 1968. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Sept. 1968. 67p. Dep. CFSTI.

CRITICALITY STUDIES—measurements of critical number of aluminum-plutonium fuel rods in water moderator, effect of lattice pitch and boron concentration on; microscopic lattice parameter measurements on water-moderated plutonium oxide (PuO_2)-uranium oxide (UO_2) rods; measurements of boron worth and concentration for cold and xenon-free PRTR Batch Core

NEUTRONS—capture-to-fission ratio measurements in plutonium-239; multiplication factor determination in water-moderated plutonium oxide (PuO_2)-uranium oxide (UO_2) lattices; resonance escape probability determination for uranium-238

18298 MEASUREMENT OF THE NEUTRON CAPTURE AND FISSION CROSS SECTIONS AND OF THEIR RATIO ALPHA FOR ^{233}U, ^{235}U, AND ^{239}Pu. de Saussure, G. (Oak Ridge National Lab., Tenn.); Weston, L. W.; Gwin, R.; Ingle, R. W.; Todd, J. H.; Hockenbury, R. W.; Fullwood, R. R.; Lottin, A. pp 233-49 of Nuclear Data for Reactors. Vienna, International Atomic Energy Agency, 1967.

From IAEA Conference on Nuclear Data, Paris. See STI/PUB-140(Vol.2); CONF-661014-(Vol.2).

A technique was developed to measure simultaneously the neutron capture and fission cross sections of fissile nuclei. Since the two cross sections are measured simultaneously, errors associated with uncertainties in the relative energy resolution and calibration of the two measurements are eliminated. Measurements of σ_c and σ_f for ^{235}U in the neutron energy range of 3.25 to 25 eV have been published. These measurements have now been extended to cover the range of 1 to 100 eV, and the precision and energy resolution have been greatly improved. The fission cross section is in good agreement with recent measurements using different techniques. At low energy, where the instrumental resolution is small compared to the Doppler broadening and where resonance scattering is unimportant, the directly measured capture cross section is consistent over many of the resonances with that obtained indirectly by subtracting the fission and potential scattering from the total cross-section. The capture and fission resonance integrals and their ratio α_{RI}, obtained from our measurements, agree within the uncertainties with the direct integral measurements of these parameters. Further measurements on ^{235}U and ^{233}U over the neutron energy range of 1 eV to a few keV are now in progress. The limitations of the experimental method are discussed, and a detailed comparison of cross sections obtained by different techniques are presented. To compute the breeding ratio, the Doppler coefficient, and other parameters of large fast power reactors, it is important to know α, the ratio of capture-to-fission, for the main fissile isotopes, and particularly for ^{239}Pu, in the keV neutron energy region. Hopkins and Diven have performed direct measurements of α for ^{233}U, ^{235}U, and ^{239}Pu with monoenergetic neutrons at 30, 60, and above 175 keV. But the value of α for ^{239}Pu varies by more than a factor of two between 30 and 60 keV, and a detailed knowledge of the variation of this parameter with energy in the keV neutron energy range appears desirable. The application of the time-of-flight technique permits extending the direct measurements of α to energies where monoenergetic neutron sources are not readily available. Detailed measurements for α for ^{239}Pu are now in progress, at few-keV intervals in the range of 10 to 100 keV and at 100-keV intervals in the range of 100 to 600 keV. The technique has already been used with ^{235}U and the results, now published, were found in agreement with those of Hopkins and Diven in the range of overlap. The results for ^{239}Pu are compared with those of Hopkins and Diven and with values of α obtained from direct measurements of η by Spivak et al. The factors limiting the precision of the measurements are discussed in some detail. (auth)

35288 (BNWL-774) PCTR INTEGRAL MEASUREMENTS OF ^{239}Pu ALPHA ($\overline{\alpha}$) IN FAST SPECTRA. Lanning, D. D.; Busselman, G. J.; Bennett, C. L.; Newman, D. F.; Bierman, S. R. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Apr. 15, 1968. Contract AT(45-1)-1830. 42p. Dep. CFSTI.

An analysis of the feasibility of integral measurements of $\overline{\alpha}$ for ^{239}Pu in the Physical Constants Test Reactor (PCTR) as a function of the carbon-to-uranium ratio in plutonium oxide-uranium oxide-carbon mixtures is summarized. The experiments proposed will use 8 to 14% PuO_2 (of low ^{240}Pu) in UO_2 mixed with graphite. It is concluded that integral values of $\overline{\alpha}$ for ^{239}Pu can be measured in the PCTR with an uncertainty less than ±0.06. (D.C.W.)

18291 MEASUREMENTS OF ETA, ALPHA, AND NEUTRON CROSS SECTIONS FOR ^{239}Pu ON THE HARWELL NEUTRON TIME-OF-FLIGHT SPECTROMETER. Patrick, B. H.; Schomberg, M. G.; Sowerby, M. G.; Jolly, J. E. (Atomic Energy Research Establishment, Harwell, Eng.). pp 117-27 of Nuclear Data for Reactors. Vienna, International Atomic Energy Agency, 1967.

From IAEA Conference on Nuclear Data, Paris. See STI/PUB-140(Vol.2); CONF-661014-(Vol.2).

Eta values and total and fission cross sections were measured using the Harwell 45-MeV electron linear accelerator neutron time-of-flight spectrometer. The 34.9-m flight path, with a resolution of 7.2 ns/m, was used for the energy range 10 eV to 1 keV; and the 97.5-m flight path, with a resolution of 2.5 ns/m, was used for the energy range from 50 eV to 30 keV. The experimental methods employed were similar to those of Brooks et al., in which the fission neutron yield and the transmission of a number of samples of different thicknesses are measured together with the shape of the incident neutron spectrum, all the measurements being made with the same resolution. In the yield measurements, fast neutrons from fission events were detected in liquid scintillators, and pulse-shape discrimination was used to reject events caused by gamma rays. The transmission measurements were performed with a lithium glass scintillator, and a 1/V detector was used to measure the spectrum. Backgrounds were measured using the "black resonances" technique. The measurements at 34.9 m were normalized in the region of 10 eV to the eta values obtained by Brooks et al. and the eta values from the 97.5-m measurements were normalized to the 34.9-m measurements in the region of 50 eV. The results were corrected for multiple scattering in the samples.

Capture cross sections were derived from the total and fission cross sections by assuming values for the scattering cross section. A direct measurement of alpha (the ratio of the capture-to-fission cross sections) is in progress over the energy range 10 eV to 30 keV. This will lead to a better estimate of the capture cross sections since alpha measurements are not so sensitive to the magnitude of the scattering cross section. (auth)

37364 MEASUREMENTS OF ^{235}U AND ^{239}Pu NEUTRON CAPTURE-TO-FISSION RATIO FOR RESONANCE INCIDENT NEUTRON ENERGIES. Ryabov, Yu. V.; Don-Sœk, So; Chikov, N.; Yaneva, N. At. Energ. (USSR), 24: 351-62(Apr. 1968). (In Russian).

The neutron fission cross sections and capture-to-fission ratios of ^{239}Pu and ^{235}U were measured in the neutron energy ranges 5 eV to 23 keV and 0.15 eV to 30 keV, respectively, using the time-of-flight method. Fission events were registered by delayed coincidence between prompt γ-ray pulses and slowed fission neutrons. The counting rate of capture events was determined by the anticoincidences. The capture-to-fission ratios and the fission cross sections are presented as functions of energy. Values of the capture and fission resonance integrals are tabulated. (D.C.W.)

35345 NEW METHOD OF MEASURING ALPHA(E) FOR ^{239}Pu. Schomberg, M. G.; Sowerby, M. G.; Evans, F. W. (Atomic Energy Research Establishment, Harwell, Eng.). pp 289-301 of Fast Reactor Physics. Vol. I. Vienna, International Atomic Energy Agency, 1968.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.1); CONF-671043-(Vol.1).

A special liquid scintillator detector has been developed for the purpose of measuring alpha(E) for ^{239}Pu in the energy range 10 eV to 30 keV using the Harwell Linear Accelerator time-of-flight spectrometer. Alpha(E) is the ratio of capture to fission cross sections as a function of incident neutron energy. The detector has two outputs, one responding to gamma-ray interactions and the other to fast neutrons. The efficiency of the detector for gamma rays is arranged to be proportional to the gamma-ray energy. This property is achieved by utilizing an improved Moxon-Rae design and ensures that the efficiency of the detector for radiative capture events is constant irrespective of the nature of the gamma-ray cascade. The fast neutrons are also detected in the liquid scintillator and pulse shape discrimination is used to reject events produced by gamma rays. As a gamma-ray detector the device is sensitive to both radiative capture events and to the prompt gamma rays produced in fission. However, a correction for this latter component is made using the information from the fast neutron output which is essentially only sensitive to fission events. For each of the time-of-flight timing channels the ratio of the corrected counts from the gamma detector to the number of fission events detected is equal to K × alpha(E), where K is a constant determined by normalization. The technique of measuring both capture and fission simultaneously ensures that incident neutron energy spectrum charges and resolution effects are unimportant and also reduces the multiple scattering corrections. The detector system is described and some of the data obtained are shown. (auth)

32589 AN INVESTIGATION OF ^{239}Pu ALPHA IN THE keV ENERGY RANGE THROUGH SOME LATTICE EXPERIMENTS. Sehgal, Bal Raj; Price, Glenn A.; Windsor, Henry H. (Brookhaven National Lab., Upton, N. Y.). Trans. Amer. Nucl. Soc., 11: 202-3(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

6601 INSTRUMENTATION TO MEASURE THE RATIO OF NEUTRON CAPTURE TO FISSION CROSS SECTIONS AT keV NEUTRON ENERGIES. Todd, J. H.; Weston, L. W.; Ingle, R. W. (Oak Ridge National Lab., Tenn.). Contract W-7405-eng-26. Nucl. Instrum. Methods, 58: 143-50(Jan. 1968). (ORNL-P-3956).

Instrumentation for measurement of α (the ratio of the neutron capture cross section to the fission cross section) was designed, assembled, and calibrated, and measurements were made for ^{239}Pu, ^{233}U, and ^{235}U at neutron energies from 10 to 600 keV. The detector consisted of eight photomultiplier tubes located around the periphery of a 210-gal tank of gadolinium-loaded liquid scintillator. The time resolution of the system was 6.5×10^{-9} sec under operating conditions. The energy resolution at the 2.5-MeV sum peak of ^{60}Co was 35%. Neutron energies were measured by time-of-flight techniques using a flight path of 1 meter. Fission events were distinguished from capture events by the detection of thermalized fission neutrons following the primary events. Measurement of the relative efficiency of the large scintillator tank for detecting capture and fission events was made by simultaneously compiling the background and foreground pulse-height spectra for both the capture and fission events. Typical spectra for the neutron time of flight for capture and fission events along with the background and foreground spectra or both types of events are shown. The techniques and adjustments used to obtain the timing resolution on the large tank are discussed. The logic necessary for distinguishing fission events and the foreground and background pulse-height spectra associated with both types of events are described. (auth)

V. NEUTRON CROSS SECTIONS

5. Resonances

1967

4318 NEUTRON-SPECTROSCOPIC INVESTIGATION OF SEPARATED SILVER ISOTOPES. **Muradyan, G. V.; Adamchuk, Yu. V.** (Kurchatov Inst. of Atomic Energy, Moscow). Vienna, International Atomic Energy Agency, 1966, Preprint No. CN-23/107, 24p. *(In Russian)*. *(CONF-661014-27)*. DTIE.

From IAEA Conference on Nuclear Data, Paris.

The results are presented of measurements on the radiative capture of neutrons in the separated silver isotopes ^{107}Ag and ^{109}Ag in an energy range up to 1000 eV. A number of previously unreported levels were found. It is shown that the marked discrepancy between the level-spacing distribution previously reported for Ag and the Wigner distribution for various superposed level systems is only apparent. It is also shown that there is no correlation between ^{107}Ag and ^{109}Ag levels within the limits of statistical accuracy obtained. The values of the strength functions S_0 for ^{107}Ag and ^{109}Ag are found to be 0.43×10^{-4} and 0.83×10^{-4}, respectively. This marked difference in the strength function values of nuclei of almost the same atomic weight is not consistent with the optical model of the nucleus and can probably be explained on the hypothesis of compound-nucleus formation through three-quasi-particle interactions. (auth)

17351 (KFK-452) HIGH RESOLUTION CROSS-SECTION MEASUREMENT FOR SOME FAST REACTOR STRUCTURAL MATERIALS IN THE keV ENERGY RANGE. **Rohr, G.; Friedland, E.; Nebe, J.** (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Angewandte Kernphysik). Oct. 1966. 14p. (CONF-661014-47). Dep. mn.

From IAEA Conference on Nuclear Data, Paris, France.

Transmission measurements of the neutron total cross sections of Fe, ^{57}Fe, ^{55}Mn, and ^{51}V at 20 to 250 keV were made using the time-of-flight method. A multilevel analysis of the data for ^{55}Mn and ^{51}V yielded resonance parameters for these isotopes, and an area analysis of the data on Fe yielded resonance parameters for ^{56}Fe. (D.C.W.)

11950 MONTE CARLO CALCULATIONS OF RESONANCE INTEGRAL OF ^{232}Th. **Sehgal, Bal Raj** (Brookhaven National Lab., Upton, N. Y.). Nucl. Sci. Eng., 27: 95-103(Jan. 1967). (BNL-10210).

Resonance integral calculations are done for ^{232}Th infinite dilute, ^{232}Th metal rod, and $^{232}ThO_2$ rod systems. Doppler effect calculations are performed for $^{232}ThO_2$ rod systems for temperatures up to 2000°K. The resolved resonance integral for rod systems at each temperature is evaluated by Monte Carlo calculations and the resonance overlap effect between the two resonances of Th at 21.78 and 23.45 eV is taken into account. The unresolved s- and p-wave contributions were computed by standard methods. The data describing the resolved resonance parameters up to 3 keV (Γ_γ = 25.9 MeV) recommended in BNL-325 (Supplement No. 2, 1965) are used in these calculations. The p-wave strength function in the unresolved energy range is taken to be 1.63×10^{-4} $(eV)^{-1/2}$. The calculated resonance integrals and Doppler coefficients are compared with measurements; they are found to be in excellent agreement with each other. (auth)

50617 AVERAGED RESONANCE NEUTRON CAPTURE IN GOLD. **Allen, B. J.; Bird, J. R.** (Australian Atomic Energy Commission, Lucas Heights). Phys. Lett., 27B: 494-6(Sept. 16, 1968).

The intensities of resonance averaged γ rays from the capture of 10- to 60-keV neutrons in gold were measured with a Ge(Li) detector. Transition strengths to final states between 0.2 and 1.2 MeV have strong E1 reduced widths and provide evidence for a significant 4s–3p direct component in the capture mechanism. (auth)

35346 NEUTRON CAPTURE AND FISSION CROSS-SECTION DATA IN THE keV ENERGY RANGE. SOME NEW MEASUREMENTS AND RENORMALIZATIONS. Beckurts, K. H.; Kompe, D.; Menlove, H. O.; Poenitz, W. P. (Kernforschungszentrum, Karlsruhe, Ger.). pp 67-76 of Fast Reactor Physics. Vol. I. Vienna, International Atomic Energy Agency, 1968.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.1); CONF-671043-(Vol.1).

Some new measurements of capture cross sections were performed using the 3-MeV Karlsruhe pulsed Van de Graaff accelerator. A major effort was undertaken to measure the absolute gold capture cross section as a function of energy between 25 and 500 keV in order to establish a reliable standard for capture and fission cross-section renormalizations. To do this, a well-collimated pulsed beam of monoenergetic neutrons produced in a thin lithium target at 80° to the proton beam direction was used to irradiate a 0.1-cm thick sample in the center of a large liquid scintillator which detects the prompt capture γ rays. The relative neutron flux was measured with a "grey neutron detector" which is relatively insensitive to neutron energy variations in the energy range of these measurements. The gold cross-section shape was normalized at 30 keV to an absolute value fitted from various recent experiments. By a similar technique, the capture cross section of ^{238}U in the energy range 25 to 500 keV was determined. Furthermore, cross sections of several medium- and heavy-weight nuclei which may be of some interest to reactor design (Nb, Mo, Cs, Hf, Ta, W, Re) were determined by a time-of-flight method, using the gold value as a standard. The data are presented, and some implications of the new data on reactor calculations are reviewed. (auth)

48522 (LA-3917) A MULTILEVEL ANALYSIS OF THE ^{235}U FISSION CROSS SECTION. Cramer, James D. (Los Alamos Scientific Lab., N. Mex.). Apr. 16, 1968. Contract W-7405-eng-36. 9p. Dep. CFSTI.

Resonance parameters for the ^{235}U fission cross section, as measured on the Petrel experiment at the Nevada Test Site, were determined using a multilevel fitting program based on the Wigner-Eisenbud R-Matrix theory. (auth)

53168 (JAERI-1162) ON THE EVALUATION OF ^{239}Pu DATA IN THE keV AND RESOLVED RESONANCE REGION. Durston, C. (Atomic Energy Establishment, Winfrith (England)); Katsuragi, S. (Japan Atomic Energy Research Inst., Tokai. Tokai Research Establishment). June 1968. 10p. Dep.

The evaluation for ^{239}Pu cross sections in the keV region is described. The present evaluation gives a better interpretation to both the α value and the fission cross section. This evaluation confirms the applicability of the channel theory of fission. In the resolved resonance regions a fit to the cross section is obtained that is suitable for use with the GENEX code. For users convenience, the resonance parameters are listed. (auth)

50634 FISSION COMPONENTS IN ^{234}U RESONANCES. James, G. D.; Rae, E. R. (Atomic Energy Research Establishment, Harwell, Eng.). Nucl. Phys., A118: 313-20(1968).

The fission cross section of ^{234}U was measured in the energy range below 20 keV and shows three regions of strong fission yield with essentially no fission at the intervening energies. In the first group, which covers the energy range up to 740 eV, all the 20 resonances known to exist below 370 eV in the total cross section have a measurable fission width with an average $\langle\Gamma_f\rangle$ = 0.098 MeV. The second and third groups are centered at 8.33 keV and 13.9 keV and extend over about 1 keV. The area under the fission cross section curve $\sigma_f\Gamma_f$ is 97.7b · eV for the 29 resonances below 740 eV, 52 b · eV for the group at 8.33 keV and 32 b · eV for the group at 13.9 keV. An analysis of the distribution of fission widths for the 20 resonances below 370 eV shows that they fit a χ^2 distribution with ν = 1.39 ± 0.37 degrees of freedom. This evidence for a grouping of sub-threshold fission resonances is similar to that already found in ^{240}Pu and ^{237}Np and according to Weigmann and Lynn is a result of the existence of the second minimum in the fission potential barrier as predicted by Strutinsky. The well depth corresponding to a level density of 7 keV indicates that the second minimum lies at 3 MeV above the ground state of ^{234}U. (auth)

39637 (RPI-328-123, pp 1-17) NEUTRON CROSS SECTIONS. (Rensselaer Polytechnic Inst., Troy, N. Y.).

The average capture cross sections of W, ^{182}W, ^{183}W, ^{184}W, and ^{186}W in the energy range from 1 to 100 keV were determined. The p-wave strength functions for ^{54}Fe and ^{58}Ni were extracted from capture data, and the radiative width of the 4.6-keV resonance in ^{62}Ni was determined. The total cross section of ^{147}Pm was determined by transmission measurements on four $^{147}Pm_2O_3$ samples over the energy range from 0.008 to 200 eV; neutron resonance parameters were obtained for energies up to 120 eV (D.C.W.)

3723 NEUTRON RADIATIVE CAPTURE IN Mo. Weigmann, H.; Schmid, H. (Euratom, Geel, Belg.). Nucl. Phys., A104 513-24(1967).

Neutron radiative capture in natural Mo was studied in the neutron energy range from 10 eV to 25 keV. An area analysis was applied to part of the observed resonances which yields $g\Gamma_n$ for small resonances or Γ_γ for large resonances. Above 1 keV neutron energy, the observed capture rate was averaged to give the mean capture cross section which is in good agreement with earlier measurements. (auth)

V. NEUTRON CROSS SECTIONS

6. Doppler Effects

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38318 CRYSTALLINE EFFECTS ON THE DOPPLER-BROADENED CROSS SECTION AND RESONANCE INTEGRALS OF URANIUM IN A UO_2 LATTICE. Adkins, C. R. (Carnegie Inst. of Tech., Pittsburgh); Persiani, P. J.; Hwang, R. N. Trans. Amer. Nucl. Soc., 10: 228-9(June 1967).
From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

42611 (NAA-SR-12501) HIGH TEMPERATURE MEASUREMENTS OF DOPPLER EFFECT IN UC AND UO_2 BY METHOD OF TEMPERATURE-CYCLED REACTIVITY OSCILLATION. Beller, L. S.; Heneveld, W. H. (Atomics International, Canoga Park, Calif.). Aug. 1, 1967. Contract AT(04-3)-701. 33p. Dep. CFSTI.
Interim measurements of the Doppler effect in nominally 0.5-cm-diameter samples of natural uranium carbide and uranium dioxide are reported for temperatures up to 1330°C. The measurements were made as part of the investigation of the method of making Doppler-effect measurements by oscillating the temperature to produce a direct measurement of dI/dT. The measured ratio of specific reactivity effects for UC/UO_2 between 1030 and 1330°C was 1.23 ± 0.15, in satisfactory agreement with calculation. The Doppler effects in UO_2 are in agreement with extrapolations from data of others obtained at lower temperatures. The methods and equipment for making Doppler-effect measurements by this method are described. It is concluded that the method is entirely practical, producing results of precision comparable to that obtained by standard methods. Certain advantages of this method were used to identify quantitatively sources of interference not accessible to other methods. (auth)

35551 (ANL-7320, pp 334-40) DEPENDENCE OF THE DOPPLER COEFFICIENT OF REACTIVITY FOR HEAVY ELEMENTS ON CHEMICAL FORM, SURFACE-TO-MASS RATIO, AND NEUTRON SPECTRUM. Carpenter, S. G.; Mountford, L. A.; Springer, T. H.; Tuttle, R. J. (Atomics International, Canoga Park, Calif.).
A series of experiments has been carried out in a fast-neutron spectrum, characterized by a median fission energy of 62 keV, in order to measure the Doppler coefficient and other related temperature effects for a variety of materials which are of particular interest in fast power reactor technology. Special emphasis has been placed on the ^{235}U isotope which has been investigated in several chemical forms amenable to utilization in high-temperature, high-efficiency fuel elements. Changes in the size and chemical composition of samples of this isotope and other heavy-element isotopes have been made in order to evaluate the effects on the Doppler coefficient of changes in the surface-to-mass ratio and changes induced by the addition of C or O to form carbides and oxides. In addition, the effects of localized spectrum perturbation on the Doppler coefficient of Th have been studied by surrounding the sample with "blankets" consisting of heavy resonance absorbers, structural materials, and several types of scatterers, including Na. (auth)

35679 (ANL-7320, pp 345-9) RESULTS OF RECENT DOPPLER EXPERIMENTS IN ZPR-3. Gasidlo, J. M. (Argonne National Lab., Idaho Falls, Idaho).
The results of Doppler experiments in ZPR-3(Idaho) are presented. The experiments were done on two cores: one which served as a SEFOR mockup and the other which was for participation in the international comparison. Reactivity measurements due to change in materials and temperature are presented; Doppler effect measurement results are shown for both assemblies. (M.L.S.)

38684 ANALYSIS OF SMALL-SAMPLE DOPPLER-EFFECT MEASUREMENTS. Lewis, R. A.; Till, C. E. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 10: 274(June 1967).
From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

38685 VERIFICATION OF DOPPLER MEASUREMENTS IN ZONED FAST CRITICAL ASSEMBLIES. Lewis, R. A.; Till, E. F.; Groh, E. G.; LeSage, L.; Marshall, J. E. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 10: 273-4(June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

35552 (ANL-7320, pp 341-4) DOPPLER COEFFICIENT MEASUREMENTS IN THE CRYOGENIC TEMPERATURE REGION. Mountford, L. A. (Atomics International, Canoga Park, Calif.).

The Doppler-coefficient has been measured in the cryogenic temperature range. The inverse temperature dependence of the coefficient results in relatively larger reactivity changes at low temperature, although crystalline binding effects begin to be significant in this range. The effective-temperature model has been used to correct for binding effects. The power fluctuations, as a sample was oscillated in and out of the reactor, were Fourier-analyzed to give the reactivity effect of the sample. Different sample temperatures were obtained using a "Cryo-Tip" refrigerator, which uses Joule-Thomson expansion of N and H to cool the sample. Measurements are reported for Th and W in a 62-keV median-fission-energy spectrum. Also included are measurements at higher temperature to cover the temperature range from 100 to 1000°K. The measured temperature dependence of the Doppler effects agree well with calculated values, with the crystalline binding effects being evident in W. (auth)

10014 MEASUREMENTS AND CALCULATIONS OF THE DOPPLER EFFECT ON THE REACTIONS ^{238}U(n,γ), ^{235}U(n,f), AND ^{239}Pu(n,f) WITH NEUTRONS IN THE ENERGY RANGE 0 TO 35 keV. Perkin, J. L.; Fieldhouse, P.; Brickstock, A.; Davies, A. R. (Atomic Weapons Research Establishment, Aldermaston, Eng.). J. Nucl. Energy: Pt. A and B, 20: 921-37(Nov.-Dec. 1966).

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 symmetrical Sb - Be photoneutron source. The ^{238}U(n,γ) 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 spherical symmetry of the apparatus was chosen to minimize any effects of thermal expansion and to facilitate the comparison of the results obtained with those from a computer calculation based on a programme developed by Brissenden and Durston. In this program a statistical distribution of resonances based on resolved resonance data is tabulated over a fine energy mesh. This tabulation was used in a 4-region slowing down calculation resulting in a region-dependent neutron spectrum from which the particular reaction rates at various temperatures were obtained. (auth)

38373 MEASUREMENT OF THE DOPPLER EFFECT FOR ^{238}U CAPTURE AND ^{235}U FISSION IN A FAST NEUTRON SPECTRUM. Pflasterer, George Russell Jr. Stanford, Calif., Stanford Univ., 1966. 190p.

Thesis.

The Doppler effect in ^{238}U capture and ^{235}U fission was measured by means of a foil activation technique in the fast neutron spectrum core of the Mixed Spectrum Critical Assembly. Experimental results were obtained for two ^{238}U foil thicknesses and one ^{235}U foil thickness. The amount of scattering material between the foil and surrounding core fuel was varied to determine the effect on the Doppler measurement of change in the incident flux fine energy structure in the resonances. In this experiment only the foil is heated, while the core fuel remains at room temperature. The measured ^{238}U Doppler effect expressed as the ratio (change in foil activity with temperature/room temperature foil activity), R-1 was a factor of 2 higher than that calculated using a neutron energy spectrum derived from "nominal" material cross sections. Presently available cross sections in the energy range of interest are sufficiently uncertain so that it is possible to infer from them "hard" or "soft" neutron energy spectra such that the value of R-1 varies by a factor of two. The measured values for ^{238}U agreed quantitatively with those found from the "soft" neutron energy spectrum. Within the precision of the measurement no ^{235}U Doppler effect was observed. The calculated ^{235}U Doppler effect was smaller than the sensitivity of the experiment, thus within its precision (±0.002) the measurement confirms the theory. (Disser. Abstr.)

35695 (ANL-7320, pp 569-84) APPLICATION OF THE SEFOR CRITICAL EXPERIMENTS AT ZPR-3 TO SEFOR. Reynolds, A. B.; Stewart, S. L. (General Electric Co., San Jose, Calif. Advanced Products Operation).

A series of critical experiments was performed with a mockup of SEFOR at ZPR-3. Analyses of these experiments and the application of the results to the SEFOR design are discussed. Values of the critical mass were determined for 1-, 2-, and 3-segment SEFOR fuel designs in order to help establish the Pu atom fraction in the SEFOR fuel. Reactivity effects of axial fuel expansion were measured which led to selection of the 2-segment design for SEFOR. Measurements of the reactivity worth of the radial reflector established the adequacy of the SEFOR reflector-control system. The Doppler coefficient was measured. The calculated ^{238}U Doppler coefficient was in agreement with the experimental value; the measured ^{239}Pu contribution to the SEFOR Doppler coefficient was near zero. The maximum positive reactivity due to loss of sodium was measured. The ratio of prompt-neutron lifetime to delayed-neutron fraction was measured both by the pulsed neutron technique and by noise analysis. The values measured by the two techniques were in agreement. Fission ratios, fission and boron traverses, and plutonium worth distributions were measured and compared with calculations. A list of 17 references is included. (auth)

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35678 (ANL-7320, pp 319-33) ZPR-6 DOPPLER MEASUREMENTS AND COMPARISONS WITH THEORY. Till, C. E.; Lewis, R. A.; Hwang, R. N. (Argonne National Lab., Ill.).

The measurements reported were designed to study various aspects of the Doppler effect in a spectrum relevant to a fast power reactor. The method used was based on measuring the reactivity changes resulting from the in-pile heating of small samples in a zero-power assembly. After proof-tests of various kinds, the measurements concentrated on: the ^{238}U Doppler effect as a function of sample size and chemical form; the ^{235}U Doppler effect, with particular attention paid to evaluation of the extraneous effects of thermal expansion; the Doppler effect of mixtures of ^{235}U and ^{238}U; the effect of sodium voiding on the Doppler effect of ^{238}U and ^{235}U. A consistent intercomparison of the most pertinent results, and a comparison of the results with a consistent set of calculations using current theory are presented. A list of 14 references is included. (M.L.S.)

38730 DOPPLER COEFFICIENT TEMPERATURE DEPENDENCE AND THE EFFECT OF SODIUM VOIDING. Till, C. E.; Lewis, R. A.; Pond, R. B. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 10: 335 (June 1967).

From 13th Annual Meeting of the American Nuclear Society, San Diego, Calif., June 11-15, 1967. See CONF-670602.

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7920 (ANL-Trans-540) INVESTIGATION OF THE DOPPLER EFFECT IN ^{238}U IN A SHIELD MADE OF URANIUM OXIDE OF THE REACTOR BR-1. Abagyan, L. P.; Golubev, V. I.; Golyaev, N. D.; Zvonarev, A. V.; Koleganov, Yu. F.; Nikolaev, M. N.; Orlov, M. Yu. Translated from a paper presented at the IAEA Symposium on the Physics and Safety Problems of Fast Reactors, Karlsruhe, German Federal Republic, October 30-November 3, 1967. 14p. (CONF-671043-13). Dep. CFSTI. JCL \$1.10 fs, \$0.80 mf.

The Doppler effect in ^{238}U is studied by measuring the temperature increments of the effective resonance integral of absorption of thin ^{238}U-oxide sample placed in a large block of UO_2. The Doppler increment of the rate of reaction was measured from 300 to 2000°K. (D.C.C.)

35640 INVESTIGATION OF THE DOPPLER EFFECT IN ^{238}U IN THE OXIDE BLANKET OF THE BR-1 REACTOR. Abagyan, L. P.; Golubev, V. I.; Golyaev, N. D.; Zvonarev, A. V.; Koleganov, Yu. F.; Nikolaev, M. N.; Orlov, M. Yu. (Physical-Energy Inst., Obninsk, USSR). pp 317-27 of Fast Reactor Physics. Vol. II. Vienna, International Atomic Energy Agency, 1968. (In Russian). (ANL-Trans-540).

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.2); CONF-671043-(Vol.2).

Studies have been carried out on the Doppler increase in capture rate in thin samples of uranium-238 irradiated in a large block of depleted UO_2 at room temperature. The core of the BR-1 reactor was used as neutron source. The neutron spectrum at the point where the Doppler effect measurements were carried out was close to the asymptotic spectrum of the medium being studied. Measurements taken on this spectrum in the energy range below 10 keV using a set of resonance detectors showed satisfactory agreement with calculated data obtained from the system of constants evolved by V. V. Bondarenko, M. N. Nikolaev, L. P. Abagyan and N. O. Bazazyants. The reaction rate in a heated sample was measured in relation to that in a cold sample irradiated at the same position in the reactor. Gamma rays of energy 74 keV accompanied by beta decay of ^{239}U were recorded. Measurements were carried out in the temperature range 300 to 2000°K. The effect of the sample dimensions on the measurement results was established, as also was the effect of surrounding the sample by a layer of scattering material (heater material, container material, etc.). The accuracy of the results obtained was 5 to 6%. It is shown that in the temperature dependence of the proportional increase in the ^{238}U(n,γ) reaction rate in a small heated sample can be written in the form: $\Delta A/A\ (T_2) = b(\sqrt{T_g} - \sqrt{T_x})$ where T_x = 300°K and b = 0.0132 ± 0.0004 $(°K)^{-1/2}$. The experimental value for the Doppler increase in the ^{238}U(n,γ) reaction rate in a small sample at T_g = 800°K agrees well with the calculated results. The error in calculating the Doppler effect due to the inadequate knowledge of the neutron spectrum was not much greater than the error in the experimental data. (auth)

14027 THE DOPPLER EFFECT ON THE ^{238}U(n,γ) REACTION AT DIFFERENT NEUTRON ENERGIES AND AT HIGH TEMPERATURES. Fieldhouse, P.; Perkin, J. L.; Warner, G. P.; Brickstock, A.; Davies, A. R. (Atomic Weapons Research Establishment, Aldermaston, Eng.). J. Nucl. Energy. 21: 847-55(Nov. 1967).

Measurements of the Doppler effect on the ^{238}U(n,γ) reaction were made on uranium metal samples, heated to 770°K, with neutrons of different energies in the range 0 to 54 keV from the ^{7}Li(p,n) reaction. The measurements made in one particular energy range (0 to 10 keV) were extended to temperatures up to 1960°K using uranium dioxide samples. The results obtained were compared with computer calculations. In the uranium metal samples the Doppler effect at different neutron energies was found to agree, within the rather large experimental errors, with the theoretical calculations. In the uranium oxide samples the measured Doppler effect over the temperature range 290 to 1960°K agreed with that calculated, except for the first 200°K of this range where the measured effect was greater than that calculated. (auth) (UK)

33037 DOPPLER EFFECT MEASUREMENTS AND ANALYSIS IN ZPR-3 ASSEMBLY-50. Gasidlo, J. M.; Nicholson, R. B. (Argonne National Lab., Idaho Falls, Idaho). Trans. Amer. Nucl. Soc., 11: 240-1(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

33039 SENSITIVITY OF SMALL-SAMPLE DOPPLER EFFECT MEASUREMENTS TO ENVIRONMENT. Lewis, R. A.; Johnson, T. W. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 11: 242-3(June 1968).
From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

32790 RECENT STUDIES OF THE DOPPLER EFFECT IN HETEROGENEOUS SYSTEMS. Miller, L. B.; Hwang, R. N. (Argonne National Lab., Ill.). Trans. Amer. Nucl. Soc., 11: 206(June 1968).
From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

3706 ACTIVATION MEASUREMENT OF THE DOPPLER EFFECT FOR ^{238}U CAPTURE AND ^{235}U FISSION IN A FAST-NEUTRON SPECTRUM. Pflasterer, G. R., Jr. (General Electric Co., San Jose, Calif.); Sher, R. Nucl. Sci. Eng., 30: 374-94 (1967).
The Doppler effect in ^{238}U capture and ^{235}U fission was measured by means of a foil activation technique in the fast-neutron spectrum core of the Mixed Spectrum Critical Assembly. Experimental results were obtained for two ^{238}U foil thicknesses and one ^{235}U foil thicknesses. The amount of scattering material between the foil and surrounding core fuel was varied to determine the effect on the Doppler measurement of change in the incident flux fine-energy structure in the resonances. In this experiment, only the foil is heated, while the core fuel remains at room temperature. The experiment is analyzed by means of the collision-probability method which is used to develop and expression for the resonance integral of a thin absorber which is separated from a homogeneous reactor fuel region by a purely scattering medium. The general expression for the foil resonance integral is simplified, and numerical results are presented for the case in which the dominant resonances are weak; that is, for a fast reactor in which the 0.5- to 3.0-keV energy region dominates the ^{238}U Doppler effect. The measured ^{238}U Doppler effect expressed as the ratio R − 1 = (change in foil activity with temperature/room temperature foil activity) typically was of the order of 0.015 ± 0.002. This was a factor of 2 higher than that calculated using a neutron energy spectrum derived from "nominal" material cross sections. Presently available cross sections in the energy range of interest are sufficiently uncertain so that it is possible to infer from them "hard" or "soft" neutron energy spectra such that the value of R-1 varies by a factor of 2. The measured values for ^{238}U agreed quantitatively with those found from the "soft" neutron energy spectrum. Within the precision of the measurement no ^{235}U Doppler effect was observed. The calculated ^{235}U Doppler effect was smaller than the sensitivity of the experiment, thus, within its precision (±0.002), the measurement confirms the theory. (auth)

35626 ENERGY- AND TEMPERATURE-DEPENDENT CAPTURE MEASUREMENTS BELOW 30 keV SUPPORTING DOPPLER EFFECT CALCULATIONS. Seufert, H.; Stegemann, D. (Kernforschungszentrum, Karlsruhe, Ger.). pp 77-94 of Fast Reactor Physics. Vol. I. Vienna, International Atomic Energy Agency, 1968. (EURFNR-400).
From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.1); CONF-671043-(Vol.1).
Energy- and temperature-dependent capture measurements below 30-keV neutron energy were performed in natural uranium, tungsten, and tantalum using the slowing-down time spectrometer technique. The experimental set-up used for the experiments consists of a lead block of 1.3 m side length containing two experimental channels of 10 × 10 cm^2 cross-section. Into the first channel the target of a 14-MeV neutron generator is introduced, whereas the second channel is used for insertion of the heated samples. Pulses of 14-MeV neutrons, having a pulse width of about 1 μs, are used. The neutron energy is degraded first by inelastic collisions; afterwards only elastic collisions take place so that a specific relationship holds between mean neutron energy in the lead pile and the time after occurrence of the neutron pulse Because of this time-energy relation a time analysis procedure for the detector counts is applied. Because the energy range below 30-keV neutron energy is most interesting for Doppler-effect investigations the slowing-down time spectrometer is used to measure the capture ratios of hot-to-cold samples of natural uranium, tungsten, and tantalum. Thin samples were heated to different temperatures for this purpose, and the capture γ-rays were detected by proportional counters. Because hot-to-cold capture ratios are measured a knowledge of the neutron flux is not necessary; therefore, a direct comparison of calculated and measured temperature-dependent cross-sections is possible. A theoretical analysis of the experimental data for uranium is given. (auth)

32587 DOPPLER EFFECT IN ^{238}U AT INTERMEDIATE S/M VALUES. Sher, R.; Whitesel, R. N. (Stanford Univ., Calif.). Trans. Amer. Nucl. Soc., 11: 184-5(June 1968).
From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

5744 MEASUREMENTS OF THE TEMPERATURE COEFFICIENT OF REACTIVITY OF ^{235}U, ^{239}Pu, AND HIGHER Pu ISOTOPES. Springer, T. H.; Tuttle, R. J.; Mountford, L. A. (Atomics International, Canoga Park, Calif.). Trans. Amer. Nucl. Soc., 10: 562-3(Nov. 1967).
From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

33038 VARIATIONS IN THE FAST-SPECTRUM DOPPLER COEFFICIENTS OF Th AND ^{238}U AS A FUNCTION OF THE TYPE AND TEMPERATURE OF THE ENVIRONMENT. Springer, T. H.; Tuttle, R. J. (Atomics International, Canoga Park, Calif.). Trans. Amer. Nucl. Soc., 11: 241-2(June 1968).

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32562 . (AE-314) ACTIVATION DOPPLER MEASUREMENTS ON ^{238}U AND ^{235}U IN SOME FAST REACTOR SPECTRA. Tiren, L. I.; Gustafsson, I. (Aktiebolaget Atomenergi, Stockholm (Sweden)). Mar. 1968. 40p. Dep.

Measurements of the Doppler effect in ^{238}U capture and ^{235}U fission were made by activation technique in three different neutron spectra in the fast critical assembly FR-O. The experiments involved irradiation of thin uranium metal foils or UO_2 disks, which were heated in a small oven placed in the core center. The measurements on ^{238}U were extended to 1780°K and on ^{235}U to 1470°K. A core region surrounding the oven was homogenized to facilitate the interpretation of results. The reaction rates in the uranium samples were detected by gamma counting. The experimental method was checked with regard to systematic errors by irradiations in a thermal spectrum. The data obtained for ^{238}U capture were corrected for the effect of neutron collisions in the oven wall, and were extrapolated to zero sample thickness. In the softest spectrum (core 5) a Doppler effect (relative increase in capture rate) of 0.260 ± 0.018 was obtained on heating from 343 to 1780°K, and in the hardest spectrum (core 3) the corresponding value was 0.030 ± 0.003. An appreciable Doppler effect in ^{235}U fission was obtained only in the softest spectrum, in which the measured increase in fission rate on heating from 320 to 1470°K was 0.007 ± 0.003. (auth) (Sweden)

35651 STUDIES OF THE DOPPLER COEFFICIENT AND THE REACTIVITY EFFECT OF POLYTHENE IN SOME FAST REACTOR SPECTRA. Tiren, L. I.; Gustafsson, I.; Hellstrand, E.; Hakansson, R. (AB Atomenergi, Studsvik, Sweden). pp 143-70 of Fast Reactor Physics. Vol. II. Vienna, International Atomic Energy Agency, 1968.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.2); CONF-671043-(Vol.2).

The Doppler broadening of the ^{238}U capture and the ^{235}U fission cross-sections in fast reactor spectra has been studied by irradiating heated foils or plates of the isotopes of interest in a small furnace at the center of the zero-energy fast reactor FRO. The experiments have been applied to cores fueled with 20% enriched uranium and diluted with graphite and polythene. Very thin plates of the different core materials have been used in a region surrounding the furnace in order to minimize heterogeneity effects. The measured quantity is the difference in the induced γ-activity of samples irradiated at ambient temperature and at temperatures up to 1500 C. The effects of varying sample thickness and of scattering in the furnace wall have also been studied. Measurements of the reactivity effect of polythene have been made in two FRO assemblies. The neutron spectra of the cores were broadly similar to those of current steam-cooled fast reactor concepts. The experiments include a study of the spatial distribution of the reactivity coefficient of polythene. The results are in reasonable agreement with calculated values. The latter are sensitive to small changes in the absorption cross-section in the low neutron energy range. Most of the calculations have been made with a one-dimensional diffusion theory program using 16 energy groups but a two-dimensional code and a transport theory code have also been used. Additional measurements have been made on vertical polythene rods, 2.2 and 4.6 cm thick, inserted in the central fuel element. The measured distributions of the fission rate in ^{235}U and the capture rates in ^{55}Mn, ^{115}In and ^{197}Au inside and around the rods have been compared with results of multigroup calculations. (auth)

VI. LABORATORY SUMMARY REPORTS AND MISCELLANEOUS

1967

23534 (ANL-7246) PHYSICS DIVISION SUMMARY REPORT. Annual Review, April 1, 1965–March 31, 1966. (Argonne National Lab., Ill.). Contract W-31-109-eng-38. 180p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

7856 (ANL-7255) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, SEPTEMBER 1966. (Argonne National Lab., Ill.). Oct. 26, 1966. Contract W-31-109-eng-38. 101p. Dep. mn. CFSTI $4.00 cy, $0.75 mn.

Summaries of progress are presented concerning the EBWR Pu recycle program, EBR-II operation and development, physics experiments in ZPR-3 and ZPPR, burnup measurements for fast reactors, Na technology, fuel development and processing, reactor physics development, fuels and cladding development and fabrication, heat transfer and fluid flow studies, mechanics of materials studies, fluoride volatility process development, liquid metal direct conversion generator research, AARR design and development, and research on nuclear safety. (J.R.D.)

12231 (ANL-7267) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, OCTOBER 1966. (Argonne National Lab., Ill.). Nov. 22, 1966. Contract W-31-109-eng-38. 98p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Measurements of the uniform temperature coefficient of reactivity of the fully loaded EBWR core with eight spike assemblies in the first shim zone between room temperature and 360°F were made with boric acid concentrations near 6 g/gal. The temperature coefficient is positive at low temperatures, becomes very small as the temperature is raised, and apparently becomes negative before 360°F is reached. With planned modifications and maintenance on EBR-II completed during the recent scheduled shutdown, Run No. 22, projected for 1000 MWd, was begun on October 21. Almost half of this run was completed at month's end. Thirty-two of 54 total piling foundation holes for the ZPPR were completed. The floor slab and foundation for the vault-workroom and service floor of the support wing were poured. Detailed reports of measurements made in ZPR-3 during the last four months on Assembly 48, a large, clean, plutonium-fueled core, were compiled and a summary is presented. The AARR Title I report was submitted to the AEC for approval on October 7, 1966. The AARR core design was changed to a HFIR type, and all research and development work on the stainless steel cermet fuel was terminated. (auth)

13946 (ANL-7279) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, NOVEMBER 1966. (Argonne National Lab., Ill.). Dec. 21, 1966. Contract W-31-109-eng-38. 98p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Items of significant technical progress which have occurred in both the specific reactor projects and the general engineering research and development programs are summarized. Program activities are reported under five broad categories: Plutonium Utilization, Liquid-Metal Fast Breeder Reactors, General Reactor Technology, Advanced Systems Research and Development, and Nuclear Safety. The Experimental Boiling Water Reactor Pu recycle program is discussed under Plutonium Utilization. Liquid-Metal Fast Breeder Reactors includes the Experimental Breeder Reactor-II operations, fast zero-power assembly development, and fast reactor physics, components, and fuels development. General Reactor Technology examines applied and reactor physics, fuels and cladding, engineering, chemistry, and chemical separations. The Argonne Advanced Research Reactor development is summarized under Advanced Systems Research and Development. Nuclear Safety activities include coolant dynamics, fuel meltdown, materials behavior, energy transfer, Pu volatility studies, and TREAT operations. (H.D.R.)

13947 (ANL-7286) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, DECEMBER 1966. (Argonne National Lab., Ill.). Jan. 26, 1967. Contract W-31-109-eng-38. 92p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Items of significant technical progress which have occurred in both the specific reactor projects and the general engineering research and development programs are summarized. Program activities are reported under five broad categories: Plutonium Utilization, Liquid-Metal Fast Breeder Reactors, General Reactor Technology, Advanced Systems Research and Development, and Nuclear Safety. The Experimental Boiling Water Reactor Pu recycle program is discussed under Plutonium Utilization. Liquid-Metal Fast Breeder Reactors includes the Experimental Breeder Reactor-II operations, fast zero-power assembly development, and fast reactor physics, components, and fuels development. General Reactor Technology examines applied and reactor physics, fuels and cladding, engineering, chemistry, and chemical separations. The Argonne Advanced Research Reactor development is summarized under Advanced Systems Research and Development. Nuclear Safety activities include coolant dynamics, fuel meltdown, accident analyses, energy transfer, engineering safeguards systems, and TREAT operations. (H.D.R.)

21755 (ANL-7302) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JANUARY 1967. (Argonne National Lab., Ill.). Feb. 24, 1967. Contract W-31-109-eng-38. 106p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

EBWR has been operating at a steady power of 42 MW since November 11, 1966. The reactor will be shut down in February 1967 so that special fuel pins can be removed for isotope analysis. In preparation for Run No. 25, the EBR-II core size was increased from 81 to 91 subassemblies. Since the enlarged core has moved additional driver fuel out to Row 6 locations, the question of relatively high gamma heating in blanket Rows 7 and 8 has assumed new importance. Therefore, detailed calculations of expected temperatures and examination of blanket elements and their contained depleted U which have achieved varying degrees of burnup are being made before initiating Run No. 25. In addition to routine processing and fabrication of fuel subassemblies at the Fuel Cycle Facility an Argon Cell electromechanical manipulator was repaired in-cell, with special tools, and an Air Cell crane trolley which was transferred to the cell roof enclosure for repair. The Mark-IB modified fuel restrainer was tested with a new thinner shank. Fabrication of 3 experimental subassemblies, 7 special subassemblies with the materials therein controlled for later evaluation, and 5 subassemblies of the half-worth type for use in making reactivity adjustments in the reactor are also reported. Experiments were continued in ZPR-3 Assembly 48, a Pu-fueled critical assembly with a distribution of materials similar to a large carbide reactor. This assembly is being used to make measurements relevant to the FFTF design. Heterogeneity measurements were made using thin ^{235}U, ^{238}U, and manganese foils in several configurations. Relative ^{238}U capture rate was measured at seven positions near the core midplane, using a method developed at ZPR-3. Excavation of the ZPPR reactor pit is complete. The six reactor pit posts have been steel capped and the concrete tops for them have been poured. The AFSR reactor pit and room foundations have been poured. Placement of structural steel for the support wing is 95% complete. The precast concrete floor of the support wing is in place and all walls and support columns of the vault, workroom, and inside equipment rooms have been poured. A supplement to the AARR Preliminary Safety Analysis Report was completed. (J.R.D.)

23589 (ANL-7308) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, FEBRUARY 1967. (Argonne National Lab., Ill.). Mar. 28, 1967. Contract W-31-109-eng-38. 106p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Regular and special fuel pins were removed from the EBWR for analysis of specific isotopes. The fuel in the plutonium zone had an average exposure of 1300 MWd/tonne. A detailed study to determine swelling in the uranium and correlation of this to exposure time and temperature was made. Neutron flux wire irradiation was conducted at low powers to obtain distribution rates of fission and plutonium production in the radial blanket. Radial blanket subassemblies of the Fuel Cycle Facility reactor were examined for external and internal change in physical properties. Foil irradiations were used to measure the relative ^{238}U capture and fission rates and ^{235}U fission rates in ZPR-3. Further studies were also made to determine the effect of the distortion of the flux distribution on the linearity of k_{eff} subcritical measurements in reactors with a softened neutron spectrum. In ZPR-6 the effects of Na voiding on the fission densities in ^{235}U and ^{238}U and the capture density of ^{238}U were investigated by foil measurements. Material sample worths and Doppler coefficients were measured for ^{238}U and ^{235}U in the ZPR-9; these results are listed in tabular form. Construction activities are detailed for the ZPPR. Three systems of fission-gas pressure transducers for the FFTF are discussed. Failed fuel detection methods, sensor leads for fluid signals, in-core flowmeters, and fuel-pins are some component systems which are briefly discussed. Irradiation of U–Pu alloy fuels is discussed; ceramic, cermet, and mixed-carbide fuel irradiation is also discussed. A method for the preparation of (U,Pu) C fuel is given. Development of refractory-metal alloys for service in oxygen contaminated systems is detailed; corrosion of jacket and structural materials by Na in the reactor is examined and results presented. Microhardness tests were made on Ni-base alloys; results are presented. A relatively detailed outline for fast reactor fuels is given. In the area of general research and development the following topics are discussed: fast reactor core-parameter study, uses of ^{235}U in a fast-reactor era, neutronic investigations, thermal design studies, and fuel cycle costs. An oscilloscope display for an on-line computer is described; changes in the cross section computer programs are given; reactor parameter calculations are discussed in some detail; fuels and materials development are discussed at length; radiation damage to structural materials is outlined; techniques for fabrication and testing of fuels are presented. Various phases of engineering development are discussed in some detail. Results of corrosion studies for the Argonne Advanced Research Reactor secondary coolant loop are tabulated. A general discussion of the reactor physics measurements and experiments in the AARR is given. A survey of work done in nuclear safety is given; the following subjects are covered: linkage of heat transfer and two-phase coolant flow nodules, Na expulsion, superheat, critical flow, electron-bombardment heater tests, convective instability, thermophysical properties of Na, component dynamics, transient in pile loop experiment with EBR-II Mark I fuels, summary and analysis of single-pin results, seven-pin-cluster Na loop experiments, vacuum-inert gas glovebox, TREAT operations, metal-water reactions, and Pu volatility safety. (M.L.S.)

30093 (ANL-7317) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MARCH 1967. (Argonne National Lab., Ill.). Apr. 28, 1967. Contract W-31-109-eng-38. 124p. Dep. CFSTI.

The EBWR was operated at the maximum permissible power level of 70 MW. During the approach to power, pile-oscillator measurements of the reactor transfer function were made at 28, 38, 49 and 57 MW. In preparation for Run No. 25, several zero-power reactor runs were made to obtain measurements of reactivity worth of the stainless steel subassemblies in Rows 7 and 8, which have replaced subassemblies of depleted U. An extensive program of analysis was initiated to determine the sources and extent of Cu found in the reactor primary Na system. Removal of the Cu electrodes associated with the auxiliary primary pump showed that the exposed Cu ends were severely pitted and eroded and had indeed lost Cu to the primary Na. The exposed Cu ends of the electrodes were sheathed in stainless steel and the electrodes were reinstalled. The pump was checked out and is operating satisfactorily. Analysis for further sources of Cu in the primary system is continuing and results are being evaluated. Postirradiation examination of radial-blanket subassemblies from the 7th, 8th, 9th, and 12th rows of the reactor revealed that length, diametral, and density increases were greatest for the innermost rows, decreasing along the radius from the core, with no significant increases noted for the 9th or 12th row subassemblies. Concrete for the ZPPR cell floor and pit and for the blanket storage room floor was poured. Experiments are in progress with ZPR-3 Assembly 48B, a reactor with a two-zoned Pu core and with a 12 × 15-in. central region that contains Pu with 22% ^{240}Pu substituted for 4.5% ^{240}Pu. Critical mass was determined after control-rod calibrations were made as well as measurements of the worth of core-edge material and of fuel spiking of the safety rods. In the high-Pu-content central region, measurements were made of Na substitutions, fission ratios, perturbation reactivity measurements with small samples, and of fine flux variations across the cell. ^{235}U fission ratios were measured versus ^{233}U, ^{234}U, ^{236}U, ^{238}U, ^{239}Pu, and ^{240}Pu. Reaction rate traverses were made in a radial direction at the core midplane and reactivity traverses have been made in the radial direction using stainless steel, Ta, ^{10}B, ^{235}U, ^{239}Pu, and ^{238}U. Work on an improved proton recoil neutron spectrometer is also reported. Work done to define more narrowly the limits of precision of ZPR-3 experiments by refinement of measurements related to the ZPR-3 gap interface is reported. (J.R.D.)

34076 (ANL-7342) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 1967. (Argonne National Lab., Ill.). Contract W-31-109-eng-38. 139p. Dep. CFSTI.

Summaries of progress are presented concerning the EBWR Pu recycle program, EBR-II operation and shutdown, physics experiments in ZPR-3, development of ZPPR and AARR, burnup measurements for fast reactors, Na technology, fuel development and processing, reactor physics development, fuels and cladding development and fabrication, heat and fluid flow, fluoride volatility process development, liquid metal direct conversion generator research, and research on nuclear safety. (J.C.W.)

44932 (ANL-7371) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, AUGUST 1967. (Argonne National Lab., Ill.). Sept. 28, 1967. Contract W-31-109-eng-38. 126p. Dep. CFSTI.

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REACTIVITY—measurements in EBR-II core, effects of power level on; perturbation measurements in ZPR-3, use of small samples for central

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CRITICAL ASSEMBLIES—fission ratio measurements for uranium isotopes in ZPR-3, central; neutron spectrum in ZPR-6, effects of sodium voiding on; fission ratios for uranium and neptunium in ZPR-9, measurements of; pressure vessel for ZPPR, safety factor calculations for

FISSION—ratio measurements in ZPR-3 of uranium 233/uranium 235; ratio measurements in ZPR-3 of uranium 234/uranium 235; ratio measurements in ZPR-3 of uranium 236/uranium 235; ratios for uranium and neptunium, measurements in ZPR-9 of

URANIUM ISOTOPES U-238—foil activations in ZPR-3; fission ratios for, measurements in ZPR-9 of

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NEPTUNIUM ISOTOPES Np-237—fission ratios for, measurements in ZPR-9 of

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30091 (ANL-7210) REACTOR PHYSICS DIVISION ANNUAL REPORT, JULY 1, 1965 TO JUNE 30, 1966. (Argonne National Lab., Ill.). Dec. 1966. Contract W-31-109-eng-38. 446p. Dep. CFSTI.

Brief descriptions of work on the following subjects are presented: fission properties and cross section data including fast neutron scattering studies, elastic neutron scattering from elements of intermediate weight, elastic neutron scattering from Mg and Si, fast neutron scattering from nuclei in the mass region A = 95-130, the interaction of fast neutrons with the 182, 184, and 186 isotopes of W, a search for fluctuations in the fission cross section of ^{235}U, neutron flux measurements in the 10-200 keV region, (d,n) stripping reactions, fast neutron total cross sections using a monoenergetic source and an automatic facility, fast neutron energy degradation through the (n,γn') process, unitary models of nuclear resonance reactions, the ^{252}Cf fission neutron spectrum from 0.003-15.0 MeV, direct and absolute measurements of average yield of neutrons in the thermal fission of ^{235}U and spontaneous fission of ^{252}Cf, spontaneous fission half-lives of ^{242}Cm and ^{244}O Cm; thermal reactor physics including High Conversion (HI-C) critical experiment, HI-C uniform lattice calculations; initial critical experiments of the EBWR Pu recycle program, measurement of capture-to-fission ratios of ^{239}Pu and ^{241}Pu in the Pu loading of the EBWR, control rod evaluation for thermal and intermediate reactors, small reactivity measurements in the Argonne Thermal Source Reactor (ASTR), neutron beam spectra extracted from the High Flux Irradiation Reactor, Argonne Advanced Research Reactor (AARR) critical experiments-preface, AARR critical experiments-control blade worths, AARR critical experiments-prompt neutron lifetime measurements by the Rossi-alpha technique, AARR critical experiments-Cd ratio measurements, AARR critical experiments-activation and power distribution measurements, AARR critical experiments-void and material reactivity worths and temperature coefficients, AARR critical experiments-beam tube experiments, AARR critical experiments-startup source requirements and instrument response, AARR calculations-preface, AARR calculations-analysis of the critical experiments, AARR calculations-general reactor physics design analysis, AARR calculations-reactor physics characteristics of the ITC, AARR calculations-factors in optimization of experimental fluxes, AARR calculations-shield design analysis, AARR calculations-analyses of hypothetical accidents; fast reactor physics including the neutron energy spectrum in a dilute UC-fueled fast critical assembly, neutron spectra in depleted U, calculations of Na-void coefficients in large fast neutron carbide cores in assemblies No. 2 and 3 of ZPR-6, calculations of the effect of thin slab heterogeneities on the non-leakage reactivity component of Na voiding, non-linearity in the spectral component of Na void effect as a function of Na content, effect of parameter uncertainties on Na void effect and critical size of fast reactors, Doppler-effect measurements on a dilute carbide fast assembly-ZPR-6 assembly No. 4Z, measured physics parameters in a zoned fast UC core-ZPR-IV assembly No. 42, analysis of the uncertainties in the interpretation of zone loaded experiments, measurement of the spatial distribution of the importance of fission neutrons in ZPR-6 assembly No. 4Z, standard deviation of ion chamber current measurements in ZPR-6 assembly 4Z, measured reactivity removal rates in ZPR-6 assembly No. 4Z, the Argonne National Laboratory of ZPR-3 assembly No. 48, critical assembly comparison calculations using new cross section data, comparative neutronic characteristics of metal, oxide and carbide EBR-II driver fuels, the effect of fuel and blanket changes on the EBR-II flux, FARET Core I fuel irradiation program and reference design, twenty-six group cross section set for W-based rocket systems, physics measurements in fast W rocket reactor critical experiments, measurement of space-dependent material worths in several ZPR-9 assemblies, rocket critical assemblies analysis, physics measurements in an operating fast breeder power reactor, further neutronic studies of the 1000 MW(e) metal-fueled fast breeder reactor, reactor physics calculations for a 10,000 MW(th) fast Na-cooled breeder, fast breeder reactors for water desalting, criteria for the density of monitoring points in large reactors; fast reactor safety including capabilities of the present TREAT facility core as a fast flux loop meltdown facility, meltdown experiments using the Mark I integral Na loop, analyses of single pin loop meltdown experiments, properties of irradiated UO_2 pins prior to TREAT facility transients, photographic fast reactor safety experiments on irradiated oxide pins at the TREAT facility, transient in-pile tests on UO_2-W cermet rocket fuel samples, design of the Mark II integral TREAT facility Na loop, calibration mockup for the large loop test section for the TREAT facility, transient response of stand-off pressure transducer assemblies on the TREAT facility integral Na loops, extensible multi-purpose vacuum glove box, experimental results and improvements in the fast neutron hodoscope, the exact three-dimensional solution for thermoelastic stresses and displacements in finite and infinite tubes, transient vaporization of Na in reactor coolant channels, convective heat or mass transfer with phase changes, theoretical prediction of thermodynamic and transport properties of metal vapors, equation of state of reactor materials at high pressures and temperatures, a modified equation of state for hydrodynamic calculations in the AX-1 numerical program, properties of refractory ceramics at extremely high temperatures (UC liquid expansion), modification of the high temperature W filament furnace, failure pressures of thick-walled doubly-reinforced concrete cells, maximum permissible body burdens of Pu isotopes and resulting release criteria, fast reactor meltdown accident analysis code, PREAX; experimental physics techniques and facilities including a low geometry α counting chamber, absolute determination of fission rates in ^{235}U and ^{238}U and capture rates in ^{238}U by radiochemical techniques, precision fission rate measurements by fission track counting, solid-state Compton spectrometer for measurement of reactor γ spectra, feedback stabilization of nuclear counting channels, signal splitting into fast and slow channels, design and construction of an improved Mn bath counting system, low flux measurement of ^{235}U epicadmium capture-to-fission ratio, reactor response to an oscillating neutron source, neutron fluxes required for activating probe materials, a code to permit fission product decay corrections without the use of a reference foil, determination of the k-constant for the Dy substitution method, additional calculations of the activation of spheres in a nonisotropic neutron flux, use of a small digital computer in data analysis and control of critical facilities, a Ge(Li) detector system for the measurement of γ-rays following inelastic neutron scattering, a multi-angle fast neutron time-of-flight system, multiple angle detector apparatus for neutron elastic scattering and polarisation measurements, multiple scattering correction, automated computed control of a fast neutron laboratory; reactor computation methods and theory including the Argonne Reactor Computation (ARC) system, the ARC system glossary, the Multigroup Constants Code (MC^2), modification of THERMOS to generate transfer cross sections, generation of multigroup cross sections using a coupled MC²–THERMOS code, variation of thermal cross sections with buckling in consistent P1 and B1 calculations, development of a code to study fuel management, AMC–A Monte Carlo code, development and analysis of Monte Carlo methods, quasistatic treatment of space dependent reactor transients, space dependent kinetic calculations using the WIGLE code, reactor systems analysis and hybrid computers, computation of the coupled error function by continued fractions, treatment of source discontinuities in the solution of the diffusion equation, revision of the bulk shielding code MAC for the CDC-3600 computer, codes for analysis of elastic scattering angular distributions, multilevel cross sections for a fissionable isotope, the effect of interference on the resonance integral mixtures of Th and U, the effect of randomness on group cross sections, the chemical binding effects on the resonance line shapes of ^{238}U in a UO_2 lattice, equivalence between homogeneous and heterogeneous resonance integrals in cylindrical geometry, effect of the fluctuations in collision density on fast reactor Doppler effect calculations, an approximate calculation of space dependent flux using a variational principle, neutron-wave analysis; miscellaneous including energy spectrum of fast cosmic-ray neutrons near sea level, a CO_2 system for direct conversion of nuclear energy to coherent laser light, theory of plasma oscillations–generation of thermionic RF energy and interactions with DC, circulating shield reactor for space power, and improvised shutter design for the JANUS reactor. A total of 609 references is listed throughout the report. (M.L.S.)

46938 (BNWL-501) FAST FLUX TEST FACILITY. Quarterly Progress Report, January–March 1967. Astley, E. R. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Apr. 1967. Contract AT(45-1)-1830. 202p. Dep. CFSTI.

FAST FLUX TEST FACILITY—experiment facilities for, design of; designs for, evaluation of various; reactor core configuration for, evaluation of split wedge and conical; electrical supply systems for, schematics for external; heat removal systems for, process design conditions and construction costs for; computer simulations of, digital and analog; coolant circulation in, analog simulation of natural; structural materials for, radiation effects on; materials properties handbook for
HEAT TRANSFER—heat removal systems for FFTF, evaluation of
LOOPS—design of closed test, layout studies for
REACTOR FUEL ELEMENTS—remote handling of FFTF, evaluation of gas-cooling system for; examination facilities for, design of FFTF; thermal analysis of coolant flow blockage in FFTF; cladding for FTR, radiation effects on
REACTOR EXPERIMENTAL FACILITIES—design parameters for FFTF; heat transfer analyses for FFTF rabbit capsules
FAST TEST REACTOR—core for, parametric studies on; cooling systems for, analysis of; tube materials for, stress analysis for; core for, hot channel factor determination for; core for, radioisotope transport rate in driver
REACTOR CONTROL SYSTEMS—design for FFTF
REACTOR FUELS—radiation testing of FFTF; neutron flux and power peaking factors in FFTF, parametric analyses of; burnup effects on FFTF, safety analysis for; cladding for fast, analysis of
NEUTRONS—flux distributions in FFTF, evaluation of postulated effects of skewed
THERMOCOUPLES—time response of ungrounded, analysis of
REACTOR SAFETY—scram requirements for single channel fuel meltdown
REACTOR MODERATORS—effects on FFTF neutron physics parameters of beryllium oxide
URANIUM ISOTOPES U-238—Doppler coefficients for, effects of crystalline binding on
REACTIVITY—worths of control rods for FFTF, two-dimensional transport models for calculation of
REACTOR CONTROL ELEMENTS—reactivity worths of FFTF, two-dimensional transport models for calculation of
SHIELDING—calculations for FFTF

7885 (NAA-SR-12175) AEC UNCLASSIFIED PROGRAMS. Quarterly Technical Progress Report, July–September 1966. (Atomics International, Canoga Park, Calif.). Nov. 25, 1966. Contract AT(11-1)-Gen-8. 173p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Developments are reported for studies on: stability and properties of cladding and structural alloys in a high temperature, liquid metal, fast neutron environment; boiling studies for sodium reactor safety; fast spectrum Doppler measurements; Na component development; sodium reactor experiment operation; Piqua development; reactor physics; reactor fuels and materials; high temperature reactor fuels and materials; Na chemistry; reactor safety; fission product and contamination control; characterization of Na fires and fission product release; high temperature and radiation chemistry; electronic structure of metals and alloys; and radiation damage in crystalline solids. (P.C.H.)

40564 (NAA-SR-12395) AEC UNCLASSIFIED PROGRAMS QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY–MARCH 1967. (Atomics International, Canoga Park, Calif.). May 31, 1967. Contract AT(04-3)-701. 184p. Dep. CFSTI.

Developments are reported for studies on: cross section analysis; integral reactor-physics experiments; basic-theory improvement; fast spectrum Doppler measurements; boiling studies for sodium reactor safety; measurement of Doppler coefficients; reactor dynamics simulator development; fission product contamination control; characterization of Na fires and fission product release; fission effects in metal fuels; Sodium Reactor Experiment operation; refractory moderator materials; FBR fuel development, fuel cladding, and structural materials; coolant chemistry; materials management; mechanical properties; dynamic loop testing; modes of failure and He degradation; basic properties of UC type fuels; Na chemistry; HNPF retirement; PNPF fuel elements; high temperature chemistry; radiation chemistry; electronic structure of metals and alloys; radiation damage in crystalline solids; and radiation chemistry of chromosomes. (P.C.H.)

23711 (BNWL-340) REACTOR PHYSICS DEPARTMENT TECHNICAL ACTIVITIES QUARTERLY REPORT; JULY–SEPTEMBER 1966. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Oct. 15, 1966. Contract AT(45-1)-1830. 72p. CFSTI $3.00 cy, $0.65 mn.

A recent revision in HRG (Hanford Revised Gam), a neutron slowing down code, modifies the calculation of the effective resonance integral of individual resonance by using for normalization an approximation to the flux used in the resonance integral calculation itself rather than a 1/E flux. A calculation study of thermalization within a simple cell has been made using the RBU Monte Carlo, THERMOS, and Program S-XIII codes. A comparison of the results shows the RBU Monte Carlo thermalization routine is formulated correctly and free from detectable numerical error. Experiments have been conducted in the PRCF using 2 wt % PuO_2-UO_2 fuel in H_2O. These experiments are directed towards determining the physics properties of Pu-fueled H_2O moderated reactor systems. Results are given for a two-zone critical experiment. Also given are results of measurements in a single-zone loading of fuel which contains 24 wt % ^{240}Pu. A calculational study of thermalization in PuO_2-UO_2 fueled H_2O moderated lattices has been performed. The purpose of the study is to determine whether errors are incurred in making assumptions pertaining to scattering processes, boundary conditions, and the source of thermal neutrons. Measurements have been made in the PCTR with a mixed lattice of Pu–Al fuel and thoria targets in alternating cells. New techniques were developed to adapt the PCTR type of measurement to this lattice array of supercells. The results include values of k_∞ for the array with and without water coolant surrounding the fuel and targets. Neutron activation of gold foils in the thermal column of the PCTR have been made to measure the relative neutron flux intensities for various thermal column conditions. The addition of a polyethylene reflector to the graphite stack was found to improve the thermal neutron intensity and to reduce the fast neutron component of the total flux. The presence of a small cavity in the center of the thermal column did not appreciably reduce the neutron flux gradient in the standard foil irradiation position. Neutronics calculations were performed for an 800 liter, PuO_2-UO_2 FTR "reference" core. Principal physics statistics and kinetics parameters were determined. To assess the accuracy of the present cross-section set in use for design calculations, numerous critical assembly results have been analyzed. In general, the computed fissile fuel fission ratios are in reasonable agreement with experiment, whereas the computed $^{238}U/^{235}U$ fission ratios are consistently higher than experimental values. A new group cross-section "collapsing algorithm" has been devised which makes use of a pseudo absorption cross section in each coarse group. The use of such "perturbation" cross sections is much more accurate for computing moderator reactivity coefficients than corresponding models using flux weighted sections. The removal-diffusion code MAC has been used to compare four different axial shield arrangements for the FTR. Energy and spatial distributions of both neutron and gamma fluxes have been obtained. Basic shielding requirements for a number of FTR fuel processing and handling facilities have been specified. Subcritical neutron interaction experiments were performed with bare and Plexiglas reflected arrays of bottles containing aqueous solution of ^{233}U. The critical numbers of bottles in arrays were determined for various spacings. In addition, the effects of internal moderation brought about by positioning Lucite sheet in various thicknesses between the bottles comprising several arrays were studied. The concentration of U in the experiments was 330 g ^{233}U/liter. The experiments provide data for nuclear safety guidance in handling, storage, and shipment of the material and for checking interaction calculations. A series of criticality experiments were begun, using a vessel of unique design, for determining bare and reflected critical thicknesses of plutonium solution in slab geometry. The slab vessel was designed to permit adjustment of its thickness over the range 3 to 9 in. Criticality experiments were performed, bare and water reflected, in the thickness range 4.5 to 6.5 in. with $Pu(NO_3)_4$ solution containing ~58 g Pu/liter with an acid molarity of 2.3. Infinite slab thicknesses are estimated from the measurement data. To permit studying the fundamental behavior of neutrons in Pu-fueled systems, critical approach, pulsed neutron source and foil activation measurements were made on three PuO_2-polystyrene-fueled assemblies. (auth)

30103 (BNWL-400) REACTOR PHYSICS DEPARTMENT TECHNICAL ACTIVITIES QUARTERLY REPORT, OCTOBER, NOVEMBER, DECEMBER 1966. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.), Apr. 15, 1967. Contract AT(45-1)-1830. 66p. Dep. CFSTI.

Thermalization theory integral formulation is presented; single particle wave functions describing low energy elastic scattering by target nuclei of He and O are described. A computer program, BROAD, for direct calculation of temperature dependent cross sections has been written. Modifications were made to UNICORN, HRG, TEMPEST, ZODIAC 2, ZODIAC 3, and ZODIAC F. The computer code OP has been written for shielding studies. CONVRT has been written to convert foil activation data from paper tape to digital computer language. Reactivity coefficients are presented for a Pu fueled light water reactor. A description of the gamma facility used to acquire data from irradiated fuel elements is given. The development of the reactivity technique for determination of effective ^{10}B concentration in samples of borated D_2O is described. The results of critical experiments in the PRCF are given. Results from an extensive set of reactivity measurements of neutron absorbing rods in $PuO_2-UO_2-H_2O$ lattices are summarized. The worth of the HTLTR control rods are given. Comparisons of unzoned and two-zoned concepts for the FTR are detailed. Crystalline binding effects on the FTR Doppler coefficient are discussed. FTR fuel heterogeneity effects are examined. Fast reactor investigations for the FTR are presented. Reactivity measurement techniques using pulsed neutrons are discussed.

Analyses of critical experiments in hydrogenous media are given. Crystal spectrometer studies of atomic and molecular motions of water at 268°K and 299°K are presented. Cross section data for slow neutron inelastic scattering in H_2O and D_2O are presented. (M.L.S.)

44830 (GA-7396) HTGR BASE PROGRAM. Quarterly Progress Report for the Period Ending August 31, 1966. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Nov. 15, 1966. Contract AT(04-3)-167. 58p. Dep. CFSTI.

PRESSURE VESSELS—testing of high temperature gas cooled reactor prestressed concrete

REACTORS, GAS-COOLED—pressure vessels for high temperature, development of prestressed concrete; fuel elements for high temperature, design effects on reprocessing of; kinetics computer codes for high temperature, development of space-time; fuel cycle for high temperature, economics of

GAMMA RADIATION—buildup in heterogeneous media of, calculation of; flux calculations using transport theory, effects of quadrature on

IN-PILE LOOPS—fuel element irradiated in GAIL, postirradiated testing of

REACTOR FUELS—radiation effects on coated particle, tabulation of GAIL IV; cycle economics for high temperature gas-cooled; coatings for, development of barrier; diluent particles in, chemistry studies of; irradiation of coated particle, results of

URANIUM ISOTOPES U-238—resonance absorption as a function of lattice parameters, calculation of

FISSION PRODUCTS—deposition on stainless steel specimen from GAIL IV fuel element; deposition in stainless steel pipe downstream from GAIL III-B fuel element; release from beryllia fuel compact of steady-state gaseous

FILTERS—radiochemical analyses of GAIL IV main loop, results of

REACTOR FUEL ELEMENTS—reprocessing of high temperature gas-cooled, effects of fuel element design on; reprocessing of high temperature gas-cooled, economics of

COATINGS—behavior of silicon carbide reactor fuel, effects of strongly corrosive atmosphere on; fission product release from barrier

NEUTRON CROSS SECTIONS—evaluation of uranium 235

PLUTONIUM—diffusion studies on

GRAPHITE—radiation effects at high temperatures on; thermal expansivities of isotropic and anisotropic (M.L.S.)

40674 (GEAP-5294) ANALYSIS OF THE SEFOR MOCKUP CRITICAL EXPERIMENTS IN ZPR-3. Reynolds, A. B.; Stewart, S. L. (General Electric Co., Sunnyvale, Calif. Advanced Products Operation). Mar. 1967. Contract AT(04-3)-540. 54p. Dep. CFSTI.

For Southwest Atomic Energy Associates.

A series of critical experiments were performed on a mockup of SEFOR at ZPR-3. Analyses of these experiments and the application of the results to the SEFOR design are discussed. Critical mass values were determined for 1-, 2-, and 3-segment SEFOR fuel designs to help establish the Pu atom fraction in the SEFOR fuel. Reactivity effects of axial fuel expansion were measured which led to selection of the 2-segment design for SEFOR. Measurements of the reactivity worth of the radial reflector established the adequacy of the SEFOR reflector control system. The Doppler coefficient was measured. The calculated ^{238}U Doppler coefficient was in agreement with the experiment; the measured ^{239}Pu contribution to the SEFOR Doppler coefficient was near zero. It was demonstrated that the SEFOR Doppler is significantly more negative than the conservative value assumed for safeguards analysis. The maximum positive reactivity caused by loss of sodium was measured; the measured reactivity was small (+6¢) and close to the calculated value. The ratio of prompt neutron lifetime to delayed neutron fraction was measured both by the pulsed neutron technique and by noise analysis. The values measured by the two techniques were in agreement. Fission ratios, fission and boron traverses, and Pu worth distributions were measured and are compared with calculations. (auth)

32734 (ANL-7310, pp 137-248) FAST REACTOR PHYSICS. (Argonne National Lab., Ill.).

URANIUM ISOTOPES U-238—neutron cross sections for, calculated effects of reactor environment on absorption; fission densities in, sodium void effects on absolute; neutron capture densities in, sodium void effects on absolute; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3

NEUTRON CROSS SECTIONS—calculation of uranium-238 absorption, effects of reactor environment on

CRITICAL ASSEMBLIES—reactivity effects of uranium-238 on fast, comparison of measured and calculated; reactivity worths and expansion effects in fast ZPR-6, measurement of; neutron flux spectra in ZPR-6 and ZPR-9, comparison of real and adjoint; Doppler effect measurements in ZPR-6 and ZPR-9; reactor dimensions and material compositions of fast ZPR-9 assemblies 12-17; Doppler reactivity measurements in fast ZPR-9 assemblies 13-17; critical mass determination for large uranium carbide fast ZPR-6; fuel loading patterns for uranium carbide fast ZPR-6; core for fast ZPR-6, heterogeneity effects on; fuel loading in uranium carbide fast ZPR-6, absolute fission rates for bunched and unbunched uranium-238 in; core for fast ZPR-6, sodium void effects on absolute fission densities in uranium-238 and uranium-235 and capture density in uranium-238; sodium void coefficient measurements for large uranium carbide fast ZPR-6; sodium void coefficients for ZPR-6, effect of reactor environment and loading pattern on; reactivity measurements in large uranium carbide fast ZPR-6, temperature effects on; cores for ZPR-6 and ZPR-9, measurement of central fission ratio in; core for ZPR-6, measurement of fission neutron density in; design parameters for ZPR-9 assemblies 11 and 12; cores for ZPR-6 and ZPR-9, comparison of calculated real and adjoint fluxed at centerline of; Doppler effect measurements in ZPR-9 assemblies 11 and 12; cores for ZPR-6, neutron spectrum comparisons for zoned and homogeneous fast; sodium void reactivity calculations for ZPR-6, heterogeneity effects on; neutron die-away experiments in ZPR-6, tabulation of results from; reactivity measurements in ZPR-6, pulsed neutron techniques for; neutron decay constant measurements in ZPR-6, use of pulsed neutron technique for prompt; reflectors for fact ZPR-3, savings properties of inconel-sodium; core for ZPR-3, symmetry effects of embedded reflectors in blanket around; reactivity worth of, relation of reciprocal flux to; reactivity worths in ZPR-3, comparison of measured and calculated; plutonium fission rates in, calculation of

REACTIVITY—effects of uranium-238 on fast critical assemblies, comparison of measured and calculated; effects of fissile material expansion in fast reactors, calculational procedure for determining; measurements in fast ZPR-9 assemblies 13-17, effects of temperature on; measurements in ZPR-6, pulsed neutron techniques for; determinations for fast converter reactors, effects of sodium voiding on

URANIUM OXIDES UO_2—fuel pellets of, temperature effects on radial expansion of axial constrained

URANIUM ISOTOPES U-235—expansion effects on fast ZPR-6 reactivity; fission densities in, sodium void effects on absolute; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3

MOLYBDENUM—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3

TUNGSTEN—reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12; Doppler effect measurements in ZPR-9 assemblies 12-17
NICKEL—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3
TANTALUM—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3
IRON—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide ZPR-6; reactivity worth determination in ZPR-3
CARBON—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3
BORON—Doppler effect measurements in ZPR-9 assemblies 12-17
STAINLESS STEEL—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurement in large uranium carbide ZPR-6 on Type-304; reactivity measurements in ZPR-9 assemblies 11 and 12 on Type 304; reactivity worth determination in ZPR-3 of Type 304
ALUMINUM—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3
BERYLLIUM—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12
ETHYLENE POLYMERS—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12
SODIUM—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3
PLUTONIUM—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12
URANIUM—Doppler effect measurements in ZPR-9 assemblies 12-17 on enriched and depleted
BORON ISOTOPES B-10—reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3
VANADIUM—reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12
LITHIUM—reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12
ZIRCONIUM—reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemassemblies 11 and 12
BORON CARBIDES—reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12
NIOBIUM—reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12
CHROMIUM—reactivity measurements in large uranium carbide ZPR-6; reactivity worth determination in ZPR-3
URANIUM ISOTOPES U-233—reactivity measurements in large uranium carbide ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12
IRON OXIDES Fe_2O_3—reactivity measurements in large uranium carbide ZPR-6
RHENIUM—reactivity measurements in large uranium carbide ZPR-6
ZIRCONIUM HYDRIDES—reactivity measurements in large uranium carbide ZPR-6
TITANIUM—reactivity measurements in large uranium carbide ZPR-6
HAFNIUM—reactivity measurements in large uranium carbide ZPR-6
EUROPIUM—reactivity measurements in large uranium carbide ZPR-6
PLUTONIUM ISOTOPES Pu-239—reactivity worth determination in ZPR-3
PLUTONIUM ISOTOPES Pu-240—reactivity worth determination in ZPR-3

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7788 (ANL-7382) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, SEPTEMBER 1967. (Argonne National Lab., Ill.). Oct. 31, 1967. Contract W-31-109-eng-38. 155p. Dep. CFSTI.

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CRITICAL ASSEMBLIES—Doppler effect measurements in plutonium-fueled ZPR-3; reactor fuel elements for ZPR-3, foil-activation measurements of heterogeneity effects in; reactivity measurements in sodium-plutonium core of ZPR-3 perturbed; resonance interaction effects measurements in ZPR-9

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9988 (ANL-7391) REACTOR DEVELOPMENT PROGRAM. Progress Report, October 1967. (Argonne National Lab., Ill.). Nov. 30, 1967. Contract W-31-109-eng-38. 185p. Dep. CFSTI.

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CRITICAL ASSEMBLIES—Doppler coefficient and plutonium systems studies in ZPR-3, summary of; use of ZPR-3 for FFTF mockup studies; reactivity worth measurements in ZPR-3, FTR

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24925 (ANL-7427) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, FEBRUARY 1968. (Argonne National Lab., Ill.). Mar. 26, 1968. Contract W-31-109-eng-38. 151p. Dep. CFSTI.

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CRITICAL ASSEMBLIES—neutron spectrum in ZPR-3 Assembly 51, central, (T); neutron spectrum in ZPR-6 Assembly 6 with and without sodium, central, (E); sodium-void coefficient measurements in ZPR-6 Assembly 6

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URANIUM ISOTOPES U-235—neutron fission rates in EBR-2, (E/T); neutron fission cross section measurements from 30 to 1500 keV

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30118 (ANL-7438) REACTOR DEVELOPMENT PROGRAM. Progress Report, March 1968. (Argonne National Lab., Ill.). Apr. 26, 1968. Contract W-31-109-eng-38. 135p. Dep. CFSTI.

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CRITICAL ASSEMBLIES—reactivity worth measurements in the FTR Phase B mockup in ZPR-3, sodium; reactivity measurements in ZPR-3, effects of fuel compaction on; reactivity worth measurements in 4000-liter oxide core in ZPR-6; Doppler coefficient measurements in uranium oxide zoned core of ZPR-9 fast; development and construction of ZPPR, status as of March 1968 of; reactivity coefficients for, effects of geometry on

TANTALUM—reactivity worth of, measurement in ZPR-3 of,

BORON CARBIDES—reactivity worth of, measurement in ZPR-3 of

URANIUM—reactivity worth of, measurement in ZPR-3 of

REACTIVITY—Doppler coefficient of, measurement in uranium oxide zoned core of ZPR-9

PLUTONIUM ISOTOPES Pu-239—alpha cross section for, effects on ZPR-3 reactivity of

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PLUTONIUM—reactivity worth of, measurement in ZPR-3 of; separation from ruthenium, use of transpiration technique for; removal from plutonium hexafluorides, effects of presence of fluorine gas on

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35422 (ANL-7445) REACTOR DEVELOPMENT PROGRAM. Progress Report, April 1968. (Argonne National Lab., Ill.). June 12, 1968. Contract W-31-109-eng-38. 155p. Dep. CFSTI.

This monthly progress report includes information on EBR-II, ZPR-3, ZPPR, LMFBR, general reactor technology, and reactor safety. For detailed information, see the annual reports from the following ANL Divisions: Chemical Engineering, Metallurgy, Reactor Engineering, and Reactor Physics. 13 references. (J.M.J.)

37401 (ANL-7457) REACTOR DEVELOPMENT PROGRAM. Progress Report, May 1968. (Argonne National Lab., Ill.). July 2, 1968. Contract W-31-109-eng-38. 148p. Dep. CFSTI.

This monthly progress report includes information on EBWR, LMFBR, ZPR-3, ZPR-6, ZPR-9, ZPPR, EBR-II, general reactor technology, and reactor safety including TREAT operations. For detailed information, see the annual reports from the following ANL Divisions: Chemical Engineering, Metallurgy, Reactor Engineering, and Reactor Physics. 25 references. (D.C.C.)

42158 (ANL-7460) REACTOR DEVELOPMENT PROGRAM. Progress Report, June 1968. (Argonne National Lab., Ill.). July 29, 1968. Contract W-31-109-eng-38. 136p. Dep. CFSTI.

Highlights of project activities are summarized for the month of June 1968. For detailed summaries, see the annual reports from the following ANL divisions: Chemical Engineering, Metallurgy, Reactor Engineering, and Reactor Physics. During June the EBR-II was operated for 659 MWdt in a run with a full complement of fueled experimental subassemblies and a reference run with reflector subassemblies of stainless steel rather than depleted uranium. Nearly normal production was resumed in the Fuel Cycle Facility hot line. Experiments continued with the ZPR-3 Assembly 52, the second core of the FTR Phase-B critical program. Installation of the ZPPR reactor assembly and associated equipment was completed, and seal door and containment structure testing is in progress. Nuclear safety studies, including TREAT operations, and chemical separations activities are discussed. (H.D.R.)

50699 (ANL-7487) REACTOR DEVELOPMENT PROGRAM. Progress Report, August 1968. (Argonne National Lab., Ill.). Sept. 26, 1968. Contract W-31 109-eng-38. 112p. Dep. CFSTI.

Highlights of project activities are summarized for the month of August 1968. Fuel development programs for the LMFBR are described. Operations of the ZPR-3 Assembly 51 and 52 for LMFBR physics developments during August 1968 are described. Loading and startup programs of the ZPPR are presented. Operations for the EBR-2 during August 1968 are described. Nuclear safety studies, including TREAT operations, and chemical separation activities are discussed. (D.C.C.)

32960 (ANL-7310, pp 31-135) THERMAL REACTOR PHYSICS. (Argonne National Lab., Ill.).

CRITICAL ASSEMBLIES—cadmium ratio measurements in high conversion uniform lattice, gold and indium

GOLD—cadmium ratios for, measurement in high conversion uniform lattices of; cadmium ratios for, determination of neutron temperature and epithermal index from activation

URANIUM ISOTOPES U-235—neutron cross sections for, effects on reactivity of variation of; cadmium ratios for, determination of neutron temperature and epithermal index from activation

INDIUM—cadmium ratios for, measurement in high conversion uniform lattices of

URANIUM ISOTOPES U-238—neutron cross sections for, effects on reactivity of variation of

REACTIVITY—neutron cross section variation effects on, uranium-235 and uranium-238; temperature effects on AARR excess; insertion accidents in AARR, analysis of; transients in AARR, summary of reactor responses to

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EXPERIMENTAL BOILING WATER REACTOR—poisoning for, reactivity worth of boric acid; control rods for, reactivity worth of; temperature coefficient of reactivity for, measurement of uniform; plutonium recycle experiment in, power operation history of; plutonium fuel loading in, measurement of capture-to-fission ratio of plutonium-239 and plutonium-241 in; cadmium ratio measurements in, test conditions for

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NEUTRON CROSS SECTIONS—variations in uranium-235 and uranium-238, *effects on reactivity of*; determination of AARR internal thermal column effective group

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30155 (NAA-SR-12296) AEC UNCLASSIFIED PROGRAMS. Quarterly Technical Progress Report, October–December 1966. (Atomics International, Canoga Park, Calif.). Oct. 25, 1967. Contract AT(04-3)-701. 151p. Dep. CFSTI.

NEUTRON CROSS SECTIONS—evaluation for fast reactor physics analysis, (E/T)

NEUTRONS—spectra in coolant and structural material of sodium-cooled reactors, (E); spectra distribution in fast reactors, analysis of energy and spatial, (T)

REACTORS, FAST—reactivity variations in, Doppler measurements on

REACTORS, LIQUID METAL COOLED—stability of sodium-cooled, effect of boiling and two-phase flow on, (E/T); cooling system of, behavior of fission products released in sodium, (E); coolant fires in, energy and fission product release in sodium, (E/T)

REACTOR SAFETY—coolant boiling and two-phase flow on sodium-cooled reactors, (E/T); fission product behavior in cooling system of sodium-cooled reactors, (E); coolant fires in sodium-cooled reactors, energy and fission product release in, (E/T)

REACTIVITY—Doppler coefficient of, method for determination of. (E/T)

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27396 (AI-AEC-12638) AEC UNCLASSIFIED PROGRAMS. Quarterly Technical Progress Report, October–December 1967. (Atomics International, Canoga Park, Calif.). Contract AT(04-3)-701. 209p. Dep. CFSTI.

NEUTRON CROSS SECTIONS—SCORE evaluation system for, description of

THORIUM ISOTOPES Th-232—neutron cross section for, use of optical model for evaluation of

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URANIUM ISOTOPES U-238—neutron cross section for, use of optical model for evaluation of; reactivity worth measurements in fast spectrum reactor

REACTORS, FAST—core for, temperature coefficient of thorium metal

THORIUM—reactivity worth measurements in fast spectrum reactor

PLUTONIUM—Doppler effect of, effects of isotopic composition on

REACTIVITY—worth measurements of thorium and uranium-238 in fast spectrum reactor; temperature coefficient of, effects of plutonium isotopic content on; worth measurement of uranium-235, effects of mass on

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32740 (BAW-3647-7) PHYSICS VERIFICATION PROGRAM. PART II. Final Report. Baldwin, M. N.; Clark, R. H.; Rogers, J. E. (Babcock and Wilcox Co., Lynchburg, Va. Research and Development Div.). Apr. 1968. Contract AT(30-1)-3647. 123p. Dep. CFSTI.

CRITICAL ASSEMBLIES—description of B & W Critical Experiment Laboratory uranium oxide (UO_2)-fueled water-moderated multi-rod

CRITICAL STUDIES—reactivity measurements for aluminum-clad uranium oxide (UO_2) multi-rod water-moderated, analysis of

REACTOR FUEL ELEMENTS—assemblies of aluminum-clad uranium oxide (UO_2), criticality of multi-rod circular

RADIATION DETECTORS, PROPORTIONAL—calibration of B & W Critical Experiment Laboratory multi-rod critical assembly, description of

RADIATION DETECTORS, FISSION CHAMBER, RADIATION DETECTORS, IONIZATION CHAMBER—calibration of B & W Critical Experiment Laboratory multi-rod critical assembly, description of

NEUTRONS—radiation measurement of, effects of concrete shielding thickness on B & W Critical Experiment Laboratory multi-rod critical assembly

REACTIVITY—measurement of aluminum-clad uranium oxide (UO_2) multi-rod water-moderated critical assembly, analysis of

RADIATION DETECTORS, ACTIVATION—measurement of neutron flux distribution in aluminum-clad uranium oxide (UO_2) multi-rod water-moderated critical assembly, analysis of; design of rhodium wire, description of; performance of rhodium wire, description of; calibration of aluminum–dysprosium wire, description of; performance of aluminum–dysprosium wire, description of

RHODIUM—use for neutron flux measurement, description of

ALUMINUM ALLOYS AND SYSTEMS—Al–Dy, use for neutron flux measurement, description of; Al–Cd–In, control rod reactivity worth measurements for, description of

REACTOR CONTROL ELEMENTS—reactivity worth of aluminum–cadmium–indium, measurement of (D.C.C.)

6090 (EURFNR-395) COMPARISON OF MEASUREMENTS IN SNEAK-1 AND ZPR III-41. Barleon, L.; Boehme, R.; Boehnel, K.; (and others) (Kernforschungszentrum, Karlsruhe (West Germany)). Oct. 1967. 34p. (EUR-3670e; KFK 626; CONF-671043-2). Dep. CFSTI.

Work performed under United States-Euratom Fast Reactor Exchange Program.

From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany.

Data for critical mass measurements, prompt neutron decay constants, flux profiles, and fission ratios are presented and compared for the SNEAK-1 and the ZPR III-41 reactors. (D.C.C.)

12085 (BNWL-472, pp 3.1-33) THERMAL REACTORS. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

Measurements to obtain adjoint weighted excess neutron production cross sections were completed for four lattices of PuO_2-UO_2 fuel in graphite moderator. The measurements were made in the Physical Constants Test Reactor (PCTR), and a value of the infinite medium neutron multiplication factor (k_∞) was obtained for one lattice using the traditional PCTR null reactivity technique. This value of k_∞ was compared with the corresponding value obtained from measurements for the same lattice. Analysis of the CAF-Phoenix water reflected core with 17 energy groups yielded a calculated reactivity within 1% of experiment using diffusion theory and 3% using transport theory. Pre-experiment calculations on the PRCF-Phoenix core revealed that k dropped as more core-reflector interface detail was added. Final critical loading estimates were less than 9% low. Placement of flux suppressors for the MTR-Phoenix core was investigated to avoid having the far-out flux increases outweigh the close-in flux decreases. A full-core banked-shim critical loading was achieved in the PRCF-Phoenix fuel experiment, with the shims 65% withdrawn. Power distribution measurements in the shimmed core have shown the expected power peaking in the fuel followers just below the bottom of the core. Compilations have been issued of the burnup data obtained by destructive analysis of Al-1.8 wt % Pu and Al-2.6 wt % Pu fuels which were irradiated in PRTR. Analysis of the burnup data from the Aluminum-Plutonium fuels irradiated in PRTR was completed. The isotopic concentration data were processed using multivariable regression analysis to obtain a unique set of cross section ratios. An empirical formula was derived for use in fitting critical mass data from cylindrical arrays of rods, moderated with H_2O. The formula provides a method of accurate interpolation to obtain the fuel-to-moderator ratio for minimum critical mass. Relative rod power measurements were made in PRTR to determine the power match between fuel rods in elements in the Batch Core and fuel rods in several possible test configurations in the Fuel Element Rupture Test Facility (FERTF). The relative rod powers were determined by gamma scanning. The reactivity changes associated with loss of coolant from the PRCF were also measured for several fuel compositions and test configurations. A theory-experiment correlation study of ratios of effective cross sections was performed. A result of the study is that the calculated ratios agree best with the experimental ratios using the Leonard normalization for the ^{239}Pu and ^{241}Pu thermal cross sections. Calculations of the photoneutron production in D_2O moderated reactor systems fueled with 19-rod clusters of Al-Pu, UO_2-PuO_2, and UC were made. A calculational study was made comparing values of the Dancoff correction obtained via various methods. The corresponding effect of resonance integrals and reactivity were also determined. Calculations were made of power peaking factors in H_2O moderated loadings of UO_2-2 wt % PuO_2 fuel. The calculated values agree with measured values to within ±5%. Calculations were performed to help predict critical masses and various lattice parameters for H_2O moderated lattices containing UO_2-4 wt % PuO_2 fuel rods. The study was made assuming two types of fuel differing in ^{240}Pu content. (auth)

12086 (BNWL-472, pp 4.1-43) FAST REACTORS. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

A survey was performed to investigate the effect of BeO on FTR neutronics. Items considered included the sodium coefficient, Doppler coefficient, critical mass, and flux spectrum dependence on BeO volume fractions. A detailed analysis of FTR control rod configurations was carried out. A variety of representative rod patterns were examined and their reactivity worth determined. The effects of fuel failure and fuel relocation (slumping) in the FTR are being re-examined. A series of kinetics calculations were performed for a range of reactivity ramps and shutdown coefficients. The possible utilization of fuel-rod plugs to minimize the effects of fuel slumping was also considered. To check calculational techniques used in FTR design, analysis of the sodium void experiments performed on the SEFOR mockup in ZRP-III was undertaken. A fairly detailed analysis of the Doppler coefficient measurements in the same assembly is also being carried out. Incorporation of crystalline binding effects in the analysis caused the Doppler coefficient to diverge from the 1/T dependence in qualitative agreement with the experimental results. The two-dimensional perturbations code, 2-D PERT, is not operational. This code will be used for driver fuel and test management studies. The Phase-A critical experiments in ZPR-III were completed, and analysis of the experiments was started. The attenuation characteristics of a number of alternate shield arrangements were compared analytically to provide a basis for the conceptual design of the FTR. Estimates of gamma intensity at selected locations within two different closed loop cell concepts were also made. (auth)

14251 (BNWL-531) REACTOR PHYSICS DEPARTMENT TECHNICAL ACTIVITIES. Quarterly Report, April–June 1967. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Jan. 1968. Contract AT(45-1)-1830. 84p. Dep. CFSTI.

NEUTRON CROSS SECTIONS—analysis of, use of optical model for theoretical
COMPUTER PROGRAMS—description of fast reactor FCC-IV
REACTOR FUEL ELEMENTS—reactivity worth of PRTR, measurement of; power distributions in Phoenix, measurement in PRCF of; neutron multiplication factors for Phoenix, comparison of experimental and calculated values for; Doppler coefficients for FTR, effects on fuel temperature of variations of
PLUTONIUM RECYCLE TEST REACTOR—neutron fluxes in, ratios of axial peak-to-average; fuel element recycle test facility for, measurement of fuel element reactivity worths in
BORON—reactivity worth of, PRTR batch core measurement of
REACTORS—lattice parameters for uranium dioxide 19-cluster, analysis using RBV Monte Carlo code for; fuel lattices for, temperature coefficient calculations for uranium-233 oxide (UO_2)-thorium-232 oxide
PLUTONIUM RECYCLE CRITICAL FACILITY—reactivity worth measurements of Phoenix core in
HIGH-TEMPERATURE LATTICE TEST REACTOR—Doppler coefficient measurements for, procedures for
FAST TEST REACTOR—physics characteristics of, analysis of split-cone; control rods for, reactivity worth of safety; accident conditions in, fuel pin center-line temperatures under; control rods for, calculations of re-activity worths of; components for, heat generation rates and neutron damage rates in
REACTOR SAFETY—FTR fuel pin centerline temperature under accident conditions
CRITICAL ASSEMBLIES—Doppler coefficients for ZPR-3, comparison of measured and calculated values for; neutron fission rates in ZPR-3, uranium-238 and plutonium-239; neutron spectral measurements in ZPR-3 and ZPR-6,
URANIUM ISOTOPES U-238—neutron fission rates in ZPR-3
PLUTONIUM ISOTOPES Pu-239—neutron fission rates in ZPR-3
REACTOR CONTROL ELEMENTS—reactivity worth calculations for FTR
URANYL NITRATE—critical uranium-235 enrichment for solutions of, determination of uranium
REACTIVITY—worth of uranyl nitrate as function of Uranium-235 enrichment determination of maximum excess
PHYSICAL CONSTANTS TEST REACTOR—reactivity worth measurements in, determination of critical uranium-235 enrichment for uranyl nitrate, using
NEUTRONS—monochromatization of, use of multiple Bragg reflection for high-resolution
CHARGED PARTICLES—relativistic motion of, description of computer program for calculating and plotting eight-dimensional phase-space electromagnetic plane wave-driven (M.L.S.)

20779 (BNWL-624, pp 3.1-159) REACTOR PHYSICS. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

Experiments were conducted in the Critical Approach Facility and the Plutonium Recycle Critical Facility to investigate the detailed physics properties of plutonium-fueled reactor systems. Measurements have been made using UO_2-2 wt % PuO_2 rods in H_2O moderator. Experiments were conducted at lattice spacings covering the range of moderator to PuO_2-UO_2 volume ratios of 0.61 to 9.7 and the plutonium contained either 8, 16 or 24% ^{240}Pu. The results included critical masses, bucklings, flux and power distributions, reactivity coefficients, kinetic parameters, and control rod worths. The PCTR is being used for evaluating the effect on reactivity of finite PuO_2 particles in UO_2 and for obtaining additional data on water lattices not obtainable from other facilities which are required for further evaluation of analysis methods. The initial results of the PuO_2 particle size experiments indicate that the reactivity decreases as the size is increased from 100 μ to 350 μ for the fuel composition used (2.0 wt % PuO_2, 8.05 wt % ^{240}Pu). The design of the first experiment in the water tank is complete and will contain PuO_2-UO_2 fuel on a 1.0 in. lattice pitch. A large scale burnup experiment has been initiated in the D_2O-moderated Plutonium Recycle Test Reactor using 19-rod clusters of UO_2-2 wt % PuO_2 rods. The first phase of the experiment has been completed and consisted of an extensive set of tests at zero power during the initial loading of the core. A series of power tests designed primarily to verify predicted operational characteristics of the reactor were conducted when the loading was completed. An irradiation experiment was conducted in the Experimental Boiling Water Reactor at Argonne National Laboratory. This experiment is part of the joint ANL-PNL program to demonstrate the utilization of plutonium in a boiling H_2O power reactor and to obtain useful physics information on the behavior of a plutonium fuel in such a reactor system. At three stages of burnup a series of rods were removed from the plutonium zone for nondestructive and destructive analysis. The series of rods contained Al-Pu, natural UO_2, and UO_2-PuO_2 fuels. In addition, the fuels which contained plutonium differed in their ^{240}Pu concentration. Some of the rods are selected from positions in the core so that the spatial distribution of burnup can be obtained from the nondestructive and destructive analysis. Fuel rods removed from selected locations in the EBWR and from past and present experiments in the PRTR are both nondestructively and destructively analyzed to obtain fission and fuel concentration and isotopic composition. From these data effective cross section ratios are derived for use in evaluating burnup analysis methods. New techniques for this purpose using multi-parameter, non-linear regression methods have been developed and applied to data from four sets of irradiated Al-Pu samples. The PNL Gamma Scanner has been improved and continues to be used for the nondestructive analysis. Information on the physics characteristics of pressurized water power reactor systems which were obtained during the Saxton plutonium program is compared to results obtained with zero, one, and two-dimensional diffusion theory methods. The conclusion reached is that a one-dimensional cylindrical model of the reactor seems adequate from the standpoint of reactivity calculations for single zone lattices. Analytical survey studies have been performed to provide information on the physics characteristics for various reactor systems of interest. It is shown that the UO_2-4 wt % PuO_2-H_2O cores which were studied would be undermoderated and that the characteristic parameters such as η, f, and p play an important role in the physics behavior of these cores. A survey study was conducted to determine the approximate magnitude of various reactivity coefficients for UO_2-PuO_2 fueled light water reactors. Calculations were also performed to determine the reactor characteristics of thorium loadings in D_2O moderator. The feasibility of the thorium loading in the PRTR was determined either as a batch core or as a driver region for a UO_2-PuO_2 core. The physics characteristics of metallic fuel and of ceramic fuels were investigated along with the variation in rod size in thorium enrichment. Slow neutron inelastic scattering cross sections have been reported for H_2O and D_2O. The double differential cross section and corresponding Egelstaff scattering law have been obtained from measurements for room temperature H_2O and D_2O and for H_2O of five degrees below its freezing point using the Battelle Rotating Crystal time-of-flight Spectrometer. In addition, results of measurements for H_2O at 95°C using the Battelle Triple-Axis Spectrometer have been reported. An extensive evaluation of the RBU Monte Carlo Code has been completed. Results of this evaluation indicate that the RBU Monte Carlo code is free of gross program errors and can be used reliably for reactor physics calculations. Knowledge of the Legendre moments of moderator scattering cross sections is of particular importance in the prediction of the thermal neutron spectrum of plutonium fueled reactors. Based on the Egelstaff-Schofield formalism, two methods for calculating scattering moments for moderators have been developed and programmed on the computer. A model for water is being evaluated using available experimental data. Several computer codes have been adapted, modified, or improved for use in the physics programs. The RBU Monte Carlo code has been thoroughly tested and appears to be working satisfactorily. The ZODIAC-2 burnup code was modified and enlarged to increase its burnup capabilities. The HRG (Hanford Revised GAM) code spectrum model used in resonance integral calculations was improved. Several special purpose codes have been developed. Assistance was provided to Brookhaven National Laboratory in preparing nuclear data for the ENDF/B. Data were furnished for the ten isotopes which Battelle accepted responsibility to evaluate as a member of the Cross Section Evaluation Working Group. Adaption of the ENDF/B system to the local UNIVAC 1108 is in progress. Theoretical studies have been directed toward developing a mathematical physics model which accurately predicts the observed physics behavior of power reactor systems. A measure of the validity of the mathematical model is obtained by comparing measured and calculated integral parameters such as reactor multiplication and effective cross sections. Comparisons have been performed for numerous plutonium and/or uranium fueled H_2O lattices. Comparisons of flux and power densities, reactivity coefficients, kinetic parameters, and effective cross section ratios, have also been made. The results of the comparisons show that in many cases the calculational techniques need to be refined. (auth)

32743 (BNWL-739) PLUTONIUM UTILIZATION PROGRAM. **Technical Activities Quarterly Report, December 1967–February** 1968. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Apr. 1968. Contract AT(45-1)-1830. 84p. Dep. CFSTI.

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PLUTONIUM OXIDES PuO_2 — $PuO_2 - UO_2$, radiation effects on, analysis of neutron; $PuO_2 - UO_2$, fuel loading for PRCF reactor of, physics measurements for critical, (E/T)

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CRITICALITY STUDIES — measurement of plutonium oxide (PuO_2) - uranium oxide (UO_2), effects of plutonium enrichment on, (E/T)

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REACTIVITY — measurement of PRCF uranium oxide (UO_2)-fueled core excess, analysis of; measurement of effects of boron concentration in water on TTR, analysis of, (E/T)
PLUTONIUM RECYCLE CRITICAL FACILITY — core loading for, reactivity measurements for uranium oxide (UO_2) and plutonium oxide (PuO_2) - uranium oxide (UO_2),
URANIUM OXIDES UO_2 — fuel loading for PRCF reactor of, physics measurements for critical, (E/T)

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42164 (BNWL-775) **REACTOR PHYSICS DEPARTMENT TECHNICAL ACTIVITIES QUARTERLY REPORT, JANUARY-MARCH 1968. (Battelle-Northwest, Richland, Wash. Pacific** Northwest Lab.). July 1968. Contract AT(45-1)-1830. 104p. Dep. CFSTI.

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CRITICAL ASSEMBLIES — kinetics for plutonium oxide (PuO_2) - uranium oxide (UO_2)-fueled NPTF, comparison with FTR of

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PLUTONIUM NITRATES $Pu(NO_3)_4$ — criticality of slab-shaped, analysis of
CRITICALITY STUDIES — measurement of slab-shaped plutonium nitrate ($Pu(NO_3)_4$), description of; measurement of rectangular-shaped polystyrene - plutonium oxide (PuO_2) lattices, description of
PLUTONIUM OXIDES PuO_2 — polystyrene - PuO_2, criticality of rectangular-shaped lattices of, analysis of (D.C.C.)

20916 (ANL-Trans-556) EXPERIMENTAL STUDY OF NEUTRON CHARACTERISTICS IN FAST MULTIPLYING MEDIA, USING THE THERMAL-FAST CRITICAL ASSEMBLY "ERMINE." Bouchard, J.; Mougniot, J. C.; Vidal, R. Translated for Argonne National Lab., Ill., from Paper Presented at Conference on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany, Oct. 30–Nov. 3, 1967. 22p. (CONF-671043-15). Dep. CFSTI. JCL $2.60 fs, $1.10 mf.

The fast-thermal ERMINE assembly in the MINERVE reactor is described. Reactivity and flux measurements and measurement techniques are described. Doppler effect and neutron importance measurements are discussed. Spectral index values are tabulated; results of reactivity measurements are listed. Data are shown graphically. (M.L.S.)

49817 (EURAEC-2036) USE OF PLUTONIUM AS NUCLEAR REACTOR FUEL. Quarterly Report No. 17, January 1-March 31, 1968. (Centre d'Etude de l'Energie Nucleaire, Brussels (Belgium)). Apr. 1968. Contract 001-64-1-TRUB. 70p. (EUR-3986; BN-6804-01). Dep. CFSTI.

Work Performed under United States-Euratom Joint Research and Development Program.

Fabrication of the 260 homogeneous fuel rods containing 2% ^{235}U and 2.7% ^{239}Pu is completed. Fabrication has started on the making of 250 heterogeneous type rods containing 2% ^{235}U and 2.7% ^{239}Pu. A part of these rods will be fabricated in the self-contained unit on the pneumatic vibrator. Checking of the chlorine content of the plutoniferous rods has continued satisfactorily. Two checking procedures of the density measurements with mercury and the determination of the humidity content of solid samples have been established. A new series of eight trip burnouts has been carried out in the burnout testing. The results on fuel assessment of the dimensional measurements and the tapping tests are given for the six rods irradiated in hydraulic conveyor (HR-1 runs 5 and 6). The swelling of the rods reached several per cent and there is a difference between the homogeneous and heterogeneous rods in composition and volume of gas collected. Two irradiation tests of three Zircaloy 2 clad fuel rods have been effected in hydraulic conveyor at a power rate in the region of 770 W/cm (HR-1 runs 9 and 10). Following the 9 and 10 HR-1 runs, three sets of three stainless steel clad fuel rods were irradiated up to a power of 1250 W/cm. The dimensional measurements effected before and after irradiation showed that the homogeneous rods had swollen by about 1 to 2%, while the heterogeneous rods had retained their initial dimensions. The maximum power attained in CEB-4 is 800 W/cm. The burn-up ratios are 22500 MWd/t in the homogeneous rod and 18500 MWd/t in the heterogeneous rod. The PANTHER code has been improved by proceeding with the treatment of the heterogeneity in the fast field, the broadening of the resonance of ^{240}Pu at 1.05 eV with the temperature, and modifications in the fast library of some isotopes. The rectangular configuration with a 1.303 cm square pitch has been studied with 4/0 fuel. The critical mass has been determined, together with the axial and radial bucklings. The reactive effects and the power distributions have been measured in the presence of a number of perturbations (water films, aluminum plate, absorbent rods). Analysis of the data from the sub-critical VENUS-Vulcain tests has continued. A supplementary program of sub-critical tests on 4/0 fuel with an $N\sqrt{2}$ pitch has been outlined. The study of the power distribution during the 5th cycle has been undertaken in the case where the 7th core layer contains a UO_2-PuO_2 area with 3.6 wt % of PuO_2. (J.C.W.)

39875 (BNL-50082, pp 1-18) REACTOR PHYSICS DIVISION. Chernick, J.; Kouts, H. (Brookhaven National Lab., Upton, N. Y.).

Developments are reported for studies on: reactor safety; Na temperature coefficient distribution in plenum reactors; fast reactor criticals; fast neutron transport; reactor dynamics; neutron thermalization; iteration methods; reactor kinetics; fluid dynamics; reactor physics HAMMER CODE; analysis of BNL ^{235}U-graphite critical experiments; analysis of Brookhaven $^{233}U-ThO_2$ lattices; spectrum-shift lattice experiments; small reflected assemblies; uranium-graphite critical assemblies; californium experiment; $^{233}UO_2-ThO_2-D_2O$ exponential experiments; anisotropy of neutron migration; physics of fast Na-cooled reactors; and support work for experiments. (D.C.C.)

50847 FISICA DEL REATTORE. CONVEGNO INDETTO DALLA, SOCIETA LOMBARDA DI FISICA, MILANO-PAVIA, DICEMBRE 1963. (Reactor Physics. Meeting of the Lombarda Physics Society, Milan-Pavia, Italy, December 1963). Congressi, Convegni, e Simposi Scientifici. 9. Rome, Consiglio Nazionale delle Ricerche, 1966. 857p. (In English and Italian). (CONF-469).

Included are 66 papers on theoretical and experimental methods for the analysis of nuclear reactor physics. Individual abstracts were prepared for 61 papers. Abstracts of the remaining 5 papers have appeared in previous issues of NSA. The abstract number of one paper is NSA 19: 12796 and the other 4 papers may be located under CONF-469 in the report number indexes. (H.D.R.)

For abstracts of individual papers see: 48885, 49074, 49101, 49102, 49776, 50235, 50239, 50510-50516, 50604-50606, 50752-50754, 50760-50765, 50813-50819, 50822, 50825, 50826, 50828-50833, 50846, 50848-50854, 50861-50866, and 50868-50871.