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

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

1. CRITICAL EXPERIMENTS
Reasonably Homogeneous

1967


The fast reactor FRO is a split-table machine with vertical fuel elements. A quantity of 120 kg ²³⁵U 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 VAPIT (29% Zr) 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, worth 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 fission activation have been initiated. The latter have been carried out at 500°C, and preliminary results for ²³⁵U and ²³⁸U 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)


The various phases of fast reactor physics considered in the conference were: differential and group cross-section data, tests of cross-section sets, 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, 19416, 56613; one paper is abstracted under the report number TID-Report-13641. (M. L. S.)

For abstracts of individual papers see: 35414, 35581 — 35584, 35547, 35549 — 35563, 35545, and 35568 — 35570.


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)
The relative fission rates of SW, 2SU and 235U were measured in 1967.

Doerr, R. C.; Knapp, W. G.; Almenas, K. K.; Karam, R. A. The study of this assembly, No. 48, in the ZPR-3 series, is still in progress, and 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)

Critical Experiments: Reasonably Homogeneous


The critical masses of oil rejected, enriched uranium spherical assemblies of inside uranium radius, 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, 31.6, 52.1, and 81.3 kg. Critical masses were determined by reciprocal multiplication measurements on fully reflected assemblies and are compared with calculations. (auth)


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)


The results of measurements made on four fast critical experiments performed in ZPR-3 summarized in support of the rocket-design effort. The fuel is highly enriched 235U, the major fissioning isotope is 235U, and the reflector material is Al. Data on cross section s for W, Re, and 18 are included. (J.C.W.)


The materials buckling of four BeO moderated 239Pu - Al alloy fuelled systems having 252-239Pu atomic ratios of 1707, 2499, 3749 and 4999 have been measured by the exponential method. Relative fission rates of 235U, 239Pu and 239Pu have been measured by the exponential method. Measurements were also made in the equilibrium spectrum region of the same assemblies. Because of the heterogeneous nature of the assemblies, finite structure corrections were applied. Some calculations using the CRAM diffusion code and the GYMEYA code are included. (auth)
Critical Experiments:
Reasonably Homogeneous


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)


1967

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 2DP-III Assembly 17 (the SFOC 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 previously 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)
Critical Experiments: Reasonably Homogeneous


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 moderated volume Pu and U assemblies analyzed, the data predict k within about 1.5%. (auth).


35549 (ANL-7320, pp 270-5) USE OF INTEGRAL MEASUREMENTS AS SUPPLEMENTARY DATA IN NEUTRON CROSS-SECTION EVALUATION. Pazy, A.; Rakavy, G.; Reza, Y.; Gelv, 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. Persiapi, F. 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 Assemblies 48, 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)
Critical Experiments: Reasonably Homogeneous


The test program and procedures for an accurate mock-up of a gaseous UF₄ experiment in thin cavity reactor using sheet fuel followed by an actual UF₄ 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 eff, 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 UF₄.


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

The fast core of the Mixed Spectrum Critical Assembly (MSCA) or the Vallenite Atomic Laboratory, contains a loading of enriched U₃O₈, and Inconel. The neutron spectrum in this assembly is spatially asymptotic and representative of a dilute fast ceramic reactor. Measurements reported include fisalion ratios of Th₂O, Th₃O, Pr, and Th³Np. Neutron lifetime determined by pulsed neutron and 1/2 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, high spectrum, refractory metal critical experiments related to space power reactor applications. Be reflected cores in the 20 to 30 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.


The fast core of the Mixed Spectrum Critical Assembly contains a loading of 405 kg Th₂O, 1540 kg Th₂O, and 1013 kg of Inconel. Work in the 710-CE concentrated on small, high spectrum, refractory metal critical experiments related to space power reactor applications. Be reflected cores in the 20 to 30 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.


Descriptions of the cores, critical masses, sodium-coefficient studies, and radial reactivity worths of core materials of large dilute uranium carbide Assemblies 42 and 5 in ZPR-9 and Assembly 11 in ZPR-9 are presented. Assemblies 42 and 11 are zoned core systems. Assembly 5 is a 2600 liter carbide reactor. The cores of all three Assemblies are similar.
Critical Experiments; Reasonably Homogeneous


The critical parameters of single homogenous 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 calculation scheme, the various effects of diluents (including moderators), reflectors, density changes, 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)


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)


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


Criticality data from the literature are compiled into tabulated form. These tables are broken down into: single unmoderated UO2 cores; single unmoderated plutonium cores; single 235U 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)
Critical Experiments: Reasonably Homogeneous

12087 (BNWL-472, pp 5.1-16) CRITICAL MASS PHYSICS, Battelle-Northwest, Richland, Wash., Pacific Northwest Lab.), Criticality experiments, bare and water reflected, were conducted with the 42 x 42 in. slab assembly having adjustable thickness. The plutonium concentration was 54 g/liter and axial modalities were 2.3 and 5. The plutonium contained 4.6 wt % U. Evaluation of the effect of vessel walls, lattice reenforcement, and bed depth was made. Correlation of slab experimental data and initial calculation theory was in poor agreement. The close critical data and water reflected infinite slab was experimentally estimated to be 15.7 and 10.1 cm respectively for 55 g Pu/liter at an acid modality of 2.5. Criticality experiments were performed to provide data for nuclear safety guidance on handling, storing, and shipping of United Kingdom hot type slabs containing 6.6 kg plutonium metal each. Experiments were carried out with the nine slabs bare and reflected with Lucite. Effect of Lucite moderator and cadmium plating was studied qualitatively. The bare array included criticality at about 19 slabs; the reflected array gave 1/2 slabs for criticality. Critical bucklings and masses were measured for a range of lattice spacings of 2.1 wt % enriched U fuel taken in light water. Criticality experiments were performed in support of the Gas-Cooled Fast Breeder Reactor (GCFBR) program. The experiments were designed to stimulate water entry into the GCFBR core and to check basic neutron data and computational techniques. An experimental program to provide data for determining the minimum critical 235U 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 dual time problem connected with fast-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 such a way that it is useful for epithermal and fast neutron systems. A system for experimentally measuring the P2 probability of real 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 PuO2 polyethylene composites 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 239Pu on intermediate neutron spectra plutonium systems. The current series of experiments are concerned with fuel having an atomic Pu content of 11.5 wt %.

1968


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 3, analysis of one-dimensional 24-group spherical-geometry.

CRITICALITY STUDIES - critical measurements using plutonium nitrate in slab geometry.


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.
Critical Experiments: Reasonably Homogeneous


The experimental program of the Karlsruhe Test ZrO Powder Reactor SNEAK started in the autumn, 1966 with measurements on a 440-liter uranium assembly, a mock-up of ZP 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 full activation, and several techniques for determining reactor power and f/1, e.g., Rossi- and pulsed neutron source measurements. In the experimental program quantities such as critical mass, reactivity rate ratios, neutron spectrum, material weights in the center, radial and axial traverses, f/1, and reactor power were determined. The results were generally in good agreement with those of the ZP III experiments and the remaining discrepancies are discussed. There 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 ASM cores. The agreement with the experimental spectrum is too hard, and both KFK and ASM give better agreement with the experimental spectrum. (auth)


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

Critical parameters for mixtures of fuel having an 11.5 wt % PuO₂ isotope concentration and an atomic H/PU ratio of 5 are presented. Experimental data are obtained from both bare and reflected rectangular parallelepipeds of PuO₂-polystyrene fuel. (D.C.C.)


A description of an Al-235-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)


- 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 400-2590 L. Sensitivity of reactivity to changes in cross section is evaluated; cross sections are tabulated for each core volume. Effects of cross section variation on initial conversion factors is discussed. (M.L.S.)


An attempt to achieve a near-critical assembly with a minimum average 235U enrichment of an unreflected, uranium metal, 21-inch-diameter 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-inch-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 mass as a function of reflector steel thickness. (auth)

Critical Experiments: Reasonably Homogeneous


Presented in 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. (ATU)


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


Critical masses were experimentally determined for steel-moderated, oil-reflecting, spherical and hemispherical enriched uranium assemblies having inside radii from 0.0 to 12.6 cm. (auth)

18573 EXPERIMENTS ON THE ZR-3 CRITICAL ASSEMBLY IN CONNECTION WITH THE DEVELOPMENT OF THE WWR-SM CORE. Franki, Laszlo; Gacsi, Laszlo; Szabo, Ferenc; Szakalajda, Laszlo; Varkonyi, Laszlo; KFKI (Korp, Fiz, Kut, Intez.) Koztem, 161 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)


From, 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.
Critical Experiments: Reasonably Homogeneous


A cubical core of enriched (93.15%) UO_2 if foil in a cubical reflector was used to establish minimum critical mass measurements. Three core sizes, of bases approximately 6 in., 8 in., and 10 in. square, were made critical by adjusting core height. Results of the experimental study are presented. (D.C.C.)


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

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.). [Ind], 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.)
1968


A reactor of the BM-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 Pu, 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 Bi proportional counters. Neutron propagation in UC3 and the Doppler effect in UC2 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. (H.D.R.)

Critical Experiments: Reasonably Homogeneous


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


Reflected and unreflected critical experiments were performed at II 235U atomic ratios from 0 to 3 in rectangular geometry with layers of enriched uranium and polyethylene. Base dimensions of the assemblies were 3 x 10 and 10 x 10 in. In some unreflected experiments, the metal was interleaved with Plexiglas and with Teflon at the outer boundary. 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 kinf calculated with the KENO Monte Carlo code agree with the experimental values. By equating geometrical buckling 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)

14267 EXPERIMENTAL REACTOR WITH GASEOUS UF6. No. 1. THE CALCULATION OF THE CRITICAL MASS. Ogushi, Terumune (Chuo Univ., Japan); Yumoto, Ryozo, ChuO Digi (NSA of Japan).

The critical mass of experimental reactor with gaseous UF6 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 104 cm height. The Be moderator was in the form of tubes of 4.0 cm square in cross-section. Gaseous UF6 was filled in 14 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 these 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 UF6 at 0.9 atm. and 80 °C. (auth) (NSA of Japan)
Critical Experiments: Reasonably Homogeneous

10069 THEORETICAL ANALYSES OF HOMOGENEOUS PLUTONIUM CRITICAL EXPERIMENTS. Hickey, C. R. ( Battelle Memorial Inst., Richland, Wash.). Nucl. Sci. Eng., 31: 32-9(1968). A computational analysis was made for the large number of advantageous critical experiments with homogeneous mixtures. The calculations were made using both the INC and diffusion theory with 18 energy groups obtained with the GANTEC-II code. Reactions are captured by the isotope superscripts of Pu. The calculations were treated in the NR and NIA approximations. The results are given as a parametric survey for Pu densities ranging from 0.015 to 1.0 g/cm³. The calculated minimum critical mass of Pu is 547 g for water-reflected aqueous PuNO₃ solutions and 521 g for similar mixtures of Pu and water. 14 references. (auth)


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 H₂O, D₂O, and graphite are included. However, discussions are centered around the use of plutonium in H₂O reactors because these reactor types are of most interest in the United States. Problems connected with calculating H₂O-plutonium systems are illustrated using the data, and areas mentioned in which needs for additional data still exist. Cross sections, 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 H₂O moderated reactors. 391 references. (auth)


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

1968


A series of experiments were conducted at the Idaho Test Station on a large "cavity" reactor consisting of a cavity 185 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 buffets in the reflector and insertion of various structural and operating materials characteristic of a nuclear rocket reactor. 11 references. (auth)


A neutron cross section evaluation and critical experiment analysis conducted in support of PuO₂ 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) MC² 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 18. A comparison of calculated with experiment is presented for these critical experiments. (auth)


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 ²³⁵U, ²³⁶U, and ²³⁹Pu, 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 ²³⁹Pu 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)

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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 how 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)

Critical Experiments: Reasonably Homogeneous


Computations of the effective neutron multiplication factor of single units of aqueous solutions of 238UO2F2 and 239PuO2F2 were 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 (DTT) codes in the S_4 approximation with Hansen-Roach cross sections were utilized and the results agree with relevant experiments to within 0.01 in k_eff. (auth)
II. CRITICAL EXPERIMENTS

Lattices

1967


The results of critical experiments on cores IV, V, and VI are reported. In all assemblies the basic lattice consisted of 0.475-inch-ID rods of 2.45%-enriched UO₂ arrayed on a square pitch of 0.81 inch and moderated by H₂O. The water was poisoned with H₂113 to obtain critical assemblies approximately 5 feet in diameter. The reactivity worth of 4 control pins arranged in configurations typical of power reactors was measured in these cores. The results of measurements of ρₚ, the modified conversion ratio, and the epithermal neutron spectrum are also reported. (auth)


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 Th₂O, Th₂O₃, and lumped fission products. Evaluation work was done on various reactor concepts being considered. (N.C.G.)


The performance of the lattice code WIMS was studied by the analysis of graphite moderated exponential experiments fuelled with clusters of UO₂ or Pu/UO₂ rods at temperatures up to 300°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 ²³⁵Pu η value of 2.008 (at 2200 m/s). The use of the IAEA recommended ²³⁵Pu η value of 2.114 in the work showed a significant dependence of reactivity on ²³⁵Pu enrichment, which was largely removed by using an η value of 2.008. (UK)

15764 (CONF-660221-, pp 287-90) ANALYSIS OF UNIFORM LATTICE EXPERIMENTS WITH THORIA-URANIA FUEL IN HEAVY WATER AND LIGHT WATER AS MODERATORS. Bhalla, H. K. (Atomic Energy Establishment, Trombay (India)); The MENTUS/FLASH and CAROL codes for ThO₂-²³⁵U₉ or ThO₂-²³³U₂ lattices with heavy or light water as moderator were assessed. Experimental bucklings were used to calculate the k₊ for uniform critical lattice experiments performed at Argonne National Laboratory, Brookhaven National Laboratory, and the Babcock and Wilcox Company. (H.D.R.)


The results of critical experiments, performed with organic moderated plate-type assemblies containing U enriched to 90% in ²³⁵U, 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₊ for large-size cores. 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)
A program of critical experiments has extended the range of measured nuclear parameters of light-water-moderated, slightly enriched, UO₂ lattices further into the under-moderated regime. The II-to-²³⁵U atom ratios in the cores assembled ranged from about 5 to 0.5. The initial conversion ratio, II-to-²³⁵U fissile ratio, II enriched 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 data 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 II-to-²³⁵U ratios. (auth)


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 % ²³⁵U enriched U (2.34-in. OD; 1.79-in. ID); 1.25 wt % ²³⁵U enriched U (2.37-in. OD; 1.80-in. ID); and 1.95 wt % ²³⁵U enriched U (2.28-in. OD; 1.41-in. ID). The tube-in-tube assemblies measured were: 0.02 wt % ²³⁵U outer tubes (2.34-in. OD; 1.79-in. ID) containing 1.002 wt % ²³⁵U inner tubes (1.18-in. OD; 0.94-in. ID); and 1.25 wt % ²³⁵U outer tubes (2.37-in. OD; 1.80-in. ID) containing 0.05 wt % ²³⁵U inner tubes (1.18-in. OD; 0.94-in. ID). Maximum bucklings for the tubes were found to be 20.00, 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)


EVALUATION OF SOME UNCERTAINTIES IN THE COMPARISON BETWEEN THEORY AND EXPERIMENT FOR REGULAR LIGHT WATER LATTICES

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 codes is established with regard to leakage, fast fission events in \(^{235}U\), resonance capture in \(^{238}U\) and thermal neutron capture factors. The computer code NAMuke was used for the calculations.

The results of a calculation for a lattice of \(^{235}U\) using both critical and exponential techniques are given and further comparisons are made for a selection of experiments using both \(^{235}U\) and \(^{238}U\) metal fuel. The results of reducing the leakage by poisoning with \(^{238}U\) in these experiments are also described. The results obtained support the need to introduce a modification to the resonance integrals for \(^{235}U\) 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-independent error in the fast fission section and a discrepancy in temperature coefficient are identified, but efforts to isolate the cause of these errors were not successful.

CRITICAL EXPERIMENTS: LATTICES

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 \(\text{UO}_2\) 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 allow qualitative examination of some of the effects associated with different void spacings.

CRITICAL EXPERIMENTS: LATTICES

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 Monteuccolino 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 \(k_{enr}\) of natural-U/graphite lattices by means of zero reactivity activity. The same applies to the procedure for analysis of the experimental data. The error in \(k_{enr}\) 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.

Critical Experiments: Lattices

CANDU-BLW EXPERIMENTS IN ZED-2: PART III. BUCKLING AND LOSS OF COOLANT EXPERIMENTS

Experiments have been performed in simulated CANDU-BLW lattices in ZED-2. The lattice consisted of square arrays of 26-rod \(\text{UO}_2\) cladding at a spacing of 27.94 cm or 30.48 cm. The experiments were carried out to determine the residual fuel volume of the lattice with \(\text{LiCl}\) or air as "coolants," and the flux perturbation and reactivity effects of removing the \(\text{LiCl}\) coolant from 50% of the fuel assemblies in three geometric arrangements. The buckling for the \(\text{LiCl}\)-cooled lattice was 1.154 ± 0.031 mm, and for the air-cooled lattice 3.549 ± 0.035 mm. 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.
1967


The intra-cell structure of the fast neutron flux was measured in several TRX lattices with 235U (fission) and Al and AlO2 absorbers. The lattices were light-water-moderated, with cylindrical, 0.367-in. dia fuel rods of slightly enriched U. The fuel rods were arranged in hexagonal arrays, with h2O: U volume ratios of 1.0, 2.35, 4.02, and 8.1. Measured activation shapes and integral fuel 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)


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Neutron flux distribution and shielding 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 B2 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.)

Critical Experiments:


A series of experiments were completed to determine the critical parameters of lattices of 235U 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 UO2(NO3)2 solution of two concentrations (to simulate dissolver environments) as moderator and reflector; one solution concentration was 3.97 g of UO2/1 and the other was 8.02 g/1. 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)


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 submersed. Seven outer annuli in the same pattern were critical when separated 1.50 in., and seven inner annuli were subcritical when in contact. (auth)

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.; Johnson, E. B. (Atomic Energy of Canada Ltd., Chalk River (Ontario)). Dec. 1966. 46 p. Dep. mmm. CFSTI. CAN. $3.00 cy., $0.65 mn. AECL $1.00.

Experiments have been performed in the ZED-2 critical facility to determine the influence in Mn and Lu 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 Tn. and compared with values predicted by the Chalk River lattice reaction code LATRTP. (auth)

5822 (NW-3197) PLUTONIUM METAL DISOIL 1 9h.


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 homogenous solution system, whichever is smaller. The enriched uranium metal - uranium solution system was also studied. Experiments are suggested to verify the results. (M.L.G.)
Critical Experiments:

Lattices


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 wt% enriched UO2 clad in 0.8 mm thick Al. 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 0.9955 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 thermal, epithermal and fast neutrons amount to about 20, 19, and 17%, respectively, in the reflector region adjacent to the core. The fact that these discrepancies cannot be removed by multigroup P1 calculations would point toward insufficiency, of the diffusion or P1 calculation in this region. (auth)


A series of small-scale 3-dimensional arrays of 3-kg cylindrical 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: 2P (2 x 2 x 2) arrays, bare, reflected and moderated; 2 x 2 arrays of 6-kg pairs bare and moderated; 3P arrays bare and reflected; and 3P 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)


Physics calculations, preliminary to experiments, for determining cell-averaged lattice constants are presented. Calculations were made for 7.65 wt % and 23.5 wt % 240Pu 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 40Pu cores. The calculated thermal neutron energy spectrum is shown. (M.L.S.)


The angular distribution of neutrons emerging from a sphere of Orapel-90 (metallic uranium containing 90% 235U) and the critical mass of two interacting spheres of Orapel-90 were measured. The following assemblies were investigated: a solid sphere 13.5 cm in dia without a reflector; a solid sphere 18.5 cm in dia without a reflector; a solid sphere 16.7 cm in dia without reflector; a sphere 16.3 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 18.5 cm and L = 25 cm described above. (TTT)


A series of critical experiments using mixed-oxide (PuO2-UO2) plutonium fuel was carried out at the Westinghouse Reactor Evaluation Center (WREC). Two plutonium fuels with a variation in the 238U isotop 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 6% 235U fuel and two clean lattices with the 24% 235U fuel. With the 8% 238Pu 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. In single and multi-region cores. In single and multi-region cores. Water holes and water slot power peaking effects were measured in clean and borated cores. Power distribution measurements were made in cores containing concentric regions of different fuels, multi-region slab cores, and in cores containing interspersed fuels. (auth)
1967


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 UO₃ and U metal with D₂O, He and organic (HB-40) coolants. 18 and 22 cm triangular pitch lattices. A two-group heterogeneous reactor calculation program (MICRET) was used to determine the bucklings of the test rod lattices from the critical height changes. For D₂O-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 in general. The results are in agreement with those made in lattices containing a large number of rods. (auth)


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, D₂O-moderated lattices. Three different fuel assemblies were used and data were obtained in both bare and reflected systems. From the measured decay constants, k, the infinite reproduction constant, and Bₐ, the material buckling, were evaluated. Theoretical studies were made of the multiplicity, multigroup subcritical and the experimental decay constants of both bare and reflected systems agreed well with a two-region, two-group model. From the combined pulses and static experiments, the dispersal law for multiplying media was derived. (auth)

14641 (AE-254) BUCKLING MEASUREMENTS UP TO 250 MHG ON LATTICES OF AGOSTA CLUSTERS AND ON D₂O ALUMINUM IN THE PHISIUIZED EXPONENTIAL ASSEMBLY TZ. Persson, R.; Andersson, A. J. W.; Wilkahl, C.-U. (Aktiebolaget Ahlén & Örsberg, Stockholm (Sweden)), Nov. 1966, 58p. Dep. in. buckling determinations by means of flux mapping were performed in TZ up to 250 MHG on two lattices of Agosta fuel assemblies in D₂O and on D₂O alone. Most of the flux measurements were made with fiducial 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 (AgHCO₃) 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 1.6 m⁻², respectively, for the two lattices. During part of the experimental period about 30% unexplained excess absorption occurred in the heavy water. (auth)

Critical Experiments: Lattices

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 Högskolan, Göteborg (Sweden)). 1966. 15p. Dep. The thesis is an interpretation given to the various kinds of buckling measurements performed in heavy water moderated experimental and critical facilities. Problems connected with boundary transients, anisotropy, and heterogeneity are discussed. The experiments are adapted 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: 17366; NSA 18: 33053; NSA 19: 17206; NSA 20: 40377; NSA 21: 45677; NSA 22: 41661; 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)


19457 (BNL-50012) ORGANIC-COOLED, HEAVY WATER-MODERATED, Pu-239 FUELED LATTICE EXPERIMENTS. Price, G. A.; Windsor, H. H.; Tumney, W. J.; Hellstrand, E. (Brookhaven National Lab., Upton, N. Y.). Aug. 25, 1966. Contract AT(30-1)-Gen-16. 38p. Dep. in. Material neutron bucklings and thermal flux activation rates were measured for four lattices containing 239Pu and ThO₂ with D₂O 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 kₚₚ. There is a need for improvement in the reactor calculations, although the source of error is not obvious. (auth)


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 experiments were performed in heavy water moderated experimental and critical facilities. Problems connected with boundary transients, anisotropy, and heterogeneity are discussed. The experiments are adapted 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: 17366; NSA 18: 33053; NSA 19: 17206; NSA 20: 40377; NSA 21: 45677; NSA 22: 41661; 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)

A bibliography of 100 references is included. (J.C.W.)
1967


Calculated values of the extrapolation distance for water-reflected 7–235U–232Th 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 238U–235U–232Th 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 tools 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 238U–235U–232Th 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, Germany). 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)


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 235U concentrations, 1.143 and 1.028%, and three spacings, 1.55, 1.75, and 2.50 in. The following quantities were measured in each lattice: the ratio of epichalcium to cadmium capture rates in 235U (p28); the ratio of epichalcium to subcadmium fission rates in 235U (p25); the ratio of the total capture rate in 235U to the total fission rate in 235U (C/t); the 235U-to-238U fission ratio (p25); the intracellular distribution of the activity of bare and cadmium-covered gold foils; and the axial and radial activity distributions for 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 Icak 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 p25, p25, and C/t 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 p25, p25, and C/t 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)


1967


Methods for estimating the number of components required for criticality of unreflected and paraffin reflected systems of sub-critical 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)


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. Fission activation analyses and time-of-flight spectral measurements were made in erbium nitrate. The and of the first experimental lattice was calculated. The possible use of a pressure vessel for elevated temperature measurements was evaluated. (M.L.S.)


Using spontaneous fission as the sole primary neutron source, measurements were made of the lattice constants of a sub-critical assembly fueled with natural uranium rods of 1.2 in. diameter and moderated by water ($V_e/V_a = 1.42$). The experimental procedures are 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/cm² s; nevertheless the buckling was determined as $(6.80 \pm 0.42) \times 10^{-4}$ cm², a value in good agreement with that obtained from conventional exponential experiments. (auth)


The neutron multiplication of high isotopic purity $235\text{Pu}$ assemblies in various planar arrays in both water and air media was determined. The isotopic distribution of the $\text{Pu}$ was approximately $98\% 239\text{Pu}, 1\% 238\text{Pu}, 2\% 240\text{Pu},$ and $1\% 241\text{Pu}$. Two $242\text{Pu}$ assemblies were placed in a cylindrical calorimeter pressure containers, each containing approximately 130 grams of the $242\text{Pu}$ isotope, were assembled in three planar arrays consisting of 3-in., 1-in., and $\frac{1}{4}$-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 = 1.025) did exist in the $\frac{1}{4}$-in. planar array but was not large enough to be used in reliably extrapolating to the critical mass of $242\text{Pu}$. (auth)

Critical Experiments: Lattices


The neutron multiplication of high isotopic purity $235\text{Pu}$ assemblies in various planar arrays in both water and air media was determined. The isotopic distribution of the $\text{Pu}$ was approximately $98\% 239\text{Pu}, 1\% 238\text{Pu}, 2\% 240\text{Pu},$ and $1\% 241\text{Pu}$. Two $242\text{Pu}$ assemblies were placed in a cylindrical calorimeter pressure containers, each containing approximately 130 grams of the $242\text{Pu}$ isotope, were assembled in three planar arrays consisting of 3-in., 1-in., and $\frac{1}{4}$-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 = 1.025) did exist in the $\frac{1}{4}$-in. planar array but was not large enough to be used in reliably extrapolating to the critical mass of $242\text{Pu}$. (auth)

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 the 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 DYNSTEEL at A.E.E., Winfrith, in a small central fast reactor zone driven critical and equipped with a surrounding thermal reactor zone. This system produces 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/cm² equivalent steam density); the fissioned condition was fit with removing the plates.

35641 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 is discussed. Thermal studies and neutronics calculation 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.)


From Symposium on Heavy-Water Power Reactors, Vienna, Austria. See STI/PUB-165; CONF-670917.

A series of k measurements by the zero-reactivity method and by detailed parameter determinations has been carried out jointly by CEN 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 ORIGEL-type reactor, to investigate the feasibility of the zero-reactivity method and to test 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-rood uranium metal and 7-rood 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 9 ORIGEL elements which represent 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 result. (auth)


From Symposium on Heavy-Water Power Reactors, Vienna, Austria. See STI/PUB-165; CONF-670917.

A series of k measurements by the zero-reactivity method and by detailed parameter determinations has been carried out jointly by CEN 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 ORIGEL-type reactor, to investigate the feasibility of the zero-reactivity method and to test 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-rood uranium metal and 7-rood 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 9 ORIGEL elements which represent 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 result. (auth)

Critical Experiments: Lattices
1968


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 summarized; measured and calculated values are compared graphically. (M.L.S.)


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.)


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 235U and the importance function are also presented. These experimental values are compared with values calculated from transport theory. (auth)


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)

Critical Experiments:

Lattices


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 coefficients 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 measurement and design studies, and the B group, transport theory code, WIND, which has superseded TIME1. (auth)


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 buckling, using moderator to fuel volume ratios from 3:1 to 0.75. One assembly was heated to 100°C; in addition, the coolant density change on heating to 250°C was simulated by inserting aluminium void pins into the lattice. (UK)


The measurement of the infinite multiplication factor of natural uranium-graphite lattices in the critical assembly RX-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, 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 MARFIE, 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)


The results of neutron multiplication measurements performed with arrays of $^{238}$U solution apply to criticality safety considerations in handling solutions at a concentration of $10^9$ $^{238}$U/liter and are useful in checking computational methods. The measurements were made with $17.3$ kg $^{238}$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 0.02 of unity for those cases compared. (auth)
Critical Experiments: Lattices


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

25086 (IEA-139) PROGRESSIVE SUBSTITUTION EXPERIMENTS IN UO2 LATTICES MODERATED BY D2O-H2O MIXTURES. Franzen, H. R. (Instituto de Energia Atomica, Sao Paulo (Brazil)). Apr. 1967. 30p. Dep.

Buckling measurements for cores of uranium oxide (3.0% enriched in 235U) in different mixtures of D2O/H2O 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 D2O concentration of 99.50% and a lattice pitch of 6.644 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)


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

1968


Experiments on graphite-uranium lattices are compared to the CONGRIAF 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 elements carried out in the heavy-water and graphite facilities Avignon and César are also compared with calculations. (auth)


Reactor physics parameters were measured in three heavy water lattices consisting of 0.25-in. D2O 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 epilumadium to subcadmium radiative captures in 235U; the ratio of epilumadium to subcadmium fissions in 235U; the ratio of radiative fissions in 235U to fissions in 238U; and the fissions in 235U to fissions in 238U. The maximum number of fuel elements available was 500, these having a useful length of 37.6 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.)


Critical Experiments:

Lattices

1865 (JPRS-42922, pp 192-197) CRITICAL ENERGY OF INTERACTING CRITICAL ASSEMBLIES OF FISSILE MATERIALS. Translated from pp 169-201 of Kriticheskoe Parametry Sistem a Delysschchamiya Veshcheutavami 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.)


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 results 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%188) 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 heterogenous 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 3.652 or 10.922 cm were centrally located in the 2.5 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) SP2 transport calculations with five energy groups of finite height cylindrical cells are used. The axial leakage rates per source point, obtained from the cell calculations, are incorporated into one-dimensional radially finite reactor calculations in obtaining subcritical leakage cross sections for each energy group to be used for axial neutron streaming out of the voided region of the reactor. This reduction factors, which are also obtained from the two-dimensional cells, are important especially for the homutron method that contains the most 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)


1968


A description is given of a series of experiments carried out in pursuit of basic reactor assessment studies on heavy water moderated lattices containing natural UO2 cluster fuel elements. The experiments, which involved measurements of material buckling and diffusion coefficients, were designed to test the results of the measurements on single-zone lattices in the Swiss subcritical assembly MINO in those given by substitution measurements in the Swedish reactor HO. 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 UK/NEA 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 Rodost (auth).

12032 (NAA-SR-Memo-12472) NUCLEAR SAFETY SRE CORE III FUEL ELEMENT STORAGE. Ketzlach, Norman (NAA-SR-Memo-12472) NUCLEAR SAFETY SRE.


1968


A series of small seed- and-blanket critical assemblies were studied at the Ilettis Atomic Power Laboratory. Rod-type seed fuel elements which contained either 235U or 239U were utilized so that a direct comparison could be made between the lattice character of the two fuels. Also, blanket regions which contained rod type elements with either natural ThO2 or 1 wt. % of 233U were evaluated. 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 1:2. The second was a square array having a central seed region surrounded by a tightly packed dry blanket with a metal-to-water ratio of about 0.2. All experiments on the two-type assemblies were conducted at 29°C temperature. A set of two of the wet blanket assemblies were studied in the High Fluence Test Facility at 100°F and 230°F. For critical experiments.


From an 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 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 places. Uranium metal rods of uniform diameter. The following compositions were compared: Control act: natural uranium; 5 set used as an intercomparison standard: uranium very slightly depleted or enriched in 235U (0.04% - 0.85% - 0.85%), 100 set to be studied containing plutonium: 1P: natural uranium + 0.015% plutonium; 2P: highly depleted uranium + 0.3% plutonium; 3P: similar to 2P but with plutonium containing a higher proportion of the isotope 239U. 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 235U 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. For critical experiments. Part I, Section C.


Experimental values for infinite medium neutron multiplication factor $k_c$ have been determined for a graphite moderated supercell utilizing 2 1/4 enriched U fuel and 2.93 wt. % 235U in a target in separate reactor tubes. In a three-to-one ratio, Determinations of $k_c$ were made with the coolant channels wet and dry to obtain an estimate of the reactivity coefficients due to water loss. Both a 5 x 5 and a 5 x 5 array of test cells were inserted in the PCTR test cavity for these measurements. Results of experimental measurements are listed. For critical experiments. Part I, Section D.


Lattice parameter measurements of heavy-water-moderated lattices of 19-element clusters of ThO2 containing 1.5 wt % enriched UO2 (93 at. % 235U). Bucklings were determined from critical subassembly measurements in the ZED-2 reactor using MHF-1, a two-group heterogenous reactor core. Neutron density distributions and Westcott spectrum parameters and $k_c$ were determined from measurements in full detectors. Measurements were made for 1/3, 1/4, and 1/5 full arrangements and for 22, 24, 26, and 28 full triangular lattice pitches. For critical experiments.


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 best values show systematic trends with pitch and type of fuel cluster, and with coolant. Explanation for these trends is sought. In addition, lattice parameter measurements on D2O-moderated lattices of 9-element rods of ThO2 containing 1.5 wt. % 235U are reported. The measurements were made for four coolants over a range of lattice pitches.


Lattice parameter measurements of heavy water moderated lattices of 19-element clusters of ThO2 containing 1.5 wt % UO2 (93 at. % 235U) are presented. Relative fissile, 235U and 238U fission rates, fast fission ratio, and conversion ratio measurements are described. D2O, air, H2O 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 HIASMERT codes are compared with measured lattice parameters. Both codes underestimate the buckling by up to 1 m" with the largest discrepancy at the highest lattice pitch. The calculations overestimate the neutron disadvantage factor and conversion ratio.


From the 4th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680901.

Critical Experiments:

Lattices:
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<td>Experimental data to establish criticality control specifications for</td>
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<td>enriched uranium rods undergoing dissolution are extremely limited. A</td>
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<td>principal difficulty in treating the problem theoretically is that</td>
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<td>the resonance absorption is confined in the aqueous solution in which</td>
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<td>the rods are immersed. The &quot;narrow resonance&quot; and &quot;infinite mass&quot;</td>
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<td>approximations are applied; and from this application, expressions</td>
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<td>are developed for treating resonance capture by an absorbing lump</td>
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<td>embedded in a moderator doped with the absorber. The computed change</td>
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<td>in the critical buckling of a heterogeneous array on replacing the</td>
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<td>water moderator by a uranyl nitrate solution is in good agreement with</td>
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<td>experimental results from survey calculations for 3 and 5 wt % UO₂</td>
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<td>rods latticed in uranium-water mixtures are given. It was concluded</td>
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<td>that for enrichments up to 5 wt % UO₂, dissolution vessels designed</td>
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<td>geometrically safe for water-moderated arrays of uranium rods will</td>
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<td>remain safe during the dissolution process. (auth)</td>
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<td>1968</td>
<td>CRITICALITY OF ARRAYS UNDERGOING DISSOLUTION.</td>
<td>Tanner, C. J.; Andrews, D. G. (Univ. of Toronto).</td>
<td>See CONF-67102, 14th Annual Meeting of the American Nuclear Society,</td>
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<td>1968</td>
<td>MEASURED CRITICAL SPACINGS OF PLUTONIUM ARRAYS.</td>
<td>Pierce, G. A.; Morton III, J. R. (Univ. of California, Livermore).</td>
<td>See CONF-680601, 14th Annual Meeting of the American Nuclear Society,</td>
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<td>AN EXPERIMENTAL COMPARISON OF THE FAST AND THERMAL NEUTRON BUCKLINGS</td>
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<td>IN A URANIUM DIOXIDE LIGHT-WATER MODERATED CRITICAL ASSEMBLY.</td>
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<td>33047 BUCKLING MEASUREMENTS IN A HEAVY-WATER SUPERLATTICE.</td>
<td>Tanner, C. J.; Andrews, D. G.</td>
<td>See CONF-67102, 14th Annual Meeting of the American Nuclear Society,</td>
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<td>The known criticality of uranium metal enriched to 93 wt % in U₂O₅</td>
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<td>as a subcritical cylindrical component in unreflected arrays is</td>
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<td>used to explore component geometry effects in such arrays. Three</td>
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<td>basic geometries are chosen: the cube, the sphere, and the</td>
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<td>cylinder with equal height and diameter. It is shown that the</td>
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<td>useful in establishing critical arrays equivalent to a given</td>
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<td>uranium density of the equivalent arrays are the same as the</td>
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<td>and component mass of the array multiplication factor are</td>
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1968


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 \(\gamma\) spectrometry

50788 (AE-530) STUDIES OF THE EFFECT OF HEAVY WATER IN THE FAST REACTOR FRO. Tiren, L. I.; Haekaneaon, of heavy water in the core are presented. The effect of the heavy water to 24 vol. per cent, contained in thinwalled Cu cans.


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)


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 at a 6.9° 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-baffle bars for the fuel cells contained 2.0° O.D. bales 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)

Critical Experiments:

Lattices


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 \(k_e\) 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)


Lattice studies of uniformly arrayed UO\(_2\) rods and of clustered elements of a nuclear supercooler reactor were performed and analyzed. Microparameters were measured and multiplication factors for the four-factor formula with \(^{233}\)U epithermal fissions were obtained. Empirical formulæ 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)


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 FLECH, which is based on multigroup integral transport theory. (auth)


III. REACTIVITY MEASUREMENT

1967


35671 (ANL-7320, p 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. Ceschini, G.; Gandini, A. (Comitato Nazionale per l'Enenergia Nucleare, Casaccia (Italy). Centro di Studi Nucleari); Zan Boni, I.; Faleschini, B. (Comitato Nazionale per l'Enenergia 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 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)


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. 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 simpler and gives satisfactory results for fair approximation. (auth)
1967

30202 DETERMINATION OF THE EFFECTIVE MULTIPLICA-

TION COEFFICIENT OF NEUTRONS BY THE MEASUREMENT

OF THE DIFFERENTIAL ACTIVITY. Dodelkin, T. S.; Shishkin,


Relations between the effective multiplication coefficients of

neutrons in reactors and the experimental values of reactivity coef-

ficients, determined by measurements of differential activities,

were established. Correction terms were determined in integral

form. (tr-auth)

19469 (FKF-473) INTERPRETATION OF DOPPLER COEF-

FICIENT MEASUREMENTS IN FAST CRITICAL ASSEMBLIES,

Fischer, E. A. (Kernforschungszentrum Karlsruhe (West Ger-


Analytical results are presented on Doppler experiments in

which the reactivity change due to heating samples in fast criti-

cal 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 516-23 of Coupled Reactor Kinetics, Choezim, C. G.; Koehler,


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.J.S.)

7923 MEASURED SODIUM-VOID COEFFICIENT AND FIS-

SION RATIO IN A LARGE ZONED FAST REACTOR. Karam,

It. A.; Kato, W. Y.; Main, G.; Rusch, G. K. (Argonne National


25730 CENTRAL REACTIVITY MEASUREMENTS ON AS-

SEMBLIES 1 AND 3 OF THE FAST REACTOR FHO. Lunden,

Stig-Olof (Aktiebolaget Atomenergi, Studsvik, Sweden), Nu-


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 FHO. 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 20% 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 spec-

trum perturbation caused by the experimental arrangement has

been analyzed and found to be rather important. (auth)

38689 FFT CRITICAL EXPERIMENTS: CONTROL-ROD

STUDIES ON ZPR-3. Long, J. K. (Argonne National Lab., Ill.);

Hess, A. L.; McVean, R. L.; Peraldi, P. J.; Ulrich, A. J.; Baird,


10: 269-70 (June 1967).

From 13th Annual Meeting of the American Nuclear Society,


7972 (GAM5-7230) NULL REACTIVITY MEASUREMENTS

OF MULTIPLICATION CHARACTERISTICS FOR HIGH-TEMPER-

ATURE GAS-COOLED REACTOR CORES IN THE MILDUR

HOT REACTOR CRITICAL FACILITY. Nichols, P. F. (General

Dynamics Corp., San Diego, Calif. General Atomic Div.), Aug. 23,

1966, Contract AT(01-1)-147. 36p. Dep. mn. CNS 11 F4.60 cv. 56.30

mm.

Measurements of the multiplication characteristics for LHR

were made in the modified LHR critical facility. The method of

measurement involves experimental determination of the physical

consistencies of a unit cell with a multiplication factor of 1:1

and 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: 159-92 (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 (BNW1-442) EFFECTS OF UO2 CRYSTALLINE

BOUNDING ON DOPPLER COEFFICIENTS CALCULATIONS

FOR FAST REACTOR SYSTEMS. Schenker, R. E. (Interte-

northwest, Richland, Wash. Pacific Northwest Lab.), June

1967, Contract AT(45-1)-1830, 12p. Dep. CFST.

The change in the Doppler coefficient due to crystalline binding

effects was studied for 3H and a UO2 lattice for fast reactor sys-

tems. The "weak binding," or the short compound nucleus lifetime

FSCNL) approximation was used, where the 3H atoms in the UO2

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 photon

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 effec-
tive temperature dependence was well represented by the Debye
model (3H = 620°K). The temperature dependence of the reactivity

was calculated for the ANL critical assembly, ZPR-111, 47, with

the result that incorporation of the crystaline binding effects

could the calculated Doppler coefficient (dx/dT) to diverge from the

"1/T" rule in the low temperature region, in qualitative agree-

ment with the experimental results of Reynolds and Stewart. (auth)
1967

22739 (GA-4868) RESULTS OF HTGR CRITICAL EXPERIMENTS DESIGNED TO MAKE INTEGRAL CHECKS ON THE CROSS SECTIONS IN USE AT GULF GENERAL ATOMIC.

Bardey, R. G.; Gillette, E. M.; Niranj, R. J.; Taylor, R. C.
Contract AT(04-3)-167, 117p. Dep. CFSTI.

Cross-section data are presented for $^{235}$U, $^{238}$U, $^{239}$Pu, $^{241}$Pu, and $^{242}$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 experimental uncertainties. (auth)

35636 CENTRAL REACTIVITY CONTRIBUTIONS OF $^{144}$Ce, $^{235}$Pu, AND $^{239}$Pu IN A BARE CRITICAL ASSEMBLY OF PlU-TUN- NED METAL.

Central reactivity contributions of gram-sized samples of $^{144}$Ce, $^{235}$Pu, and $^{239}$Pu have been obtained in a fast critical assembly of bare $^{235}$Pu in a spherical geometry. Resulting values are: $^{144}$Ce = (127 ± 5) cents/g at; $^{235}$Pu = (1393 ± 35) cents/g at.; $^{239}$Pu = (701 ± 20) cents/g at. From these data, the critical mass of a bare sphere of $^{144}$Ce is estimated to be (27.7 ± 2.5) kg at a density of 13.5 g/cm$^3$. (auth)

1968

12006 (ANL-7399) REACTOR DEVELOPMENT PROGRAM.

Critical assemblies—reactivity measurements for small/scale perturbations in ZPR-3 fast; core material samples for ZPR-3 Assembly 59; specifications for, mockup of FFTR in ZPR-3 assembly 19-41; reactivity worths of FFTR safety rods in ZPR-3, calculation of.


Critical assemblies—reactivity worths and rection rate ratios calculated for ZPR-3 and 4.


From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-660601.


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)


Capture cross section of, error analysis for, fission cross section of, error analysis for.

33
1967


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 set-up experiments. Results of reactivity measurements on control elements and on certain perturbed core configurations are given, together with a list of the measured backings of the reference core. (auth)


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. (ITT)


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), TIPSY (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 DANNET, a code for rearranging and altering the mode of the standard BCD ENDF/B library tape; ETOE, a code for preparing as MF library tape; and MC, a code for generating multigroup constants from microscopic neutron data. Calculational results have been compared with experiments as well as results obtained using other nuclear data libraries. (auth)

Reactivity Measurement


Prompt neutron decay constants have been determined for unreflected and moderated subcritical cylinders of enriched uranium (93.15 wt% 235U) by the Rossi-o technique. The cylinder diameters were 17.77, 27.93, and 38.09 cm and the heights, at these diameters, varied from 10.184 to 2.548, 8.431 to 5.399, and 7.602 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. transport theory calculations showed disagreements as large as 20%. Reactivities as much as 33 dollars subcritical were determined from the prompt neutron constant at delayed criticality and changes in the prompt neutron lifetime with cylinder height calculated by S, 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)


Prompt-neutron decay constants have been determined for unreflected and unmoderated subcritical cylinders of enriched uranium (93.15 wt% 235U) by the Rossi-o technique. The cylinder diameters were 17.77, 27.93, and 38.09 cm and the heights, at these diameters, varied from 10.184 to 2.548, 8.431 to 5.399, and 7.602 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, 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, 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)


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


An analysis of sodium reactivity measurements in fast reactor critical assemblies is presented. Volume I 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, reactivity, absorptive 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% while the ENDF/B data are under-reactive by 1.3% and 0.7% for the 235U and 238U-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)


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

1968


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 made 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. The perturbation theory analysis, cell-homogenized cross sections and fluxes obtained by flux-weighting over the CC14 using transport theory calculations of the spatial flux distributions. Overall calculations of the void reactivities with the MENDF/B and the ENDF/B 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 2. Regions with large leakage contributions tend to be consistently underestimated by about 15% with maximum discrepancies of 27% for all assemblies analyzed. Heterogeneity and transport theory effects on the void reactivities are found to be typically large for Assembly 48. (auth)

The summary is made of 6 published papers by the author and co-authors 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 $^{235}$U and $^{231}$U in some fast reactor spectra; and comparison of theoretical and experimental values of the activation Doppler effect in some fast reactor spectra.


Measurements of the ratio of $^{239}$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 $^{239}$U was detected by counting the $^{239}$Np $\gamma$-activity of irradiated foils using the $\gamma$-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 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


The neutron spectrum at the center of a large, dilute fast reactor was measured over the energy interval from 1 keV to 1 MeV. The 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 multigroup calculation. There exists fair agreement between the measured spectrum and the multigroup calculation. (auth)

35688 (ANL-7247, p. 128-9) COMMENT ON SPECTRUM MEASUREMENTS IN A LARGE, DILUTE PLUTONIUM-FUELED CRITICAL REACTOR. Brown, P. S. (General Electric Co., Piscataway, N.J., 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. (auth)


Measurements of space-dependent fast neutron spectra in water and graphite at 2.0 and 12.0 MeV for fluxes directed normally to the 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. (U.R.D.)


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 2.9 MeV. In the U-238 blanket, a proton recoil spectrometer was used to cover the energy range from 0.8 eV to 1 MeV. (auth)


Measurements of space-dependent fast neutron spectra in water and graphite at 2.0 and 12.0 MeV for fluxes directed normally to the core face with penetrations up to 60 cm are reported. Comparisons of the measurement values were made with those calculated using a a solid-state proton-recoil telescope was used to measure neutron energies. (U.R.D.)
1967


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 x 40 x 38 in³ 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 129Ce- Na۲ 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.)


The difficulties inherent in spectrum determinations in reactors at positions remote from the core center are considered, and the suitability of 4Li and 3He semiconductor scintillation spectrometers for this work is discussed. Spectra obtained in many positions in the DAPHNE reflector are compared with calculated spectra. (auth)


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 these 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


Neutron Flux Spectra


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 50-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)


The experimental results of the neutron induced reaction velocities in a large block of UO₂ (dielectric reflector of BR-1 reactor) are presented. The neutron spectrum in the large UO₂ block is investigated by the time-of-flight method and scintillation techniques. The results of the measurements are compared with the calculations of the neutron propagation in UO₂ by 25 group set of constants. The results give the possibility to estimate the accuracy of the set of constants and calculation methods. (auth)


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 background during the excitation using filters of lithium hydride (21.4 g/cm²) under conditions of good geometry. Some deviations of the measured energy distribution from the calculated and experimental fast neutron 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 E<sub>n</sub> = 2.5 MeV and 8 to 10% at E<sub>n</sub> > 6 MeV. The total cross section for carbon and lead at E<sub>n</sub> > 2 MeV satisfactorily agrees with published data. For lead and water below 3 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 ~5 MeV, a tendency to irregularities was observed. The dependence of the linearity of the stilbene crystal on the energy made it possible to determine the scattering from carbon samples. (tr-auth)
1968


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 235U; the core also contained 20% Fe, 10% Mo, and 10% 238U. 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. (T.T.T.)


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 16 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-904 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 238U sphere are consistently softer than the calculated values. The measured spectra in the 235U sphere are accurate enough to permit one to choose the better of the two cross section sets. (aut.)


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


Construction of the Subcritical Time-of-Flight Spectrometer 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.)

Neutron Flux Spectra


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


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. (Q.C.C.)


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.)


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 4Li glass detector for measuring intermediate neutron energy spectra were initiated. Some developments in detection and associated electronic equipment for measuring fast neutron spectra are also summarized. (D.C.W.)
NEUTRON CROSS SECTIONS

1. Sources of Data

1967


Input data from the Atomics International Evaluated Nuclear Data File (AIFENDF) 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 AIFENDF Angular Distribution Data Tape (ADDT). Multigroup filtering are performed in any of the formats required by the one-dimensional multigroup diffusion theory codes. ULCER, FAIM, FAIM-CF.LL, CFASAR, and CAESAR IV; the one-dimensional multigroup $\Sigma_t$, transport theory code, DTF; and the spectrum codes, FORM and AILMOE. (auth)


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


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


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 (SCIRS) to date. (S.F.L.)


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-495, second edition. (S.F.L.)


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.)
Neutron Cross Sections:

1. Sources of Data

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11932 (TID-23357) 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. 16, 1966. 327p. (EANDC-66-6U; CCND-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, 1968 and Oct. 15, 1966 is presented. Corrections and improvements to previous entries are included. (D.C.W.)


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 $^{235}$U and $^{238}$U 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 data of $^{235}$U capture data within the range of 10 to 100 keV. (auth.)

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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 (y,n) and (y,t) reactions has been included for cases in which the y-ray energy is less than ~15 MeV and the (y,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 as far as they refer to specific target nuclei. (S.F.L.)


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 $^{14}$N(n,$\alpha$) cross section. Recent re-evaluations of cross sections of spherical shell transmission also tend to reduce the disparity. Neutron scattering cross sections from 0.3 to 1.8 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 ~0.5 MeV. More data are needed from 1.5 to 5 MeV. Above 5 MeV the optical model can be used to interpolate between measurements rather widely spaced in energy. Recent $\sigma(E)$ measurements confirm the nonlinearity of $\sigma(E)$, 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.)


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, $^{232}$Th/$^{238}$U, and $^{235}$U/$^{232}$U fission ratios caused by placing shells of graphite, sodium, aluminum, iron, 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, Bueyter, and Carter. Comparisons of the experimental data with values calculated using two multiplet cross-section sets show clearly where these data sets are accurate and where they are in error. (auth.)

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Neutrons, Fast — neutrons (n,0) with plutonium-239 and -240 at 5 to 150 keV, cross sections for (E)

Neutrons, Fast — neutrons (n,0) with uranium-235 at 30 and 64 keV, cross sections for (E)

Neutrons, Fast — neutrons (n,0) with uranium-235 at 7 to 14 MeV, neutron emission in (E)
1967


The breeding gain of plutonium fueled fast reactors is strongly influenced by the capture-to-fission ratio $q_2$ of $^{239}$Pu. In the softer spectra associated with a large dilute fast reactor, the uncertainty in $q_2$ is of the order of $\pm 25\%$. 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 $k_\infty$ near to unity, so that $k_\infty$ 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, is then measured in this modified lattice; and $q_2$ 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 $\gamma$ 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 $\gamma$ rays, and the apparatus is calibrated by repeating the measurement in a thermal neutron beam for which $q_2$ is known. Some preliminary results obtained by the first technique indicate that current nuclear data sets underestimate $q_2$ significantly in dilute fast reactor lattices. (auth)


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)


A computer program, GAF, was written to compute neutron fluxes and currents from the $\beta$ 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 59 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)


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.T.S.)
1967

35382 (ANL-7370, pp 27-30) THE AUTOMATIC PH-PARA-
TIZATION OF MULTIGROUP CROSS SECTIONS FOR FAST REACTOR
ANALYSIS USING THE MC CODE. O'Shea, D. M. (Computation
Engineering, Inc., Windsor, Conn.). Toppel, R. J.; Rege, A. I.
(Argonne National Lab., Ill.).

The fully automated multigroup-cross-section preparation pro-
gram MC makes use of an evaluated nuclear data file (ENDF) as
its basic input. The degree to which the ENDF data is initially sub-
divided preparatory to averaging in variable MC is var-
ious in the resolved resonance region are calculated through use
of Doppler-broadened line shapes and include effects due to inter-
ference scattering and the influence of overlapping resonances in
various isotopes in the mixture. Unresolved resonance cross sec-
tions 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 endpoints of line segments taken
from In E vs n, In E vs p, or F vs graphs. Inelastic scattering and
n,pt matrices are computed from excitation functions for indi-
vidual 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 dis-
tribution data for scattering. A fundamental-mode-weighting spec-
trum for the problem composition may be obtained for either the
P1 or the constant P1 or P1 approximations. Iteration on buckling
to criticality is optional. The generated spectrum is used to con-
tract the fine-group cross sections to a specified set of broad-
group multigroup cross sections. (auth.

1968

33966 (PAEC(AIN)661) REACTOR CROSS SECTIONS.
A Literature Search. (Philippine Atomic Energy Commission,

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.)

35393 (ANL-7370, pp 47-53) PRESENTATION OF THE
MULTIGROUP CROSS-SECTION SET PREPARED AT CADA-
RACHE. Ravier, J.; Chaumont, J. M. (Commissariat a l'Ener-
gie Atomique, Cadarache (France), Centre d'Etudes Nu-
cleaires). 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 ref-
rences is included. (auth.

35885 (EAND(E)-76(U)) INTRODUCTION TO KFK 120
"NEUTRON CROSS SECTIONS FOR FAST REACTOR MATERI-
ALS." PART I. "EVALUATION." Schmidt, J. J. (Kernfor-
schungszentrum, Karlsruhe (West Germany), Institut fuer Neu-
The neutron data evaluation that has been performed at Karla-
rueh since the beginning of the fast breeder reactor project in
1960 is outlined. (D.C.W.)

46683 (ANL-7370, pp 197-200) NUCLEAR CONSTANTS,
Stupple, C. D.; Tevebaugh, A. D.; Bingle, J. D. (Argonne Na-
tional Lab., Ill.).

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

1968

19886 (ANL-7375, pp 176-7) NUCLEAR CONSTANTS.
(Argonne National Lab., Ill.).

Preliminary results to the program to measure capture cross
sections for fast reactor materials and to measure and calculate
capture-to-fission ratios for 238Pu, 235Pu, 239Pu, 232Th, 235U, and
238U in EBR-II as a function of position are reported. (D.C.W.)

53165 (ANL-7435, 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.)

18015 THE U.S. EXPERIMENTAL PROGRAM FOR FAST
REACTOR PHYSICS. Avery, R.; Dickerman, C. E.; Ahto, W. Y.;
Long, J. K.; Smith, A. B. (Argonne National Lab., Ill.). pp 403-
20 of Fast Breeder Reactors. Evans, P. V. (ed.). Oxford, Per-

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

1 Progress in basic cross-section measurements in the resonance,
intermediate, and continuum regions is described. Critical experi-
ments are discussed. The final conclusions resulting from the first
and second test programs reviewed include the

SEFOR project, and the out-of-pile and TREAT experiments related
to safety. (UK)

148414 (GA-8773) INTEGRAL NEUTRON THERMALIZA-
TION, Quarterly Progress Report for the Period Ending June 30,
J. U.; Suggs, E. L.; Sprevak, D.; Young, J. A. (Gulf General
Atmoic, 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 theoret-
ical scattering law for UO2 was completed; however, some addi-
tional numerical studies to be done before the work can be incor-
porated 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 DeO, including cohered 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
specally constructed samples designed to reduce multiple scat-
tering. Reports on work in progress include an outline of efforts
being made to improve capabilities for computing coherent elas-
tic scattering. Also some preliminary work is reported con-
cerning efforts to broaden significantly the scope of the theoret-
ical analyses underlying ENDF scattering laws by relating the
temperature dependence of the frequency spectra to inelastic
effects. A report of the UC total cross section experiment and
the analysis of the data constitutes the section on experimental
studies. (auth.)
**1968**


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


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.)

**1967**


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

**1966**


From IAEA Conference on Nuclear Data, Part 2: 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 eV 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, with fluctuation effects, and nuclear determination is presented. The influence of each on calculation is illustrated. (auth)

**1966**


A bibliographic guide to experimental and theoretical information on neutron cross sections, resonance parameters, thermal scattering data, 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 incident neutron energy range; the complete reference for the document; 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.)


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.)


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

Neutron Cross Sections:

1. Sources of Data

For abstracts of individual papers see: 18099–18105, 18146, 18147, 18192–18195, 18281–18289, and 18347.
1968


Some recent measurements of differential cross sections and spectra for the reactions $^1H(n,p)^2H$, $^1H(n,p)^3H$, and $^2H(n,\alpha)^4He$; the neutron total cross section of $^4He$; spectra from resonance and thermal neutron capture by $^{10}Be$, $^{11}Be$, $^{14}Be$, $^{19}F$, $^{20}F$, $^{20}Tl$, $^{22}Na$, $^{24}Na$, and $^{38}Sr$; and capture and fission cross sections for $^{238}Pu$ from thermal to 30 keV and for $^{235}U$ from thermal to 1 eV are summarized. Calculations of the elastic and inelastic scattering cross sections for $^{55}Fe$ are also reported, as is the status of the electron linear accelerator. (D.O.W.)


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 consist 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 concerted 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 $^{233}Pu$, $^{234}Pu$, and $^{238}Pu$ 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)

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Neutron Cross Sections:

1. Sources of Data


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 considerations 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


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 still are 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, nα) 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.


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 235Pu and 237Am is reviewed. An analysis of Tweed abundances is made to obtain capture-to-fission ratios for the odd- A pluttum isotopes through A = 235. 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 > 101 and N < 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 useful. However, for predicting properties of nuclei having Z > 104, N ~ 159, one needs, in this formalism, an accurate way of predicting shell corrections or nuclear masses.


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 6Li and 7Li for the first version of the Evaluated Nuclear Data File/B (ENDF/B) tape. These data were assembled in the ENDF/B format and were sent to the Cross Section Evaluation Center (CSEC) at Brookhaven National Laboratory. Most of the data are from the AWE data file originally by K. Parker of Aldermaston. Values for S_{i} and Q, along with integration coefficients for the elastic scattering angular distributions, were received from H. Alter of Atomic Energy, Plut of the original AWE cross section data converted to the ENDF/B format are presented. The ENDF/B listings for the 6Li and 7Li data, as they appeared on the first version (approximately February, 1967) of the ENDF/B tape, are shown. (auth)


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

Some measurements of the capture and fission cross sections of 234U in which target foils were exposed to a moderated flux from a nuclear explosion are summarised. Relation fission cross sections are presented for energies of 20 to 10^8 eV, as well as information on the capture-to-fission ratio between 27 and 63 eV. The results of resonace analyses using both single-level and multilevel formalisms indicate that the fission of 234U takes place through several channels. (D.C.W.)
1967


Thesis. Submitted to Univ. of New Mexico [Albuquerque]. The 235U fission and capture cross sections were measured using a nuclear-decay 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 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)


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 gamma rays following inelastic scattering. (auth)


From International Symposium on How to 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 enormous nuclear 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 radioactivity 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 1964 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 109Pd (2.7 y), a fissile product from which, through neutron capture, the reactor poison 110Pd is synthesized. Plans for future measurements include the capture cross section of 113Pd (81 d), 144Pd (27.4 d), and probably the fission cross section of 238U (26.2 y) which would require fast chemistry under field conditions. Measurements on nuclides increasingly farther 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 fission-mixture neutrons. (auth)


Evaluated neutron cross section data for the nuclides 238Pu, 244Pu, and 249Cm 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^-3 and 1.5 x 10^7 eV. These data in the ENDF/B format are available from the Cross Section Evaluation Center at Brookhaven National Laboratory. (auth)
were investigated using a pulsed 50-MeV electron beam. The sensitivity of a neutron-capture detector to prompt neutron scattering was investigated. The capture cross sections of Al, Fe, Zr, Fe, Ni, Ti, Ni, and Na were measured at energies from about 10 keV to a few hundred keV. The total neutron cross sections of C, T-, W, W, W, and W were measured from 0.1 to 20 MeV. Gamma spectra from neutron capture by Mg, Mg, Mg, and W were analyzed; transition strengths were obtained. The average neutron number/bolton of $^{15}$ was measured over the neutron energy range from 2.5 to 5.0 eV, and the scattering cross sections of $^{15}$, $^{15}$, and $^{15}$ were measured for neutron energies between 1 and 100 eV. Preliminary transmission measurements of the neutron total cross sections of $^{15}$ and $^{15}$ 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 $^{15}$ resonance. 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, NH$_2$, CH$_2$, and UCN. Cross sections were also written to ionization data reduction. Transient radiation effects in silicon were investigated using a pulsed 50-MeV electron beam. (D.C.W.)

**1967**


The sensitivity of a neutron-capture detector to prompt neutron scattering was investigated. The capture cross sections of Al, Fe, Zr, Fe, Ni, Ti, Ni, and Na were measured at energies from about 10 keV to a few hundred keV. The total neutron cross sections of C, T-, W, W, W, and W were measured from 0.1 to 20 MeV. Gamma spectra from neutron capture by Mg, Mg, Mg, and W were analyzed; transition strengths were obtained. The average neutron number/bolton of $^{15}$ was measured over the neutron energy range from 2.5 to 5.0 eV, and the scattering cross sections of $^{15}$, $^{15}$, and $^{15}$ were measured for neutron energies between 1 and 100 eV. Preliminary transmission measurements of the neutron total cross sections of $^{15}$ and $^{15}$ 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 $^{15}$ resonance. 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, NH$_2$, CH$_2$, and UCN. Cross sections were also written to ionization data reduction. Transient radiation effects in silicon were investigated using a pulsed 50-MeV electron beam. (D.C.W.)


A method for measuring neutron total cross sections using a neutron intensity spectrometer in continuous 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 Po 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)


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. (J.C.W.)

11919 (BNL-325(2nd Ed.)Suppl.2(Val.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.-6. 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.)

**1978**


An extensive evaluation of $^{238}$U 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, New York. 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 $^{238}$U 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)


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 invented. 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 $^{235}$U, $^{238}$U, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{242}$Pu, $^{243}$Am, and $^{244}$Am are reported. In addition, capture-to-fission ratios of $^{232}$U and $^{235}$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)
1967


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.)


The available resolved resonance data for the isotopes of zirconium are evaluated, with the view to compile a file of band-integrated 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 $^{92}$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


A compilation of neutron cross sections and resonance parameters for C, Cr, He, H, Fe, Mo, Ni, O, 139Pu, Na, 197U, and 238U 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.)


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 239Am and 240Am 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 239Am. The fission cross section of the doubly odd nuclide 239Am is about twice that of 239Pu over most of the neutron energy range, but only about 20% greater at 1 MeV. (auth)


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 activation spectrometry. (D.C.W.)


Theoretical studies were made of N-N scattering, electron scattering by polyatomic molecules, A-N interaction, (p,He) the four-nucleon system, Z-N interaction, and s-s scattering amplitudes. The fission cross sections of 239Am and 239U were measured from 0.02 eV to 6 MeV and from 0.004 to 2 keV, respectively. Threshold photon-neutron cross sections for Fe and Be were also measured, as were the photon-neutron cross sections of H, H, H, Zr, Zr, and Zr up to photon energies of 30 MeV. Cross sections for the reaction 239(N,n)239C were obtained from measurements of the cross section of the reaction 239(C,p,α)239N. The reaction 239(C,p,α)239N 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 239(P,n)239He were analyzed using a zero-range DWBA formalism. Differential cross sections for (α,n) reactions on 240 and 240 were measured at 9.8, 11.5, and 12.2 MeV; and evidence was also obtained for the (n,α) reaction on 239Pu for slow neutrons. The neutron scattering cross sections of 239Pu and 239U from 2 to 32 eV and from 2 to 22 eV, respectively, were studied further. Data were also obtained on the de-excitation γ rays following (γ,n) and (γ,p) reactions on 240; and the spontaneous fission half life of 239Am was measured, as were the thermal-neutron capture cross sections of Ca, 42Ca, 43Ca, and 44Ca. 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.)

Neutron Cross Sections:

2. ENDF/B Tapes...
1968

16002 (GA-6133) NEUTRON CROSS SECTIONS FOR NIO-

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 232U FISSION CROSS
DC-6946).

The neutron-induced fission and capture sections of 232U were measured by time of flight with a nuclear detonation as the neutron source. Cross-section data are presented from 20 to 100 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 Breit-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 238U FISSION AND CAPTURE CROSS SECTIONS

The 238U 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 100 eV for fission and from 20 to 63 eV for fission + capture. The resonance section (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.)

24873 SELECTED FISSION CROSS SECTIONS FOR 239Th,

The fission cross sections of 239Th, 233U, 237U, 235U, 10B, 239Pu, 235Pu, 234Pu, 233Pu, and 241Pu 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)

27289 (AERE-R-5244) THE AVERAGE NEUTRON TOTAL

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 a decreases by (101.3 ± 3.1) E^-1 + (1.95 ± 0.10) barns. The deviation of a 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 up to at least 260 keV. (auth)

16001 (GA-7462) NEUTRON CROSS SECTIONS FOR 235Pu,
3)-167. 85p. Dep. CFSTI.

A survey was made of the available experimental cross-section measurements for 235Pu, 239Pu, and 232U. Sets of recommended neutron cross sections and resonance parameters are presented for neutron energies from 0.001 eV to 15 MeV. (D.C.W.)
Neutron Cross Sections:

2. ENDF/B Tapes...


Neutron and gamma-ray production cross sections 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)


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

Results on neutron capture in 235U 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 Petred nuclear explosion. The total radiation width, $\Gamma_T$, was determined for 621 levels with $\Gamma_T = (1.9 \pm 0.6 \text{stat.}) \pm 1.4 \text{syst.}) \times 10^{-9}$ eV. There appears to be a significant variation in the value of $\Gamma_T$ from resonance to resonance. Approximately 200 weak resonances with $\Gamma_T \leq 2.0$ MeV can be identified. Analysis of these weak resonances, assuming $l = 1$, gives results consistent with an average reduced neutron width of $3.7 \pm 0.7 \times 10^{-4}$ eV; an average level spacing of $7.0 \pm 0.5$ eV; and a strength function of $1.8 \pm 0.3 \times 10^{-4}$.

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A survey was made of the available experimental cross-section measurements for $^{235}$U and $^{239}$U. Sets of recommended neutron cross sections are presented for neutron energies from 0.1 to 15 MeV. Resonance parameters are also included. (npw C.W.)


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 for 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)


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)


Thesis.

A method for measuring neutron total cross sections using a high-intensity neutron source continuously in energy ("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, Pd, Ag, Pb were measured to 1% average uncertainty in sets of 0.03 am/fm. 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. (Blaser. Abstr.)


A cubic spline curve fitting method and a statistical theory of unknown systematic errors are combined to give a practical computer-oriented 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 CFT graphical display unit. Among program output examples given are evaluated curves for several cross sections of $^{15}$B. (auth)
Neutron Cross Sections:

1. ENDF/B Tapes...


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.)


 Experimental and theoretical data for the neutron cross sections of deuterium are surveyed, and values are adopted for total and partial cross sections. To facilitate neutronic calculations, angular distribution functions based on n + d and the conjugate based on the latest evaluation of cross sections and resonances will be on the Karl shrue Nuclear Data File. KEDAK.


The neutron cross sections for D 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)


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)


The energy dependence of the neutron fission cross section of 242mAm was investigated from 0.02 eV to 6 MeV. The measurements were normalized to a value for the fission cross section of 235U at 0.0253 eV. Below 3.7 eV and above 300 keV, the cross section was corrected for 235U 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 was 0.45 ± 0.05 eV. (D.C.W.)


Work performed under United States-European 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, Hc, H, Fe, Mo, Ni, O, 235Pu, Na, 238U, and 239U. (D.C.W.)


The neutron cross sections for 10B were evaluated from 10^-4 eV to 15 MeV. The existing experimental data were 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)
Neutron Cross Sections:

2. ENDF/B Tapes...


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 235U 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)


An evaluation of the basic nuclear data for 239Pu, 241Pu, and 244Pu in the range 0.01 eV to 15 MeV was made. Partial cross sections are constructed for the thermal and fast neutron regions; 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 coine in the lab. system, and the energy spectrum of secondary neutrons from fission. (auth)

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V. NEUTRON CROSS SECTIONS

3. Wide Ranges in Energy

1967


The neutron total cross section of 239Pu was measured from 0.006 to 5500 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 l/v) thermal absorption cross section. Parameters of individual resonances below 200 eV and average parameters at higher energies give a resonance absorption integral of 1.10 ± 0.20 × 10^-4 for the s-wave neutron strength function (1/2π). (auth)


Measured values of η for 235Pu and 233U are given for the energy range 0.04 to 11 eV, together with fission and total cross sections for 235U. Average values of η were calculated and also, in the case of 233U, various ratios and integrals of the cross sections and η for energy groups of interest for reactor design studies. Total cross-section measurements were also made for 235Pu from 10 to 1000 eV. Comparisons are included between the results obtained in the experiment and those from other laboratories. (auth)


From IAEA Conference on Nuclear Data, Paris.

The Ψ 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(2H,n) reaction. Two standard neutron sources, a 241Am-Be source and the AWRE 239Pu spontaneous fission source, have 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 Ψ 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)


Some corrections to the cross-section data for 234Cm, 233Pu, and 235Pu that were presented in NAA-SR-12271 are summarized. (D.C.W.)


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 233Th(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 233Th 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 0.05 ± 0.02 barn and the value extrapolated from η(1/2π) at higher energies is discussed. (auth)
Fission Cross Sections: 3. Wide Ranges in Energy


The effective resonance integral and Doppler coefficient of $^{239}$Pu, $^{235}$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 code. (auth)


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 $^{144}$Pu, $^{145}$Pu, and $^{146}$Pu. (D.C.W.)
Neutron Cross Sections:

3. Wide Ranges in Energy

1967

21525 TOTAL NEUTRON CROSS SECTION OF $^{235}$Pa.
Simpson, F. B.; Coddington, J. W. Jr. (Phillips Petroleum Co.,

Transmission measurements on $^{235}$Pa were taken with the Ma-
terials Testing Reactor (MTR) fast chopper. The total cross sec-
tion was calculated in the energy range from 0.01 to 10,000 eV.
These measurements were made on 700 mg of chemically separated
$^{235}$Pa in an oxide form. The protactinium was produced by ir-
ducing 280 g of $^{233}$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/cm. The Bragg–Wigner
resonance parameters were obtained for the resonances below
16 eV. The average parameters give a value of $0.75 \times 10^{-4}$ for the
s-wave neutron strength function $F_s/D$. Weighting the level spac-
ing inversely by $1 + j$ 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 $g_f \cdot E$ data between 0.01 and 0.10 eV gives
2200 m/sec total neutron cross section of $5 \pm 3 \times 10^{-4}$ barns,
superseding a value of $57 \pm 5$ barns previously. The resonance-absorp-
tion integral for neutrons with energies above 0.4 eV was calculated to
be 901 ± 45 barns. (auth)

12014 (IN-1015) TABULATION OF THE TOTAL NEUTRON CROSS SECTION OF$^{235}$Pa.
Simpson, O. D.; Moore, M. S.; Berreth, J. R. (Idaho Nuclear Corp.,
Idaho Falls). Dec. 1966. Contract AT(11-1)-1230. 69p. Dep. m.$\text{CFSTI } \$3.00 cy, \$0.65 am.

The total neutron cross section of $^{235}$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 $163 \pm 10$
barns was determined. Multilevel parameters are listed for res-
onances 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$^{235}$Pu.
Simpson, O. D.; Moore, M. S.; Berreth, J. R. (Idaho Nuclear Corp.,

The neutron total cross section of $^{235}$Pu 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 $163 \pm 10$
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. (auth)

35457 (AEET-372) SELF-SHIELDED CROSS SECTIONS FOR
THE MAIN FERTILE AND FISSILE NUCLEI.
Singh, R. Shankar; Denal, G. A. (Atomic Energy Establishment, Trombay (India),

Self-shielded cross-sections for $^{232}$Th, $^{235}$U, $^{239}$Pu, and $^{241}$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 re-
sitivity coefficients due to the Doppler effect, etc. These were
extracted from resonance-integral calculations under the narrow-
resonance approximation using the latest available resonance pa-
rameters at four temperatures ($300, 750, 1500, 2500^\circ K$) and at
$g_f$ (potential scattering cross section per absorber atom) values of
$40$ and $60$ barns for $^{232}$Th and $^{235}$U and $126, 201, 300,$ and $400$ barns
for $^{239}$Pu and $^{241}$Pu. The status of resonance parameters for these
elements is also discussed in detail. (auth)

40457 CAPTURE CROSS SECTIONS OF $^{238}$U.
Slepian, Donald C.; Schmidt, Marla; Keedy, Curtis R. (Argonne Na-

Neutron-capture cross sections of $^{238}$U were measured at
200 neutrons energies between 0.15 and 1.5 MeV. The experi-
mental method was the activation technique in which the neptunium
target was irradiated with a monenergetic neutron beam and was
analyzed for the product $^{239}$Pu by gamma-ray spectrometry. (auth)

42436 THE FISSION CROSS SECTIONS OF $^{238}$U, $^{235}$U, $^{232}$U,
$^{239}$Pu, $^{240}$Pu, AND $^{241}$Pu RELATIVE TO THAT OF
$^{232}$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:

The fission cross sections were measured relative to the fia-
sion section of $^{232}$U to an accuracy of approximately ±2% at
neutron energies of 1.0, 2.25, 5.4, and 14.1 MeV. Combining these
results with the known values of the fission cross section of $^{232}$U
leads to fission cross sections having an estimated uncertainty of
±3.5% and which are mostly in agreement with other recent mea-
surements. (auth) (UK)

1968

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

The energy variation of $F$ for $^{239}$Pu was measured in 4 over-
lapping energy ranges from 0.01 eV to 10 keV; spin assign-
ments 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. Simulta-
neous capture and transmission measurements were made for
separated $\text{Hf}$ isotopes. Gamma pulse-height spectra from experi-
ments with the 1.25-m liquid scintillation capture detector are
presented for $^{240}$Pu and $^{241}$Pu. The angular distributions of neu-
trons scattered from resonances in Al were measured in order to
determine the feasibility of this approach to assigning spins to
1 to 6 resonances. The technique of using the large liquid scintilla-
tion detector as an anticoincidence mantle was extended to fission
elements; preliminary scattering data for $^{238}$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,
21-1. 5p.

Comparisons of values of $\gamma$ obtained from HAMMER calculations
for $^{233}$U and $^{238}$U and experimentally obtained values are made.
Physics parameters for $(1/\nu)$ $(dn/dT)$ lattices are tabulated. Ef-
tects of temperature variation are outlined. (M.L.S.)

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.)


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. Pulses of 14-MeV neutrons having a pulse width of about 1 usec 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 gamma 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)


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 resonance using the Harwell "loopless" pulsed neutron source and the 1.20-m and 300-m spectrometers. The s-wave strength function $S_{1}$ and distant level parameter $R_{d}$ 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 B-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 $^{99}$Mo, $^{95}$Mo, $^{133}$Rh, $^{127}$Th, $^{232}$U, $^{235}$U, $^{239}$Pu, and $^{235}$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 $^{235}$Pu is discussed. (D.C.W.)
1968


The neutron capture cross section and fission cross section for $^{235}$U were measured simultaneously in the neutron energy range 0.4 to 2000 eV. A pulsed and collimated neutron beam was passed through a $^{235}$U fission chamber placed at the center of a large liquid scintillator. Capture and fission events in the $^{235}$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)


The neutron capture cross section and fission cross section for $^{239}$Pu were measured simultaneously in the neutron energy range 0.4 to 2000 eV. A pulsed and collimated neutron beam was passed through a $^{239}$Pu fission chamber placed at the center of a large liquid scintillator. Capture and fission events in the $^{239}$Pu 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)


The neutron total cross section of $^{236}$Pu was measured from 0.008 to 6200 eV. These data give single-level Breit-Wigner parameters for resonances below 220 eV. The observed total cross section at 220 m/sec is 586 b. A value of 532 b was calculated for the effective (equivalent I/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 neutron s-wave strength function $(I^2/D)$. (auth)


The total neutron cross section of $^{234}$Pu was measured from 0.008 to 8000 eV using PuO, 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 $39 \pm 1$ b, and the effective (equivalent I/v) thermal absorption cross section derived from the measurement is $22 \pm 2$ b. Parameters of individual resonances below 180 eV and average parameters at higher energies give a resonance absorption integral of 10900 ± 60 b and a neutron s-wave strength function $(I^2/D)$ of $(0.85 \pm 0.10)10^{-4}$ (eV)$^{-2}$. (auth)


From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-162(VOl.2); CONF-871043-(Vol.2).

Integral values of $\sigma$ for $^{238}$U and $^{235}$Pu have been derived from experiments in two different cores (median $^{238}$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 ZERHA. The experiment includes reactivity measurements of the sample material and of a standard, $^{239}$Pu, as well as an absolute determination of the fission rate in $^{238}$Pu at $^{235}$U and the capture rate in $^{238}$U. The experimental values agree well with the calculated ones for the hard spectrum core measurements, both for $^{238}$U and $^{235}$Pu. The measured value for $^{235}$Pu in the soft core is slightly higher than the calculated one. The discrepancy for $^{238}$Pu is large and bitherto unexplained. (auth)

33086 (ANL-7310, pp 431-511) REACTOR COMPUTATION METHODS AND THEORY. (Argonne National Lab., Ill.).

URANIUM—neutron cross sections for, crystalline effects on Densities, broadening resonance integrals for, crystalline effects on F.M.L.S. (auth)

50605 ANALYSIS IN TERMS OF A GENERALIZED OPTICAL MODEL, OF THE CROSS SECTION OF $^{235}$U IN THE ENERGY INTERVAL (0.05 TO 15) MeV. Baldoni, Bruno; Saruia, Anna Maria (CNE, Bologna). pp 741-50 of Fisica del Reattore, Roma, Consiglio Nazionale delle Ricerche, 1966. (In Italian).

From Conference on Physics of Reactors, Milan. See CONF-668.

Use of a generalized optical model for analysis of the $^{235}$U cross section is described. Differential cross sections are shown for $F = 0.650$ 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 inclusive cross sections are shown. Behavior of the Legendre polynomial expansion coefficient is determined as a function of energy. (M.L.S.)


From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.1); CONF-871043-(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)
Neutron Cross Sections:
3. Wide Ranges in Energy


The available experimental data for the value of \( n \) for fissile and fertile isotopes are reviewed. The absolute determination of \( n \) for \( ^{235}\text{U} \) 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 \( ^{232}\text{Th} \), \( ^{235}\text{Th} \), and \( ^{238}\text{U} \). The available data relating the dependence of \( n \) 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 for practical purposes, a reasonably satisfactory fit to the data can be made with not more than two straight lines. (auth) (UK)


Recommended cross sections for \( ^{239}\text{Pu} \) and \( ^{240}\text{Pu} \) 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)

16053 INTEGRIAL AND DIFFERENTIAL CROSS SECTIONS OF \( ^{239}\text{Th} \) FISSION BY NEUTRONS. Smirenkina, L. D.; Smirenkin, G. N. 18p. (In Russian).


The energy dependence of the cross section \( \sigma(E) \) usually shows a complex structure in and fissilizable nuclei near the threshold. Characteristics of the lower fission channels can be derived by comparing the observed cross section \( \sigma \) 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 sequence of lower channels of the transitory \( ^{239}\text{Th} \) nucleus, which were excited in the \( ^{232}\text{Th} \) (n, f) reaction by neutrons having an energy \( E_n < 1.6 \text{ MeV} \): \( ^{23}\text{Y} \), \( ^{23}\text{Y}^\prime \), and \( ^{23}\text{Y}^\prime\prime \). The new channels with \( K = \frac{1}{2} \) explain the inflexion of \( \sigma_f \) in the region of neutron energy \( E_n \approx 1.1 \text{ MeV} \). (TTT)


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 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 space between the channels differed considerably. The heterogeneity of the system had an effect on the \( ^{232}\text{Th} \) fission reaction, but not on the \( ^{239}\text{Th} \) fission reaction. Both the fission chamber and foil activation methods were free of systematic errors as shown by experiments in a homogeneous region. (TTT)

Some recommended values for the number of neutrons per fission for \( ^{235}\text{Th} \) and \( ^{238}\text{U} \) in the energy range 1 keV to 14 MeV are presented. (TTT)
The capture cross section of $^{235}\text{U}$ was measured absolutely at a neutron energy of 30 keV using kinematically collimated neutrons from the $^7\text{Li}(p,n)^7\text{Be}$ 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 $^{235}\text{U}$ capture cross section measurement is $74.6\pm 1.4$ mb at 30 keV. In addition, the shape of the $^{235}\text{U}$ capture cross section was measured for neutron energies from 25 to 500 keV using neutrons from the $^7\text{Li}(p,n)^7\text{Be}$ reaction. The capture reactions in the $^{235}\text{U}$ 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 $^{235}\text{U}$ capture cross section as given by other authors. (auth)

The capture cross section of $^{238}\text{U}$ was measured absolutely at a neutron energy of 30 keV using kinematically collimated neutrons from the $^7\text{Li}(p,n)^7\text{Be}$ 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 $^{238}\text{U}$ capture cross section measurement is $74.6\pm 1.4$ mb at 30 keV. In addition, the shape of the $^{238}\text{U}$ capture cross section was measured for neutron energies from 25 to 500 keV using neutrons from the $^7\text{Li}(p,n)^7\text{Be}$ reaction. The capture reactions in the $^{238}\text{U}$ 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 $^{238}\text{U}$ capture cross section as given by other authors. (auth)


Relative fission cross sections of $\sigma_F/\sigma_o$ ($^{235}\text{U}$ to $^{239}\text{U}$) and of $\sigma_F/\sigma_o$ ($^{239}\text{Pu}$ to $^{235}\text{U}$) were determined over a neutron energy range 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 $^{235}\text{U}$ isotope is well known with a high accuracy of 2.5 to 2% at higher neutron energies, it becomes possible to derive more accurate data on the cross sections of $^{239}\text{U}$ and $^{239}\text{Pu}$. 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 Lampiere. (TTT)
"1968


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.)


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 20 keV. The capture cross section of $^{198}Au$ 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 Ca, 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


The ratios of the neutron reproduction constants \(\eta(\text{Pu-239})/\eta(\text{Pu-235})\) and \(\eta(\text{Pu-241})/\eta(\text{Pu-235})\) were determined from reactivity measurements in ARMF-I and ARMF-II. Results for 2200 m·sec and Maxwellian average values are given. (auth)


The Doppler effect of an enriched (93.2%) \(^{235}\)U target in a 1/\(\infty\) 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 \(\alpha\) and \(\gamma\), 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, \(\alpha\), 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)


Measurements were made of the \(\alpha\) (capture-to-fission ratio) for \(^{233}\)U and \(^{235}\)U. 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 \(\gamma\) values (at a \(\text{Cd} \cdot \text{sec} \) ratio for \(\text{Co} \cdot \text{sec} \)) and the computed 2200 m·sec average values are: 0.0976 + 0.0016 and 0.0986 + 0.0016 for \(^{233}\)U, and 0.1768 + 0.0016 and 0.1716 + 0.0015 for \(^{235}\)U, respectively. (auth)


The graded exposure of \(^{239}\)Pu-Al alloy, 19 rod clustered fuel elements, and the subsequent destructive sampling of the elements have provided experimental data showing the variation of \(^{239}\)Pu isotopes with irradiation. Irradiations were conducted in the heavy-water-moderated and -cooled Plutonium reactor. Using \(^{189}\)Ir as a fission indicator, the depletion of the initial \(^{239}\)Pu to 50.4 + 1.1% is determined, reactor effective cross-section ratios for the \(^{239}\)Pu isotopes are derived from the data, and results show that the capture-to-fission cross-section ratio for \(^{238}\)Pu is 0.426 ± 0.019. (auth)
1967


An experimental method for the determination of the spectral average of the capture-to-fission ratio & 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 & for materials inserted in the later chambers by an electrodeposited method, multigroup diffusion theory programs, respectively. Comparison aug-

1968

Neutron Cross Sections:

4. Capture-To-Fission Ratios


Integral values of & for 233U and 235U have been deduced from experiments in two different cores (median 233U fission energy 50 and 180 keV, respectively) in the PRO reactor. The method of measurement is the same as that used for instance at ZEBRA.

The experiment includes reactivity measurements of the sample and of a standard, &P, as well as an absolute determination of the fission rate in & Pu and 235U and the capture rate in &P. The experimental & values agree well with the calculated ones for the hard spectrum core measurements, both for 233U and 235U.

The measured value for 235U in the soft core is slightly higher than the calculated one. The discrepancy for 235U is large and hitherto unexplained. (auth)
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 $\alpha$ and $\alpha'$ for $^{238}$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 sections agree 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 important, the directly measured capture cross section is consistent over many of the resonances. This was obtained indirectly by subtracting the fission and potential scattering cross sections from the total capture cross section. The capture and fission resonance integrals and their ratio $\alpha$, obtained from our measurements, agree within the uncertainties with the direct integral measurements of these parameters. Further measurements on $^{235}$U and $^{238}$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 is presented. To compute the breeding ratio, the Doppler coefficient, and other parameters of large fast power reactors, it is important to know $\alpha$, 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 $\alpha$ for $^{238}$U, $^{235}$U, and $^{239}$Pu with monoenergetic neutrons at 30, 60, and above 175 keV. But the value of $\alpha$ for $^{235}$Pu varies by more than a factor of two between 30 and 600 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 $\alpha$ to energies where monoenergetic neutron sources are not readily available. Detailed measurements for $\alpha$ for $^{235}$Pu are now in progress, at few-keV intervals in the range of 1.0 to 10 keV and at 100-keV intervals in the range of 10 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 $^{235}$Pu are compared with those of Hopkins and Diven and with values of $\alpha$ obtained from direct measurements of $\beta$ by Spivak et al. The factors limiting the precision of the measurements are discussed in more detail. (auth)
A special liquid scintillator detector has been developed for the purpose of measuring alpha(E) for 238Pu 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 events 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-Isao 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 x 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)

Neutron Cross Sections:
4. Capture-To-Fission Ratios


A symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165 (Vol. 1).

CONF-871043 (Vol. 1).

 instrumentation for measurement of the ratio of the neutron capture cross section to the fission cross section was designed, assembled, and calibrated, and measurements were made for 235U, 238U, and 239Pu 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 x 10^-9 sec under operating conditions. The energy resolution of the 2.5-MeV sum peak of 6Li was 95%. 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)


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

V. NEUTRON CROSS SECTIONS

5. Resonances

1967

From IAEA Conference on Nuclear Data, Paris.

The results are presented of measurements on the radiative capture of neutrons in the separated silver isotopes 107Ag and 109Ag 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 107Ag and 109Ag levels within the limits of statistical accuracy obtained. The values of the strength functions S0 for 107Ag and 109Ag are found to be 0.43 x 10^-4 and 0.83 x 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)

From IAEA Conference on Nuclear Data, Paris, France.
Transmission measurements of the neutron total cross sections of Fe, 56Fe, 54Mn, and 54V at 20 to 200 keV were made using the time-of-flight method. A multilevel analysis of the data for 54Mn and 54V yielded resonance parameters for these isotopes, and an analysis of the data on Fe yielded resonance parameters for 56Fe. (D.C.W.)


Resonance integral calculations are done for 238Th infinite dilute, 238Th metal rod, and 238Th-235U rod systems. Doppler effect calculations are performed for 238ThO, 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 u- and p-wave contributions were computed by standard methods. The data describing the resolved resonance parameters up to 3 keV (1.4 x 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 1.83 x 10^-4 (eV)^-1. The calculated resonance integrals and Doppler coefficients are compared with measurements; they are found to be in excellent agreement with each other. (auth)


The intensities of resonance averaged y 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)
Neutron Cross Sections: 5. Resonances

50634  FISSION COMPONENTS IN $^{234}$U RESONANCES.

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 $\Gamma_f = 0.008$ 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 $\sigma_f$ is 97.7 b · 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 $\chi^2$ distribution with $\nu = 1.39 \pm 0.37$ degrees of freedom. Evidence for a grouping of sub-threshold fission resonances is similar to that already found in $^{238}$Pu and $^{237}$Pu and according to Wegmann and Lyon 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.

59637  (RPI-328-123, pp 1-17) NEUTRON CROSS SECTIONS. (Rensselaer Polytechnic Inst., Troy, N. Y.).
The average capture cross sections of $W$, $^{94}$W, $^{98}$W, $^{152}$W, and $^{238}$U in the energy range from 1 to 100 keV were determined. The $p$-wave strength functions for $^{56}$Fe and $^{64}$Ni were extracted from capture data, and the radiative width of the 3.6-keV resonance in $^{232}$Th was determined. The total cross section of $^{147}$Pm was determined by transmission measurements on four $^{147}$PmO$_2$ samples over the energy range from 0.008 to 200 eV; neutron resonance parameters were obtained for energies up to 220 eV.

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 yield $g_f$, for small resonances or $\Gamma_f$ 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.

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-Eckart $\mathbf{H}$-Matrix theory.

The evaluation for the cross sections in the keV region is described. The present evaluation gives a better interpretation of the $\alpha$ 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 user convenience, the resonance parameters are listed.
V. NEUTRON CROSS SECTIONS

6. Doppler Effects

1967


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/UO2 between 1010 and 1330°C was 1.25 ± 0.15, in satisfactory agreement with calculation. The Doppler effects in UO2 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 are used to identify quantitatively sources of interference not accessible to other methods. (auth)


A series of experiments has been carried out in a fast-neutron spectrum, characterized by a median fission energy of 52 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 235U isotope which has been investigated in several chemical forms and chemical compositions. 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 perturbations on the Doppler coefficient of Th have been studied by surrounding the sample with "blanks" consisting of heavy resonance absorbers, structural materials, and several types of scatterers, including Na. (auth)


The results of Doppler experiments in ZBR-3(labor) are presented. The experiments were done on two cores: one which served as a standard comparison and the other which was for participation in the international comparison. Reactivity measurements due to changes in materials and temperature are presented; Doppler effect measurement results are shown for both assemblies. (M.L.S.)


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The Doppler-coefficient has been measured in the cryogenic temperature range. The inverse temperature dependence of the coefficient results in relatively smaller 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)


Spherical samples, 2 cm in diameter, of 238U, 239U, and 239Pu were irradiated in turn at various temperatures ranging from 170 to 770 K in a central cavity of a spherically symmetrical Si-1 photo-neutron source. The 238U(n,p) reaction rate was measured by counting the 239Pu activity produced and the (n,f) reactions were measured 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 program developed by Britsch and Darston. 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)


Thesis.

The Doppler effect in 235U capture and 239U 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 235U foil thicknesses and one 239U 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 235U Doppler effect expressed as the ratio (change in fission activity with temperature/temperature fission 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 239U agreed quantitatively with those found from the "soft" neutron energy spectrum. Within the precision of the measurement to 239U Doppler effect was observed. The calculated 239U Doppler effect was smaller than the sensitivity of the experiment, thus within its precision (±0.002) the measurement confirms the theory. (Diss. Abstr.)

35665 (ANL-7320, pp 569-64) APPLICATION OF THE SEFOR CRITICAL EXPERIMENTS AT ZPR-3 TO SEFOR.


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 SEFOH 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 SEFOH reflector-control system. The Doppler coefficient was measured. The calculated 235U Doppler coefficient was in agreement with the experimental value; the measured 239Pu contribution to the SEFOH 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 pulse analysis. The values measured by the two techniques were in agreement. Fission ratios, fractional and boron traversals, and plutonium worth distributions were measured and compared with calculations. A list of 17 references is included. (auth)
6. Doppler Effects

35640 INVESTIGATION OF THE DOPPLER EFFECT IN \(^{235}\)U IN THE OUTSIDE BLANKET OF THE BR-1 REACTOR

58720 DOPPLER EFFECT IN \(^{235}\)U IN A SHIELD MADE OF URANIUM OXIDE...
Neutron Cross Sections:

6. Doppler Effects


From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-185(Vol.1); CONF-671045-(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 x 10 cm² 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 ratio 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. Theoretical analysis of the experimental data for uranium is given. (auth.


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


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


The Doppler effect in Tm²⁴¹ capture and ²³⁹U 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 Tm²⁴¹ foil thicknesses and one ²³⁹U 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 (neutron energy spectrum) 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 Tm²⁴¹ Doppler effect. The measured ²³⁹U 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 Tm²⁴¹ agreed quantitatively with those found from the "soft" neutron energy spectrum. Within the precision of the measurement no ²³⁹U Doppler effect was observed. The calculated ²³⁹U Doppler effect was smaller than the sensitivity of the experiment, thus, within its precision (0.002), the measurement confirms the theory. (auth.)


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


Neutron Cross Sections:
6. Doppler Effects


Measurements of the Doppler effect in 239U capture and 235U 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 UO2 disks, which were heated in a small oven placed in the core center. The measurements on 239U were extended to 1780°C and on 235U to 1470°C. 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 239U 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 1) a Doppler effect (relative increase in capture rate) of 0.760 ± 0.018 was obtained on heating from 343 to 1780°C, and in the hardest spectrum (core 3) the corresponding value was 0.030 ± 0.043. An appreciable Doppler effect in 235U fission was obtained only in the softest spectrum, in which the measured increase in fission rate on heating from 320 to 1470°C was 0.007 ± 0.003. (author (Sweden))


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 239U capture and the 235U fission cross sections in fast reactor spectra has been studied by irradiating heated foils or plates of the isotope 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 239U and the capture rates in 54Mn, 115In and 197Au inside and around the rods have been compared with results of multigroup calculations. (auth)
VI. LABORATORY SUMMARY REPORTS AND MISCELLANEOUS

1967


Summaries of progress are presented concerning the EBWR Pu recycle program, EBR-II operation and development, physics experiments in ZP-3 and ZP-4, 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 present development, liquid metal direct conversion generator research, AARR design and development, and research on nuclear safety. (J. R. H.)


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 ZP-3 during the last four months on Assembly 48, a large, clean, plutonium-fueled core, were compiled and a summary is presented. The AARR design was changed to a HFIR type, and all research and development work on the stainless steel cermet fuel was terminated. (auth)


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.)


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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 EBWR-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-worthy 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 235U, 238U, and manganese foils in several configurations, Relative 235U 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 these 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.)
1967


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, 6th, 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 x 15-in. central region that contains Pu with 22% 235Pu substituted for 4.5% 239Pu. 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 line flux variations across the cell. Fission ratios were measured versus 235U, 236U, 238U, 239Pu, and 240Pu. 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, 235U, 236U, 238Pu, and 239Pu. 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.M.H.)

34076 (ANL-7342) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 1967. (Argonne National Lab., Ill.), Contract W-31-109-eng-38. 139p, Dep., CFSTI.

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Laboratory Summary Reports...


... REACTIVITY — measurements in EBR-II core, effects of power level on; perturbation measurements in ZPR-9, use of small samples for central

... 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; ratio measurements in ZPR-9 of uranium 238/uranium 235; ratios for uranium and neptunium, measurements in ZPR-9 of.

... URANIUM ISOTOPES U-235 — fission ratios for, measurements in ZPR-9 of.

... NEPTUNIUM ISOTOPES Np-237 — fission ratios for, measurements in ZPR-9 of.

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neutron scattering studies, elastic neutron scattering from elements of intermediate weight, elastic neutron scattering from Na and Si, fast neutron scattering from nuclei in the mass region $A = 95-130$, the interaction of fast neutrons with the $^{182,184}$W isotopes, a search for fluctuations in the fission cross section of $^{129}$I, neutron flux measurements in the 10-200 keV region, stripping reactions, fast neutron total cross sections using a monoenergetic source and an automatic facility, fast neutron energy degradation through the $\nu$, $\nu'$/N pathway models of nuclear resonance reactions, the $^{129}$I 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 $^{241}$Cm and $^{240}$Cm, thermal reactor physics including Hi-C, adaptation (Hi-C) critical experiments, Hi-C uniform lattice calculations, initial critical experiments of the EBWR Pu recycle program, measurement of capture-to-fission ratios of $^{129}$I and $^{208}$~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, therein, neutron spectra in depleted U, calculations of Na-vold coefficients in large fast neutron carbide cores in assemblies No. 2 and 3 of ZPR-6, calculations of the effect of illu 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. 42, 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-8 assembly No. 47, standard deviation of ion chamber current measurements in ZPR-6 assembly 47, measured reactivity worths in ZPR-6 assembly No. 42, 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 burned fuel, properties of irradiated UO$_2$ plus prior to TREAT facility transients, photographic fast reactor safety experiments on irradiated oxide pin in the TREAT facility, transients with pin 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-rolled concrete cans, maximum permissible body burdens of Pu isotopes and resulting release criteria, fast reactor meltdown accident analysis code, PREAM, experimental physics techniques and facilities including a low geometry y counting chamber, absolute determination of fission rates in $^{235}$U and $^{239}$U and capture rates in $^{238}$U by radiochemical techniques, precision fission rate measurements by fissometer counting, solid-state Compton spectrometer for measurement of reactor y 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 epithermal 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 y-rays following fission fragment 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 computing methods and theory including the Argonne Reactor Computation (ARC) system, the ARC system glossary, the Multigroup Constants Code (MCC).
1967

Modification of THERMOS to generate transfer cross sections, generation of multigroup cross sections using a coupled MC3-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, quasi-static 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 fissible 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 233U in a UO2 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 Co2 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 699 references is listed throughout the report. (M.L.S.)


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 for; cooling systems for, analysis of; tube materials for, stress analysis for; core for, hot channel factor determination for; core for, radiotrace 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

U1THERNS—flux distributions in FFTF, evaluation of postulated effects of skewed THM HYCOMPUTERS—time response of ungrounded, analysis of REACTOR SAFETY—scram requirements for single channel fuel elements REACTOR MODERATORS—effects on FFTF neutron physics parameters of beryllium oxide U1RANIIUM ISOTOPES U-235 — Doppler coefficients for, effects of crystalline binding on REACTIVITY—worths of control rods for FFTF, two-dimensional transport models for calculation of


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; Pita 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 radiations chemistry; electronic structure of metals and alloys; and radiation damage in crystalline solids. (P.C.H.)
A recent revision in HRG (Hanford Revised Gamm) a neutron "slowing down code", modifies the calculation of the effective reactivity 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 single 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 % PuO2-UO2 fuel in H2O. These experiments are directed towards determining the physics properties of Pu-fueled H2O 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 confirms that PuO2-UO2 fuel in H2O. A calculation study of thermalization in PuO2-UO2 fuel in H2O 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 using a mixed lattice of Pu-Al fuel and thorium targets in alternating layers. New techniques were developed to adapt the PCTR type of measurement to this lattice array of supercells. The result includes values of $k_{eff}$ 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 total, a 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 thermal 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 full irradiation position. Neutronics calculations were performed for an 800 liter, PuO2-UO2 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 PuO2 fuel fission ratios are in reasonable agreement with experiment, whereas the computed 239Pu/239Pu 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 energy group.
1966


PRESSURE VESSELS—testing of high temperature gas cooled reactor prestressed concrete rector pressure vessel 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; cycle economics 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; design for, development of barrier; density calculations in, in chemistry studies of; irradiation of coated particle, results of.

URANIUM ISOTOPES U-238—prompt absorption as a function of fission parameters, calculation of.

Fission Products—deposition on stainless steel specimen from GAIL IV fuel element; deposition on stainless steel pipe downstream from GAIL IV fuel element; time from beryllium fuel compact of almost steady state gaseous.

FILTRATION—radiochemical analysis 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.

COATING—behavior of silicon carbide reactor fuel, effects of strongly turbulent atmosphere on; fission product release from barrier.

NEUTRON CROSS SECTIONS—evaluation of uranium 235.

PLUTONIUM—diffusion studies on.

GAMMA—radiation effects at high temperatures on; thermal conductivity of isotopic and anisotropic.


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 Doppler coefficient was in agreement with the experiment; the measured Doppler 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 (0.98) 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 ratio, fission and burnup traverses, and Pu worth distributions were measured and are compared with calculations. (auth)

1968

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 cross sections for, sensitivity for, effects on; 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, effects of absolute; neutron capture density in; design parameters for ZPR-9 assemblies 11 and 12; cores for ZPR-6 and ZPR-9, comparison of calculated and adjoint fluxes 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; measurement of; 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 fast ZPR-6, savings properties of inconel—sodium; core for ZPR-3, symmetry effects of embedded reflectors in blanket around; reactivity worth of, location of reciprocally flux; 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 12-17; effects of temperature on; measurements in ZPR-6, pulsed neutron techniques for; determinations for fast converter reactors, effects of sodium voiding in.

URANIUM OXIDES UO2—fuel pellets of, temperature effects on radial expansion of axial constrained.

URANIUM ISOTOPES U-238—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.
1968

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 fast ZPR-6; reactivity worth determination in ZPR-3
ALUMINUM — Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide fast ZPR-6; reaction worth determination in ZPR-3
PLUTONIUM ISOTOPES Pu-239 — reactivity worth determination in ZPR-3


1968

24925 (ANL-7127) REACTOR DEVELOPMENT PROGRAM

CRITICAL ASSEMBLIES—neutron spectrum in ZPR-3 Assembly 6; central, FT; neutron spectrum in ZPR-6 Assembly 8 with and without sodium, central, FE; sodium void coefficient measurements in ZPR-6 Assembly 6.

URANIUM ISOTOPES U-235—neutron fission rates in EBR-2, E/T; neutron fission cross section measurements from 30 to 1500 keV.

20118 (ANL-7139) REACTOR DEVELOPMENT PROGRAM.

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.

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.

35422 (ANL-7445) REACTOR DEVELOPMENT PROGRAM,

This monthly progress report includes information on FRP-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. (W.M.J.)

37401 (ANL-7457) REACTOR DEVELOPMENT PROGRAM,

This monthly progress report includes information on EBR-II, LMFBR, ZPR-3, ZPR-6, ZPR-8, ZPR-10, 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,

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 MWh 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. (D.R.)

50699 (ANL-7487) REACTOR DEVELOPMENT PROGRAM,

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 ZPR-3 are presented. Operations for the EBR-2 during August 1968 are described. Nuclear safety studies, including TREAT operations, and chemical separations activities are discussed. (D.C.C.)

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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 AAR, analysis of; transients in AAR, summary of reactor responses to

EXPERIMENTAL BoILING WATER REACTOR — poisoning for, reactivity worth of boric acid; control rods for, reactivity

NEUTRON CROSS SECTIONS — variations in uranium-235 and uranium-238, effects on reactivity of; determination of AARR internal thermal column effective group


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, (E)
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)
REACTOR SAFETY — coolant boiling and two-phase flow in 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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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 3.1-53) THERMAL REACTORS, (Batelle-Northwest, Richland, Wash., Pacific Northwest Lab.). Measurements to obtain adjoint weighted escape neutron production cross sections were completed for four lattices of PuO2- UO2 fuel in graphite moderator. The measurements were made in the Physical Constants Test Reactor (PCTR), and a value of the adjoint medium neutron multiplication factor (k) was obtained for a fuel. The calculated values agree with PCTR null reactivity test-measurements. The value of k was compared with the corresponding value obtained from measurements for the same lattice. Analysis of the Cross-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 PRPF-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 NTR-Phoenix core was investigated to avoid having the large flux increases outweigh the close-in flux decreases. A full-core buckled-shim critical loading was achieved in the PRPF-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 PRPR. Analysis of the burnup data from the Aluminum-Pu02 fuel irradiated in PRPR 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 H20. 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 PRPR 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 PRPF 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 239Pu and 235Pu thermal cross sections. Calculations of the photon neutron production in D2O moderated reactor systems fueled with 19-rod clusters of Al-Pu, UO2-PuO2, and UC were made. A calculational study was made comparing correction factors for reactivity obtained by various methods. The corresponding effect of resonance integrals and reactivity were also determined. Calculations were made of reactivity factors in H2O moderated loadings of UO2-2 wt % PuO2 fuel. The calculated values agree with measured values within ±5%. Calculations were performed to help predict critical masses and various lattice parameters for H2O moderated lattices containing UO2-4 wt % PuO2 fuel rods. The study was made assuming two types of fuel differing in 239Pu content. (auth)

**12086** (BNWL-472, pp 4.1-43) FAST REACTORS. (Batelle-Northwest, Richland, Wash., Pacific Northwest Lab.). A survey was performed to investigate the effect of D2O on FTR neutrons. 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 kinetic 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 ZPR-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 cause the Doppler coefficient to diverge from the 1/T dependence in qualitative agreement with the experimental results. The two-dimensional perturbations code, 3-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 alternative shield arrangements were computed 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)

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14251 (BNWL-534) REACTOR PHYSICS DEPARTMENT
(Battelle-Northwest, Richland, Wash., Pacific Northwest Lab.)
14251 (BNWL-540) 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 physical properties of plutonium-fueled reactor systems.
Measurements have been made using UO₃-2 wt. % PuO₂ rods in
H₂O moderator. Experiments were conducted at infinite spacings
covering the range of moderator to PuO₂-UO₃ volume ratios
of 0.81 to 3.7 and the plutonium contained either 6 or 24% Pu₂³⁷.
The results included critical mass, buckling, 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 PuO₂ particles in UO₃ and for obtaining
additional data on water lattices not obtainable from other fa-
cilities which are required for further evaluation of analysis
methods. The initial results of the PuO₂ 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. % PuO₂,
8.05 wt. % Pu₂³⁷). The design of the first experiment in the water
tank is complete and will contain PuO₂-UO₃ fuel in a 1 in. x
1 in. pitch. A large scale burnup experiment has been initiated
in the D₂O-moderated Plutonium Recycle Test Reactor using 19-
rod clusters of UO₂-2 wt. % PuO₂ 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 con-
ducted 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
H₂O 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 UO₃ and PuO₂-Pu₂³⁷ fuels. In addi-
tion, the fuels which contained plutonium differed in their
Pu₂³⁷ 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 ex-
periments in the PRTR are both nondestructively and destructively
analyzed to obtain fuel and fuel concentration and isotopic com-
position. From these data effective cross section ratios are
derived for use in evaluating burnup analysis methods. A survey study
was conducted to determine the approximate magnitude of various
reactivity coefficients for UO₃-PuO₂ fueled light water reactors.
Calculations were also performed to determine the reactivity char-
acteristics of thorium loadings in D₂O moderator. The feasibility of
the thorium loading in the PRTR was determined either as a batch
core or as a driver region for a UO₃-Pu₄ core. The physics
characteristics of metallic fuel and of ceramic fuels were in-
cvestigated along with the variation in rod size in thorium enrichment.
Small neutron inelastic scattering cross sections have been reported
for H₂O and D₂O. The double differential cross section and corre-
sponding Eikonal scattering law have been obtained from measure-
ments for room temperature H₂O and D₂O and for H₂O of five de-
grees below freezing point using the Battelle Rotating Crystal
time-of-flight Spectrometer. In addition, results of measure-
ments for $K_i$ at $30^\circ$ using the Battelle Triple-Axis Spectrom-
eter have been reported. An extensive evaluation of the IBM Monte Carlo Code has been completed. Results of this evaluation indicate that the IBM Monte Carlo code is free of gross program errors and can be used reliably for accurate 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 Eikonal-Skofield 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 IBM Monte Carlo code has been thoroughly tested and appears to be working satisfactorily. The ZODIAC-2 burnup code was modified and expanded to increase its burnup capabilities. The HRG (Stanford 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 calculated and measured integral parameters such as reactor multiplication and effective cross sections. Comparisons have been performed for numerous plutonium and/or uranium fueled H2O 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.


CRITICALITY STUDIES—measurement of plutonium oxide (PuO), neutron oxide (UO), effects of plutonium enrichment on, (E/T)

REACTIVITY—measurement of PRF uranium oxide (UO)–
plutonium fuel, core effects, 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) and
plutonium oxide (PuO)–uranium oxide (UO),

URANIUM OXIDES UO—fuel loading for PRF 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-Northwestern, Richland, Wash. Pacific
Northwest Lab.). July 1968. Contract AT(45-1)-1830, 104p,
Dep. CFSTI.

CRITICAL ASSAMBLIES—kinetics for plutonium oxide (PuO)–uranium oxide (UO)–burnup NPF, comparison with TTR of

PLUTONIUM NITRATES (Pu(NO3)2)—criticality of slab-shaped,
analysis of

CRITICALITY STUDIES—measurement of slab-shaped plutonium nitrate (Pu(NO3)2), description of, measurement of
rectangular-shaped polystyrene–plutonium oxide (PuO)–
lattices, description of

PLUTONIUM OXIDES (PuO)–polystyrene–PuO; criticality of
rectangular-shaped lattice on, analysis of (D.C.)
Work performed under United States-Euratom Joint Research and Development Program.

Fabrication of the 260 homogeneous fuel rods containing 2% $^{235}\text{U}$ and 2.7% $^{239}\text{Pu}$ is completed. Fabrication has started on the making of 260 heterogeneous type rods containing 2% $^{235}\text{U}$ and 2.7% $^{239}\text{Pu}$. A part of these rods will be fabricated in the self-contained unit on the pneumatic vibrator. Checking of the chlorine content of the plutonium 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 sweling 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 CH-1 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 PAN Trust code has been improved by proceeding with the treatment of the heterogeneity in the fast field, the broadening of the resonance of $^{235}\text{U}$ 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-Vulcan tests has continued. A supplementary program of sub-critical tests on 4/0 fuel with an N/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 $\text{UO}_2 - \text{PuO}_2$ area with 3.6 wt % of $\text{PuO}_2$. (J.C.W.)