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

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

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INTRODUCTION

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

I. Critical Experiments, Reasonably Homogeneous
II. Critical Experiments, Lattices
III. Reactivity Measurements
IV. Neutron Flux Spectra
V. Neutron Cross Sections
   1. Sources of data
   2. ENDF/B tapes and other evaluated lists
   3. Wide ranges in energy
   4. Capture-to-fission ratios
   5. Resonances
   6. Doppler effects
VI. Laboratory Summary Reports and Miscellaneous

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

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

Reasonably Homogeneous

1967


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%) and NO. 5 with VAP-12 and polythene (7.5%). All assemblies had a thick reflector of Cu except for No. 4, which had a U reflector. Among the experiments carried out with FRO the following items are dealt with: critical mass, reaction ratios, central reactivity coefficients, and heterogeneity effects. Measurements of reactivity, worths of multiplying as well as absorbing control rods are also summarized, including studies of interaction effects and streaming in empty channels. Recently, spectrometry work with proton-recoll counters and Doppler measurements by means of foil activation have been initiated. The latter have been carried out at 590°C, and preliminary results for 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 sections, tests of cross section sets, clean critical experiments and their analysis, evaluation and computation techniques, Doppler and Na reactivity effects, special experiments and their analysis, spectrum measurements, experimental techniques and equipment, and future programs. A total of 73 papers was presented. Abstracts were prepared for 58 of the papers; 8 papers are included under CONF-661019; abstracts for 6 papers appear in Nuclear Science Abstracts, Vol. 21, under the following abstract numbers: 19151, 1520, 1946, 1945, 19466, 5983; one paper is abstracted under the report number TID-Report-1364. (M.A.S.)

For abstracts of individual papers see: 34417, 25838 - 25844, 35497, 35546 - 35563, 35545, and 35568 - 35700.

35675 (ANL-7320, pp 205-14) ZPR-3 ASSEMBLY 48: STUDIES OF A DILUTE PLUTONIUM-FUELED ASSEMBLY. Broomfield, A. M.; Amundson, P. I.; Davey, W. G.; Gausdlo, J. M.; Hess, A. L.; Kenney, W. P.; Long, J. K. (Argonne National Lab., Ill.). A series of relatively simple Pu-fueled assemblies with well-degraded spectra has been designed for study in ZPR-3. Assembly 48, the first in the series, was chosen as the subject of an international comparison of fast reactor calculation techniques. Each assembly in the proposed program will consist of a cylindrical core surrounded by a depleted U blanket. In Assembly 48 the material constituents of the core are limited to Pu, depleted U, Na, graphite, and the stainless steel present in the structure and casing 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)
Critical Experiments: Reasonably Homogeneous

35881 ANALYSIS OF PuO$_2$-UO$_2$ CRITICAL EXPERIMENTS.


The critical masses of oil reflected, 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, 31.66, 31.1, and 51.3 kg. Critical masses were determined by reciprocal multiplication measurements on fully reflected assemblies and are compared with calculations. (auth)


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


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


38384 (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)


Further study into the critical mass of nuclear reactors is outlined. Results of the study are presented. (M 1.8)


A Criticality Data Center has been established at the Oak Ridge National Laboratory under the sponsorship of the USAEC for the purpose of collecting from sources, both in the United States and abroad, information applicable to criticality safety problems. The principal output of the Center is typified by TID-7028, "Critical Illusions of Systems Containing 239U, 238Pu, and 232Th," and TID-7045, "Nuclear Safety Guide," both of which were originally the result of group efforts not under the sponsorship of the Center. The former document summarizes most of the data available at the time of its publication and will require frequent expansion. The latter document is presently under revision to incorporate the results of most recent investigations and to reflect the development of reliable theoretical analysis. Both documents are internationally known and used. (auth)
Critical Experiments: Reasonably Homogeneous

1967


Critical Experiments:


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

8063 CRITICALITY STUDIES—critical mass and volume of reflected and unreflected paraffin-moderated uranium-tetrafluoride, (E)

NPURTONS, PROMPT—decay constants in reflected and unreflected paraffin-moderated uranium-tetrafluoride assemblies, (E)

URANIUM FLUORIDE UF₄—criticality studies on reflected and unreflected paraffin-moderated assemblies of, (E)

CRITICALITY STUDIES—critical mass and volume of reflected and unreflected paraffin-moderated uranium-tetrafluoride, (E)

URANIUM FLUORIDE UF₄—criticality studies on reflected and unreflected paraffin-moderated assemblies of, (E)

Critical Experiments:

35569 (ANL-7320, pp 270-5) USE OF INTEGRAL MEASUREMENTS AS SUPPLEMENTARY DATA IN NEUTRON CROSS-SECTION EVALUATION. Perel, S. J.; Rakavy, G.; Reiss, Y.; Yelin, Y. (B. M. U., 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.)


A series of small, Pu-plus-U-235-fueled fast reactor assemblies with stainless steel and axial reflectors were constructed in Argonne's Zero Power reactor-3 as part of the design program for the FARET reactor. These studies, designated ZPR-3 Assembly 48, were essentially mockups of possible loadings of the FARET core. The primary objective of the studies was to confirm the physics analysis of multifeulde, nonuniform core loadings as 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 interconnected. Analytical calculations were done for all experiments, and the methods for analysis that were adopted are discussed. (auth)
From International Conference on Fast Critical Experiments and Their Analysis, Argonne, Ill.

The fast core of the Mixed Spectrum Critical Assembly (MSCA) or the Vallenius Atomic Laboratory, contains a loading of enriched U, 0, and Inconel. The neutron spectrum in this assembly is spatially asymmetric and representative of a dilute fast ceramic reactor. Measurements reported include fission rates of $^{235}$U, $^{239}$Pu, $^{238}$U, and $^{237}$Np. Neutron life-time determined by pulsed neutron and 1/5 poison substitution, and reactivity values 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 20-50 liter range, containing essentially equal volume mixtures of W, fully enriched U metal and Ta are studied. Data from the first basic critical experiment include central fission ratios, reflector effects on power distribution and neutron lifetime, and relative reactivity worths. (auth)


The fast core of the Mixed Spectrum Critical Assembly contains a loading of 405 kg $^{235}$U, 1540 kg $^{239}$Pu, and 10130 kg of Inconel. The neutron spectrum in this assembly is spatially asymmetric and representative of a dilute fast ceramic reactor. Measurements reported include fission rates of $^{235}$U, $^{239}$Pu, $^{238}$U, and $^{237}$Np. Neutron life-time determined by pulsed neutron and 1/5 poison substitution, and reactivity values 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 20-50 liter range, containing essentially equal volume mixtures of W, fully enriched U metal and Ta are studied. Data from the first basic critical experiment include central fission ratios, reflector effects on power distribution and neutron lifetime, and relative reactivity worths. A list of 14 references is included. (auth)


The fast core of the Mixed Spectrum Critical Assembly (MSCA) or the Vallenius Atomic Laboratory, contains a loading of enriched U, 0, and Inconel. The neutron spectrum in this assembly is spatially asymmetric and representative of a dilute fast ceramic reactor. Measurements reported include fission rates of $^{235}$U, $^{239}$Pu, $^{238}$U, and $^{237}$Np. Neutron life-time determined by pulsed neutron and 1/5 poison substitution, and reactivity values 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 20-50 liter range, containing essentially equal volume mixtures of W, fully enriched U metal and Ta are studied. Data from the first basic critical experiment include central fission ratios, reflector effects on power distribution and neutron lifetime, and relative reactivity worths. A list of 14 references is included. (auth)


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


The fast core of the Mixed Spectrum Critical Assembly (MSCA) or the Vallenius Atomic Laboratory, contains a loading of 405 kg $^{235}$U, 1540 kg $^{239}$Pu, and 10130 kg of Inconel. The neutron spectrum in this assembly is spatially asymmetric and representative of a dilute fast ceramic reactor. Measurements reported include fission rates of $^{235}$U, $^{239}$Pu, $^{238}$U, and $^{237}$Np. Neutron life-time determined by pulsed neutron and 1/5 poison substitution, and reactivity values 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 20-50 liter range, containing essentially equal volume mixtures of W, fully enriched U metal and Ta are studied. Data from the first basic critical experiment include central fission ratios, reflector effects on power distribution and neutron lifetime, and relative reactivity worths. A list of 14 references is included. (auth)
1967


A series of one-, two-, and three-module cores containing highly-enriched UO₂ seeds and ThO₂ blankets have been studied. The purpose of this program was to compare design calculations with various measured core parameters. These parameters included the critical eigenvalues, seed power distributions, fast and thermal activation traverses, epithermal to thermal Th capture ratios, epithermal to thermal 235U fission ratios, thermal disadvantage factors, and fast advantage factors. In addition, some information on core intermodule coupling was obtained. Each module of the assemblies consisted of a narrow hexagonal annular 235U seed and an inner and outer ThO₂ blanket. The cores were designed to be nearly clean critical with no significant internal structure except fuel rod cladding in the active portion of the core. This allowed a fair test of the design model on highly absorbing narrow seed regions in a ThO₂ blanket. The design model was found to agree quite well with experimental results. The critical eigenvalues for all cores were consistent and close to unity. Near the seed-blanket interface, discrepancies between calculated and experimental traverses were noted and have been explained by spectrum-weighted cross sections and a higher order approximation to transport calculations. Monte Carlo calculations gave good agreement for experiment with thermal disadvantage factors and fast advantage factors. Fast leakage effects were found to be important in the calculation of the fast advantage factor in the seed region. (auth)


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

A theoretical technique to interpret critical data using one-dimensional codes for spheres, slabs, and infinite cylinders of 235U, 238U, and 239Pu was compared with data determined experimentally. Calculations for highly enriched 235U thermal and fast systems, even though cross sections for fast and epithermal cores were chosen, gave satisfactory results. A tendency for progressive overestimates of critical radii with decreasing enrichment was seen. A trend of increasing error with decreasing 235U ratio for systems enriched to 5% 235U or less was noted. For a given 235U enrichment, a calculational bias may be determined and applied with confidence. (F.S.)
1968

12087 (BNWL-472, pp 5.1-16) CRITICAL MASS PHYSICS, (Batelle-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 58 g/liter and axial nodalities were 2.3 and 5. The plutonium contained 4.1 wt % of UO2(NO3)2. The effect of voided walls, lattice reinforcement, and voided walls, was made. Correlation of slab experimental data and initial condition theory were in poor agreement. The clean critical bare and water reflected infinite slab was experimentally estimated to be 15.7 and 10.1 cm respectively for 50 g Pu/liter at an acidinity of 2.3. Criticality experiments were performed to provide data for nuclear safety guidance on handling, storing, and shipping of United Kingdom nuclear 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 indicated criticality at about 19 tanks, the reflected array gave 10.5 tanks for criticality. Critical bucklings and masses were measured for a range of lattice spacings of 2.1 wt % enriched U fuel tubes in light water. Criticality experiments were performed in support of the Gas Cooled Fast Breeder Reactor (GCFR) program. The experiments were designed to simulate water entry into the GCFR core and to check basic neutronic data and computational techniques. An experimental program to provide data for determining the minimum critical Pu enrichment of hydrogenous urany 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 thallium 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 hopes that it will be 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 Pu42-polytetrafluoroethylene compacts and the Remote Split-Table Machine. These experiments are a continuation of the basic research program to provide data for evaluating the effects of moderation and Pu42 on intermediate neutron spectra plutonium systems. The current series of experiments are concerned with fuel having an atomic II/IV of 2 and a Pu42 isotope content of 11.5 wt %.


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.

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


Criticality data are tabulated for: single 235U cores moderated by hydrogen and single Pu cores moderated by hydrogen. The results for highly enriched systems are categorized according to: unreflected spheres of aqueous UO2F2 and UO2(NO3)2; and spherical systems with water and polycarbonate; aqueous cylindrical systems—both reflected and unreflected; reflected and unreflected aqueous parallelepiped cores; uranium metal cores diluted with lucite and with lucite-graphite; and annular cylindrical aqueous cores. The results for low and intermediate enrichment systems are categorized as: bare aqueous cores of spherical, cylindrical, and rectilinear parallelepiped configurations, single material reflected systems; and composite reflectors. Criticality data for heterogeneously poisoned aqueous systems are tabulated. Pu fueled aqueous systems are categorized according to spherical, cylindrical, or parallelepiped. 57 references. (M.I.S.)
1968

6114  STUDIES OF CRITICAL ASSEMBLIES OF HOMO-
GENEOUS MIXTURES OF PLUTONIUM OXIDE AND POLY-
SYTRENE, Baxter, Alan M. (General Atomic Div., General
Dynamics Corp., San Diego, Calif.); Clayton, E. D.; Hansen, L. E.
From 15th Conference on Remote Systems Technology and
Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

6119  MINIMUM CRITICAL 238U ENRICHMENT FOR URANYL
NITRATE HYDROGENOUS SYSTEMS, Bierman, S. R.; Hess,
G. M. ( Battelle-Pacific Northwest Lab., Richland, Wash.).
From 15th Conference on Remote Systems Technology and
Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

21149 MINIMUM CRITICAL 238U ENRICHMENT OF HOMO-
GENEOUS, HYDROGENIC URANYL NITRATE SYSTEMS,
Bierman, S. R.; Hess, G. M. ( Battelle Memorial Inst., Richland,
1378). The data presented establish the minimum critical enrichment of
238U in homogeneous uranyl nitrate at 2.104 with a standard
deviation of 0.310 wt %, This results in a lower limit of 2.07 wt %
at the 99% confidence level. Optimum neutron moderation for 2.14,
2.26, and 3.01 wt % enriched uranyl nitrate homogeneous systems
occurs at H/U ratios of 0.9 ± 1.0, 9.3 ± 0.5, and 10.5, respectively.
The minimum critical enrichment is the enrichment required to
obtain an infinite neutron multiplication factor unity under condi-
tions of optimum moderation. (M.C.G.)

33062 CRITICAL EXPERIMENTS WITH HOMOGENEOUS
PuO2- POLYSTYRENE AT 5 H/Pu, Bierman, S. R.; Hansen,
L. E.; Lloyd, R. C.; Clayton, E. D. (Brookhaven National Lab.,
From 14th Annual Meeting of the American Nuclear Society,
Toronto. See CONF-680601.

37530 (BNWL-SA-1630) CRITICAL EXPERIMENTS WITH
HOMOGENEOUS PuO2-POLYSTYRENE AT 5 H/Pu, Bierman,
S. R.; Hansen, L. E.; Lloyd, R. C.; Clayton, E. D. (Battelle-
Northwest, Richland, Wash. Pacific Northwest Lab.). May 21,
Dep. CFSI.
From 14th Annual Meeting of the American Nuclear Society,
Toronto, Ontario.
Criticality parameters for mixtures of fuel having an 11.5 wt %
239Pu isotopic 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.)

Critical Experiments:
Reasonably Homogeneous

35655 COMPARISON OF MEASUREMENTS IN SNEAK-1 AND
ZPB 31-41, Boehme, R.; Barleson, L.; Boehnel, K.; (and others)
(Kernforschungszentrum, Karlsruhe, Ger.). pp 55-77 of Past
From Symposium on Fast Reactor Physics and Related Safety
Problems, Karlsruhe, Germany. See STI/PUB-165(Vol2);
CONF-671043-(Vol2).
The experimental program of the Karlsruhe Fast Zeru Power
Reactor SNEAK started in the autumn, 1966 with measurements on
a 440-lw uranum assembly, a mock-up of ZPB 31-41. Dur-
ing a four-month period the experimental installations and tech-
niques of SNEAK were successfully tested. The installations
include a movable drawer connected to an automatic sample
changer operating in a horizontal experimental channel, a ver-
tical drive unit, a pile oscillator, and a pulsed neutron generator.
The techniques used included spectra measurements with proton
recoil counters and full activation, and several techniques for
determining reactor power and 1/1, e.g., Rossou- and pulsed neu-
tron source measurements. In the experimental program quanti-
ties such as critical mass, reaction rate ratios, neutron spectrum,
material worths in the center, radial and axial traverses, 1/1, and
reactor power were determined The results were generally in
good agreement with those of the ZPB III experiments, and the
remaining discrepancies are discussed. These are partly due to
small deviations in the material composition of the two assem-
bles. The experimental data are also compared with calculations
using the 16-group YOM, 26-group KFK, and 26-group ANL cross-
section sets. While critical mass is best calculated with YOM, the
spectrum is too hard, and both KFK and ANL give better
agreement with the experimental spectrum. (auth)

25078 (CEA-R-3267) EXPERIENCES DE CRITICITE REALI-
SEES AVEC UNE SOLUTION HOMOGENE DE PLUTONIUM,
RESULTATS EXPERIMENTAUX, INTERPRETATIONS THORIQUE,
Critical Experiments Carried Out with a Homogeneous Plutonium
Solution, Experimental Results, Theoretical Interpretations,
Bouly, Jean Claude; Caizergues, Robert; Deligat, Edouard; Houelle,
Michel; Lecorche, Pierre (Commissariat a l'Energie Atomique,
(In French). Dep.
Results of a series of experimental and theoretical criticality-
study on plutonium are given. A comparison of theoretical and
experimental values for critical heights of solutions is made; ef-
teffects of nitrogen, introduced in the form of the nitrate ion, on the
reactivity of the fluid media are evaluated; the effects of 239Pu on
the reactivity of the media are analyzed. Influence of moder-
ators which are introduced into the solution is investigated; ef-
tects of dimensions of the inner cavity of annular cylinders are
analyzed. (auth)

40735 (NP-17606) CRITICALITY OF THE LIQUID MIX-
TURES OF HIGHLY-ENRICHED UF4 AND HF, Caizergues,
Robert; Deligat, Edouard; Lecorche, Pierre; Maubert, Louis;
Revol, Henri (Commissariat a l'Energie Atomique, Saclay
(France), Centre d'Etudes Nucleaires). Apr. 1968. 46p.
(In French), (R-68.1). Dep.
Critical mass is determined for a UF4-HF mixture as a function of
239U concentration; the liquid-vapor equilibrium is established for
the system. Schematics of the UF4-HF circuit are shown.
Experimental apparatus is described. Density of the binary system
is tabulated as a function of temperature; critical uranium con-
centration is shown as a function of temperature. Variation of the
effective multiplication coefficient is shown as a function
of sphere diameter. Effects of the wall of the sphere and tem-
perature effects on the reactivity of the mixture are determined.
(auth references. (M.L.B.)
1968


A description of an Al-235U-fueled critical assembly is presented. The critical assembly used 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 change in cross section is evaluated; results are tabulated for each core volume. Effects of cross section variation on initial conversion factors is discussed. (M.L.S.)


An attempt to achieve a near-critical assembly with a minimum average 235U enrichment of an unreflected, uranium metal, 1.1-in.-diameter cylinder has been completed. Data were required for the design of a low-enrichment, uranium metal, reflector system. Utilizing only the materials on hand, 2.1-in.-diameter plates of 93.3% enriched uranium and normal uranium, four low-enrichment cylinders were investigated. The thickness of the normal uranium plates dictated the exact enrichments attainable by interlayering the plates in a cyclic manner along the axis. The critical parameters were obtained by extrapolation of inverse multiplication curves, which extend to 93 to 95% of the critical height. Corrections for the reflecting properties of the vertical support structure and the building itself were applied. The significant results are tabulated. A least-squares analysis (quadratic) of the data, inverse critical height squared vs percent enrichment, yields results which extrapolate to an infinite-height enrichment of 10.5 ± 0.2%. (auth)


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


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

Critical Experiments: Reasonably Homogeneous


Presented in a handbook intended for specialists concerned with the problems of ensuring 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 ensuring nuclear safety, a review of cases of the occurrence of uncontrolled chain reactions, and the basic standards for nuclear safety are included. (ATI)


The critical dimensions of homogenous spheres containing 235U, 239U, and carbon at various C/235U 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-reflected, spherical and hemispherical enriched uranium assemblies having inside radii from 0.0 to 12.0 cm. (auth)


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.
1968


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

Criticality measurements for a water-reflected Pu sphere are presented. The high purity plutonium was fabricated and machined into a mass of 5546 g. The sphere was coated with Cu and enclosed in a 32 cm x 32 cm thick, 21 cm-wide beryllium reflector to 12 dollars for a 26.2 cm-wide beryllium reflector. The effect of composite reflectors of steel and water on the reactivity of single cylinders of aqueous solution of low-enriched UO\(_2\) is described. The results are applicable to evaluation of criticality safety of shipping containers and for verification of calculational models. (D.C.C.)


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


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


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

Critical Experiments:

Reasonably Homogeneous


A cubic core of enriched (93.15%) \(^{235}\)U foil in a cubical reflector was used to establish minimum critical mass measurements. Three core sizes, of bases approximately 6 in., 5.6 in., and 6 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. See CONF-680601.


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

The effect of composite reflectors of steel and water on the reactivity of singel cylinders of aqueous solution of low-enriched UO\(_2\) is described. The results are applicable to evaluation of criticality safety of shipping containers and for verification of calculational models. (D.C.C.)


A series of static critical experiments has been performed on an accurate mock-up of the SORA Reactor. SORA is a NaK-cooled repetitively pulsed fast reactor which will be used as a high intensity neutron source for time-of-flight experiments. The reactivity of this reactor is varied by a movable reflector. Those parameters which are related to the kinetics of the reactor have been investigated thoroughly in the critical experiments. They have been measured for beryllium and for iron reflectors of several sizes and for various core and fixed reflector configurations. The total reactivity of the movable reflectors varied from 3.7 dollars for a 11-cm-wide iron reflector to 12 dollars for a 20.3 cm-wide beryllium reflector. The reactivity of the movable reflector as a function of its position has been shown to have a parabolic dependence on position characterized by the parameter \(\alpha\), which varied from 4 to 9.9 cents/cm\(^2\). The prompt neutron time decay is described by a fast decay constant which varied between 0.39 and 0.55 \(\mu\sec\)\(^{-1}\) and a slow decay constant which varied between 0.06 and 0.10 \(\mu\sec\)\(^{-1}\). The critical masses for the various experiments was between 50.3 to 57.3 kilograms of uranium enriched to 5.2 wt % \(^{235}\)U. Using space independent neutron kinetics with one delayed neutron group, it has been shown that with a 24-cm-high, 7-cm-thick, 21-cm-wide beryllium reflector the assembly will produce pulses approximately 50 \(\mu\)sec wide at half maximum power with a peak-to-minimum power ratio of approximately 4000. (auth)

10
Breeder Reactors, London. See CONF-660502, 91 of Fast Breeder Reactors. Evans, P. V. (ed.). Oxford, Per-
A computational analysis was made for the large number of realistic critical experiments with heterogeneous mixtures. The calculations were made using both multigroup $S_2$ and diffusion theory with 18 energy groups obtained with the GAMTEC-II code. Resonance capture by the isotope $^{241}$Pu was treated in the NR and NISA approximations. The results are given as a parametric survey for $\rho$ densities ranging from 0.015 to 1.0 g/cm$^3$. The calculated minimum critical mass of $^{239}$Pu is 547 g for water-reflected aqueous $Pu(NO_3)_4$ solutions and 521 g for similar mixtures of $^{239}$Pu and water. 14 references, (auth)
Critical Experiments:
Reasonably Homogeneous
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 0.611 inch and moderated by H₂O. The water was poisoned with H₂116O (about 1.5 ppm H₂116) to obtain critical assemblies approximately 5 feet in diameter. The reactivity worth of Ag-In-Cd control pins arranged in configurations typical of power reactors was measured in these cores. The results of measurements of ρₚ, the modified conversion ratio, and the epithermal neutron spectrums 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 ²³²Th, and lumped fission products. Evaluation work was done on various reactor concepts being considered. (M.C.C.)


The performance of the lattice code WIMS was studied by the analysis of graphite moderated experimental 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 ²³⁹Pu η value of 2.098 at (2200 m/s). The use of the IAEA recommended η of 2.114 in the work showed a significant dependence of reactivity on ²³⁹Pu enrichment, which was largely removed by using an η value of 2.098. (UK)

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

The METHUSelah-I and CAROL codes for ThO₂–UO₂ or ThO₂–ThO₂ lattices with heavy or light water as moderator were assessed. Experimental bucklings were used to calculate the Keff 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 93% 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 keff's with a systematic dependence on the core radius. A satisfactory agreement is found for large-size cores. It is shown that simple calculational improvements, such as a four-group evaluation of the nuclear constants, and a more detailed treatment of core-radial reflector interface zone, lead to a homogeneously good agreement over the whole range of core dimensions. (auth)

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Critical Experiments:

Lattices

\[ k = \text{MEASUREMENT OF A NATURAL URANIUM GRAPHITE LATTICE IN RB-1 BY THE NULL REACTIVITY METHOD.} \]


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

12244 MATERIAL BUCKLINGS FOR 1.002, 1.25, and 1.95 WT PERCENT \(^{235}\)U ENRICHED URANIUM TUBES IN LIGHT WATER.


Material bucklings and extrapolation distances were measured for several slightly enriched U-metal tube lattices and tube-in-tube assembly lattices in light water. The tubes measured were: 1.002 wt \(^{235}\)U enriched (2.34-in. OD; 1.79-in. ID); 1.25 wt \(^{235}\)U enriched (2.37-in. OD; 1.80-in. ID); and 1.95 wt \(^{235}\)U enriched U (2.28-in. OD; 1.41-in. ID). The tube-in-tube assemblies measured were: 0.020 wt \(^{235}\)U outer tubes (2.34-in. OD; 1.79-in. ID) containing 1.002 wt \(^{235}\)U inner tubes (1.18-in. OD; 0.49-in. ID); and 1.25 wt \(^{235}\)U outer tubes (2.37-in. OD; 1.80-in. ID) containing 0.05 wt \(^{235}\)U inner tubes (1.18-in. OD; 0.18-in. ID). Maximum bucklings for the tubes found to be 25.00, 47.00, and 83.00 m⁻², respectively; and for the tube-in-tube assemblies, 23.50 and 38.50 m⁻², respectively. Based on the measurements, critical parameters for use in nuclear safety analyses were calculated. (auth)

13049 EXPERIMENTAL BUCKLING MEASUREMENTS WITH 2.1 WT PERCENT ENRICHED URANIUM TUBES IN LIGHT WATER.


14091 \[ k = \text{MEASUREMENT OF A NATURAL URANIUM GRAPHITE LATTICE IN RB-1 BY THE NULL REACTIVITY METHOD.} \]


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

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The principal factors influencing the accuracy of comparisons between theory and experiment for regular water-moderated lattices are examined. By the use of more elaborate theoretical methods, the accuracy of the physics methods used in the WIMS lattice code is established with regard to leakage, fast fission events in $^{235}U$, resonance capture in $^{238}U$ and thermal distribution factors for $^{235}U$. Combining this with the complete nature of the flux to buckling inferred from measurements in small exponential cores is examined and the validity of a new-dimensional analysis of this effect is questioned. Subject to this limitation it is shown that the computational methods are sufficiently accurate to allow deductions to be made concerning the gross characteristics of fundamental nuclear data through comparison of the predicted reactivities and reaction rates with experiment.

Results for a selection of $^{238}U$ lattices using both critical and exponential techniques are given and further comparisons are made for a selection of experiments using both $^{235}U$ and U metal fuel. The results of reducing the leakage by multiplying with $\Pi$ in these experiments are also described. The results obtained suggest 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-dependent error in the fast fission factor, and a discrepancy in temperature coefficient are identified, but efforts to isolate the cause of these errors were not successful.

The NASA Zero Power Reactor II (ZPR-II) has been used to determine experimentally several cylindrical configurations of aqueous fuel solutions that contain heterogeneous arrays of voids. These voids are cylindrical, are symmetrically arranged parallel to the axis of the reactor, and extend the height of the core. The study covered a wide range of highly enriched aqueous $UO_2$ 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, presented on the thermal neutron flux distributions in the central radial plane, and on the variation of void reactivity importance with radial position. These data explain qualitatively some of the reactivity effects associated with the different void spacings. (STAR)

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 Montecacciolino 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 method. 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 $\pm 2\%$. 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. (Ir-auth)


Experiments have been performed in simulated CANDU-BLW lattices in ZED-2 (square arrays of 26-rood $UO_2$ clustets at a spacing of 27.94 cm) to determine (a) the material buckling of the lattice with $H_2O$ or air as “coolants,” and the flux perturbation and reactivity effects of removing the $H_2O$ coolant from 50% of the fuel assemblies in three geometric arrangements. The buckling for the $H_2O$-cooled lattice was 1.168 ± 0.018 m$^{-1}$, and for the air-cooled lattice 3.949 ± 0.039 m$^{-1}$. The loss of coolant experiments indicated a significantly smaller increase in reactivity when alternate fuel assemblies or alternate rows of fuel assemblies were voided than when one-half of the lattice about a dia was voided. (auth)
1967


Experiments have been performed in the ZEEP and ZED-2 critical facilities at Chalk River to obtain information about the neutron spectrum in lattices of natural U fuel in heavy water moderators. Studies have been made in lattices containing clusters of U, UO₂, and UC having D₂O, air and organic liquids as coolants, all at room temperature, and some experiments with UO₂ fuel have investigated the effect of changing the coolant temperature. In general, the neutron spectrum measurements that were made in both the fuel and moderator regions of lattice cells by activating foils containing In, Mn, and Lu are described. The relative ⁶⁵In/²⁸³Mn and ⁷⁷⁷Lu/²⁸³⁷⁷Mn activity ratios have been compared with calculated ratios obtained from HAMMER, a multigroup, multiregion integral transport theory code, SOLO, a multigroup, multiregion diffusion theory code and MULTIGRO, a multigroup, two-region diffusion theory code. The activity ratios have also been interpreted in terms of the Westcott epithermal index, r, and effective neutron temperature, Tₑ. Experiment and theory are compared for selected lattices to illustrate the effect upon the neutron spectrum of fuel material, cluster geometry, coolant composition, and coolant temperature. (auth)


The intra-cell structure of the fast neutron flux was measured in several TRX lattices with ²³⁵U (mass fraction) and Al (mass fraction). 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 ²³⁵U 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 MOCAZA 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 MOCAZA. (auth)


Neutron flux distribution and backscattering 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 R₀ and B₀ 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:

Lattices


A series of experiments were completed to determine the critical parameters of lattices of IFHE fuel elements, primarily in geometries and environments of interest for transport, storage, and chemical dissolution. Arrays of these elements were made 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 U₂O₅/l and the other was 8.02 g/l. In some of the slab lattices in water, sheets of cadmium were placed between rows to serve as a neutron absorber as they might in a shipping container. (auth)


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

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

Experiments have been performed in the ZED-2 critical facility to determine the volume 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 Tₑ, and compared with values predicted by the Chalk River lattice code LATTRE. (auth)


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 less than that for the plutonium metal—water system or homogeneous solution system, whichever is smaller. The activity 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 2.925 enriched UO₂, clad in 0.8 mm thick Al tube. The theoretically calculated values for thermal, epithermal and fast neutron flux distributions, as well as the effective multiplication constant λ of the lattice, were compared with experimental data. After detailed analysis of the problems encountered in the course of the study the value of 0.9955 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%, 10%, and 3%, respectively, in the reflector region adjoining the core. The fact that these discrepancies cannot be removed by multigroup P₃ calculations would point toward insufficiency of the diffusion or P₃ 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: 2² (2 x 2 x 2) arrays bare, reflected and moderated; 2 x 2 arrays of 6-kg pairs bare and moderated; 3² arrays bare and reflected; and 3² 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)

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**Preliminary Calculations:**


Physics calculations, preliminary to experiments, for determining cell-averaged lattice constants are presented. Calculations were made for 7.65 wt % and 23.5 wt % PuO₂ 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 ²³⁵U cores. The calculated thermal neutron energy spectrum is shown. (M.L.S.)


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


A series of critical experiments using mixed-oxide (PuO₂-UO₂) plutonium fuels was carried out at the Westinghouse Reactor Evaluation Center (WREC). Two plutonium fuels with a variation in the ²³⁵U isotopic content and one low enrichment uranium fuel were used in an experimental program which included buckling, reactivity, and power distribution measurements. Buckling measurements were made in five clean lattices with the ²³⁵U fuel and two clean lattices with the ²³⁵U fuel and two clean lattices with the ²³⁵U fuel. With the ²³⁵U fuel, buckling measurements were made in two lattices at two different boron concentrations. The reactivity worths of voids, water holes, and control rods in different test arrays were determined in single and multi-region cores. Water hole and water slot power peaking effects were measured in clean and borated cores. Power distribution measurements were made in cores containing concentric regions of the different fuels, multi-region slab cores, and in cores containing interspersed fuels. (auth)
30245


A study has been made to determine whether latitude 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 UO₂ and U metal with D₂O, He and organic (IB=40) coolants, in 18 and 22 cm triangular pitch lattices. A two-group heterogeneous reactor calculation program (MICRETÉ) 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 IB=40 cooled rods. Measurements were made of the neutron flux distribution, Westcott spectrum parameters k and T, initial conversion ratio, and fast fission ratio using seven test rods and in general the results are in agreement with those made in lattices containing a large number of rods. (auth)

15791


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 side reflected systems. From the measured decay constants, k, the infinite reproduction constant, and B₄, the material buckling, were evaluated. Theoretical studies were made of the multigroup, multiregion subcritical and the experimental decay constants for both bare and reflected systems agreed well with a two-region, two-group model. From the combined pulsed and static experiments, the dispersion law for multiplying media was derived. (auth)

14041

(ASE-254) BUCKLING MEASUREMENTS UP TO 250 BUCKLINGS ON LATTICES OF AGESTA SYSTEMS AND ON D₂O ALArNAI IN THE FRANKLIN EXPONENTIAL ASSEMBLY Tz. Persson, R.; Anderson, A. J. W.; Wikahl, C.-E. (Aktiebolaget Atomenergi, Stockholm (Sweden)), Nov. 1966, 58p. Dep. man. buckling determinations by means of flux mapping were performed in Tz up to 250 on two lattices of Agesta fuel assemblies in D₂O and on D₂O alone. Most of the flux measurements were made with fission counters in pressure thimble. The perturbation 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 (AgInClO₄) 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 were about -2.4 and -1.8 m⁻², respectively, for the two lattices. During part of the experimental period about 3% unexplained excess absorption occurred in the heavy water. (auth)

38666


An interpretation is given to the various kinds of buckling measurements performed in heavy water moderated exponential and critical facilities. Problems connected with boundary transients, anisotropy, and heterogeneity are discussed. The experiments 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: 15766; NSA 16: 29993; NSA 18: 33053; NSA 19: 17206; NSA 20: 40377; NSA 21: 5677; NSA 21: 14061; and Preprint No. V, which is going to be published in a revised form in Nukleonik and will be abstracted when it appears to public. (Sweden)

8101


19457


Material neutron bucklings and thermal flux activation rates were measured for four lattices containing 233U 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. MTHUSELAH overestimates the buckling with a corresponding overestimate in κᵣ. There is a need for improvement in the reactor calculations, although the source of error is not obvious. (auth)

38659


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 to determine cadmium ratios is described. An appendix summarizes some of the 233U measurements. Measurements of the thermal disadvantage factor δᵣ are accomplished by irradiating small Dy foils in a fuel rod and in water. This technique is explained in an appendix. A fission catcher technique is used to measure the fast-to-thermal fission ratio, δF. Two techniques for measuring material buckling in exponential assemblies are described. A bibliography of 100 references is included. (J.C.W.)
1967


Calculated values of the extrapolation distance for water-reflected \(^{235}\)U-\(^{239}\)Pu-Zr ternary systems are presented. This extrapolation distance, together with previously published critical buckling data, permits the determination of critical dimensions for all possible compositions of this system. Limited data were available for the extrapolation distance for the \(^{235}\)U-\(^{239}\)Pu binary system, and no data existed for the ternary system. A quantitative determination of the extrapolation distance was achieved utilizing, in a unique manner, nuclear codes developed for reactor design purposes. Accuracy of the results was confirmed at compositions for which experimental data are available. The extrapolation distance was found to be essentially independent of the shape of the system but strongly dependent upon composition. A single diagram that presents critical buckling and extrapolation distance as a function of composition was developed. With this diagram it is possible to determine critical dimensions for a given shape and composition and optimum conditions for criticality. As an important practical example, the minimum critical limits for optimally water-moderated cylindrical arrays of \(^{235}\)U-Zr fuel elements are presented as a function of fuel-element length and composition. (auth)

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

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 \(^{235}\)U-water reactors gave a satisfactory agreement with experimental values. (tr-auth)

Critical Experiments:

Lattices


Reactor physics parameters were measured in six heavy-water lattices which were miniature versions of lattices investigated extensively in the exponential assembly at M.I.T. The lattices consisted of 0.25-inch-diameter rods in two \(^{235}\)U concentrations, 1.143 and 1.027%, and three sparcings, 1.55, 1.75, and 2.50 in. The following quantities were measured in each lattice: the ratio of equilibration to subcadmium capture rates in \(^{235}\)U; the ratio of equilibration to subcadmium capture rates in \(^{235}\)U (\(^{197}\)Au); the ratio of the total capture rate in \(^{235}\)U to the total fission rate in \(^{235}\)U (\(^{197}\)Au); the \(^{235}\)U to \(^{233}\)U fission ratio (\(^{197}\)Au); the intracellar distribution of the activity of bare and cadmium-covered gold foils; and the axial and radial activity distributions of bare and cadmium-covered gold foils. Corrections derived from theory had to be applied to account for the presence of source neutrons and boundary effects. The age-diffusion model developed by Peak was improved and corrections were obtained to extrapolate the miniature lattice data to exponential, critical, and infinite assemblies. To test the validity of the extrapolation methods, the results obtained by extrapolating the miniature lattice data to exponential assemblies were compared with the results of measurements made in the exponential assembly at M.I.T. The extrapolated and measured results agreed generally within the experimental error. It is shown that to extrapolate the values of \(^{197}\)Au, \(^{197}\)Au, and \(^{235}\)U 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 \(^{197}\)Au, \(^{197}\)Au, and \(^{235}\)U 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 subcritical units are described. A neutron nonleakage fraction parameter is defined and leads to a correlation confirmed to within 5% of the number of units by comparison with experimental data for three dimensional cuboidal arrays. A density analogue representation of the arrays is readily derivable and is shown to approximate the results from the above method, but is less precise. Factors by which the number of units in an unreflected critical array is reduced by adding a paraffin reflector are found to range from about six to greater than 30 depending on the material and on the average uranium density considered. The methods are supported by Monte Carlo calculations demonstrated to be reliable by comparison with the results of critical experiments.


Work Performed under United States–European 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. k-eff and dwell time for the first experimental lattice were 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 fuelled with natural uranium rods of 1.2 in. diameter and moderated by water (V_e/V_e = 1.43). The experimental procedure is described and a simple two-group analysis is developed for interpreting the measurements. The thermal neutron fluxes were low, being of the order of 25 neutrons/cm² s; nevertheless the buckling was determined as (9.68 ± 0.42) 10⁻³ cm², a value in good agreement with that obtained from conventional exponential experiments. (auth)

Critical Experiments:

Lattices


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

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


From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165(Vol.2); CONF-671043-(Vol.2). The design of a steam-cooled fast reactor with satisfactory safety and control characteristics and an acceptable breeding performance requires an accurate knowledge of the neutron balance and its variation with coolant density. When an assessment of such a system was undertaken the basic nuclear data available for use in the lattice calculation was of insufficient accuracy and little integral information on lattices of the required composition was available. A program of experimental measurements of the neutron balance and its variation with coolant density in lattices of this type was therefore undertaken. Measurements of important neutron reaction rates and of k-infinity were carried out in the reactor DIMELE at A.E.E., Winfrith, in a small central fast reactor zone critical by a surrounding thermal reactor zone. This system produced at its center the correct fast spectrum while requiring only 50 kg of Pu in the central fast test zone. Three lattices were studied in which appropriate polypropylene plates were inserted in a regular array to simulate the flooded condition, and the operating condition (0.1 g/cm² equivalent steam density); the welded condition was fituated by removing the planks. pp 356-41 VANATION OF k-infinity WITH COOLANT DENSITY FOR A PLUTONIUM-FUELLED STEAM-COOLED FAST REACTOR LATTICE. COMPARISON OF EXPERIMENT WITH PREDICTION, Arnold, M. J.; Fox, W. N.; Georgia, C. F.; Richmond, R. (United Kingdom Atomic Energy Authority, Winfrith, Eng.). pp 429-50 of Fast Reactor Physics. Vol. II. Vienna, International Atomic Energy Agency, 1968.

35462 Critical Experiments: Lattices


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


CRITICALITY STUDIES—measurements of critical number of aluminum-plutonium fuel rods in water moderator, effect of lattice pitch and boron concentration on; microscopic lattice parameter measurements on water-moderated plutonium oxide (PuO₂) —uranium oxide (UC) rods; measurements of boron worth and concentration for cold and xenon-free PINE Batch Core Core.

NEUTRONS—capture-to-fission ratio measurements in plutonium-239; multiplication factor determination in water-moderated plutonium oxide (PuO₂) —uranium oxide (UC) lattices; resonance escape probability determination for uranium-238.


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


As part of the study of SGHW lattices, a wide range of uniform cluster arrays was studied. Both enriched UO₂ and PuO₂/UO₂ fuels were used, and the range included pin diameters from 0.3 in. to 0.5 in. in clusters which contained from 37 to 90 pins each.

Measurements of material buckling, detailed reaction rates and void coefficient are compared with theoretical predictions using METHUSEL AH II, which is an improved version of the five-group diffusion theory code, METHUSELAH I, originally developed for SGHW assessment and de-sign studies, and the 6h-group transport theory code, WIMP, which has superseded THUUL (auth).


Measurements were made in the DIMPLE reactor on a number of regular, 3% enriched, UO₂-light water lattices. Detailed reaction rate measurements were made in addition to the material buckling, using moderator to fuel volume ratios from 3.16 to 0.76.

One assembly was heated to 190°C; in addition, the coolant density change on heating to 250°C was simulated by inserting aluminum void pins into the lattice. (UK)


The measurement of the infinite multiplication factor of natural uranium-graphite lattices in the critical assembly RB-1, Bologna, by the null reactivity method, is described. The procedure which was set up for the execution of the measurements and for their interpretation is given in detail. The error on k – 1 was estimated to be of the order of 2 to 3%, in good experimental conditions. The lattices had been previously tested in the critical assembly Maruss, 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 ²³⁵U solution apply to criticality safety considerations in handling solutions at a concentration of ∩ 330 g ²³⁵U/liter and are useful in checking computational methods. The measurements were made with 17.3 kg U²³⁵ in both reflected and unreflected arrays. Critical numbers of bottles were determined as a function of spacing, and the effect of adding moderating material between the bottles comprising an array was also examined. Monte Carlo calculations were found to reproduce the experimental data reasonably well, with k eff being computed to within about 0.03 of unity for those cases compared. (auth)


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


Resactor physics parameters were measured in three heavy water lattices consisting of 0.250-in. diam, 1.03 wt % 235U 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 epicaadium to subcadmium radiative captures in $^{238}$U; the ratio of epicadmium to subcadmium fissions in $^{238}$U; the ratio of radiative captures in $^{238}$U to fissions in $^{238}$U; and the fissions in $^{238}$U to fissions in $^{239}$Pu. These experimental results were used to calculate the following reactor physics parameters for each lattice: the resonance escape probability, the fast fission factor, the multiplication factor for an infinite system and the initial conversion ratio. Analytical results obtained by using THERMOS and GAN-T are in fair agreement with the experimental results. 11 references.


Critical Experiments:


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

Results of physics measurements carried out on subcritical assemblies using UO2 with a 5% $^{235}$U enrichment in the THETIS reactor are given. The lattices studied were of the square pitch type and the H2O/UO2 ratio had successively the following values: 1.51, 2.63, 4.13, 6.33, and 9.35. The maximum number of fuel elements available was 500, these having a useful length of 376.5 mm and a useful diameter of 7.6 mm. The cladding is steel (304) with a thickness of 0.2 mm. The following measuring techniques were used: the Inverse Multiplication Technique (IMT), the Source Ejection Technique (SET) and the Pulsed Neutrons Technique (PNT).  (D.C.C.)


From 15th Conference on Remote Systems Technology and Development Program.


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 Atomics, Sao Paulo (Brazil)). Apr. 1967. 30p. Dep.  (auth)

Buckling measurements for cores of uranium oxide (3.00% enriched in $^{235}$U) 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 70.50% and a lattice pitch of 6.544 cm, the material buckling with void was obtained by three regions, one group theory. All the results were found to agree satisfactorily with the results from critical experiments.  (auth)


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

Lattices

1968


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

1968


A derivative is given of a series of experiments carried out in support of Swiss reactor assessment studies on heavy water moderated lattices containing natural UO$_2$ cluster fuel elements. The experiments, which involved measurements of material buckling and diffusion coefficients, were designed to give a comparison of the results of measurements on single-zone lattices in the Swiss subcritical assembly MINOU with those given by substitution measurements in the Swedish reactor RO. A comparison of the two sets of experimental data confirmed that accurate results can be obtained by the substitution method using very small numbers of fuel clusters, but showed that, in some cases, the method may be significantly in error, and emphasized the need to investigate the conditions in which these errors may occur. The results of the buckling measurements were compared with the predictions of the UKAEA assessment code MUSELIAI 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 Neerolst. (auth)


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


Critical Experiments:

1968

1865 (JPRS-42222, pp 102-177) CRITICALITY OF SYSTEMS OF INTERACTING SUBCRITICAL ASSEMBLIES OF FISSIONABLE MATERIALS. Translated from pp 169-201 of Kriticheskie Parametery Sistem Dlya Reaktorochnogo Vsesoyuznogo Soslovozpadnost'. 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 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 values calculated by the PINOCCHIO code; the agreement found is satisfactory in all the cases investigated. (auth)

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

Critical experiments with fully enriched (93.2%) U$^{235}$ 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 76.2-cm-diameter core tank. The critical heights of the voided reactors ranged from 21 to about 84 cm. The calculational method consists of first computing axial leakage rates from axially finite cylindrical cells that contain the void tube and a proportional amount of fuel solution. The cell dimensions and fuel are obtained from the corresponding critical reactors. Two-dimensional $(r-z)$ $S_2$ transport calculations with five energy groups of finite height cylindrical cells are used. The axial leakage rates per source neutron, obtained from the cell calculations, are incorporated into one-dimensional radially finite reactor calculations by defining an axial leakage cross section for each energy group to result for axial neutron streaming out of the voided region of the reactor. Important reactor factors, which are also obtained from the two-dimensional cells, are used. The more concentrated fuel solutions the calculational method is satisfactory for the reactors examined and is readily adapted for use with other reactor configurations provided that two-dimensional $(r-z)$ cells can be defined appropriately. (auth)
Critical Experiments:

Lattices


Lattice parameters were measured for heavy-water-moderated lattices of 19-element clusters of ThO$_2$ containing 1.5 wt % enriched UO$_2$ (93 at. % $^{235}$U). Buckling values were determined from critical determination measurements in the ZED-2 reactor using MCNP, a two-group heterogeneous reactor code. Neutron density distributions and Westcott spectrum parameters $r$ and $T$ were obtained from measured $\text{Mn}$,$\text{Hf}$, and $\text{Th}$ activities in fuel detectors. Measurements were made for $1/3$, $2/3$, and $11/2$ cm rectangular, triangular, and lattice pitches. (auth)


Progress is described on an investigation into the adequacy of the four-factor two-group description of the neutron cycle, by comparison with experimental measurements. Maximum use is made of experimental information, but some theory and basic nuclear data are involved. The calculated $k_{eff}$ values show systematic trends with pitch and type of fuel cluster, and with coolant. Experimental results for thermal reactor fuel are also discussed. In addition, lattice parameter measurements on D$_2$O-moderated lattices of 19-element rods of ThO$_2$ containing 1.5 wt % $^{235}$U are reported. The measurements were made for four coolants over a range of lattice pitches. (auth)

44882 (AECI-2799) LATTICE MEASUREMENTS WITH 19-ELEMENT RODS OF ThO$_2$, $^{235}$U, IN HEAVY WATER MODERATOR. PART II. RELATIVE FISSION RATES, FAST FISSION RATIO, CONVERSION RATIO, AND COMPARISON WITH CALCULATIONS. Okazaki, Albert (Atomic Energy of Canada Ltd., Chalk River (Ontario)), June 1968. 41p. Dep. CFSTI. CAN $1.50.

Lattice parameter measurements of heavy water moderated lattices of 19-element clusters of ThO$_2$ containing 1.5 w/o UO$_2$ (93 at. % $^{235}$U) are presented. Relative $\text{Hf}$, $\text{Pu}$, and $\text{Th}$ fission rates, fast fission ratio, and conversion ratio measurements are described. D$_2$O, air, H$_2$O 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 buckling are given. Calculations made with the LATREP and HAMMER codes are compared with measured lattice parameters. Both codes underestimate the buckling by up to 1 m$^2$ with the largest discrepancy at the widest lattice pitch. The calculations overestimate the neutron disadvantage factor and conversion ratio. (auth)

Critical Experiments:

Lattices


Experimental values for infinite medium neutron multiplication factor, $k_{eff}$, have been determined for a graphite moderated supercell utilizing 2.5% enriched U fuel and 2.93 wt % Li-in-Al target in separate process tubes. In a three-to-one ratio. Determinations of $k_{inf}$ were made with the coolant channels wet and dry to obtain an estimate of the reactivity coefficients due to water flow. Both a 6 x 6 and 5 x 5 array of test cells were inserted in the PCTR test cavity for these measurements. Results of experimental measurements are listed. (auth)
Critical Experiments: Lattices


A comparison of the spatially dependent fast and thermal neutron flux profiles has been made in the Cornell University Zero Power Reactor, a light-water moderated and reflector, 2.1% enriched, UO₂ fueled, aluminum clad, triangular pitched critical assembly, using the core which has a nominal water-to-fuel ratio of 1.5:1. The radial buckling, defined by a least-squares fit of the radial flux profile to a J₃ Bessel function, is chosen as a convenient single-parameter characterization of the profile for comparison of flux profiles. The radial fast flux profile was determined by using a semiconductor detector to detect fission fragments from the fast fissioning of Th-232. The radial thermal flux was determined by the irradiation of manganese foil. The buckling detection was defined as the radial distance from the core center by performing the least-squares fit on data taken from the core center out to r. The thermal flux is defined by the thermal flux deviating significantly from the straight dependence for r greater than about two-thirds of the core radius. Considering the region in which the thermal flux generally follows the straight dependence, the ratio of the thermal buckling to the fast buckling in this region is used to account for the effective fission threshold for Th-232, used to experimentally determine the fast flux, is about 1.25 MeV. (Diss. Abstr.)


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1968

22976 (MIT-2344-12) HEAVY WATER LATTICE PROJECT
FINAL REPORT. Thompson, T. J.; Kaplan, L.; Driscoll, M. J.

CRITICALITY STUDIES — physics of heavy water moderated slightly enriched uranium-fueled lattices, (E/T); neutrons — buckling in heavy water moderated uranium oxide (UO2)-fueled lattices, (E/T); multiplication constant measurements in heavy water moderated uranium oxide (UO2)-fueled lattices, (E/T); uranium-235 — neutron fission ratio to uranium-238, measurement by gamma spectrometry.


Core 9 of the FRO fast critical assembly was diluted with heavy water to 24 vol. per cent, contained in thin-walled 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 neutron parameters for a low absorption target in a supercell consisting of a three-to-two fuel-to-target ratio with separate tubes of uranium fuel and bismuth metal target. Each tube is water cooled and surrounded by a graphite moderator on an 8.00" 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 pressure tube with the target elements. The graphite hole bars for the fuel cells contained 2.0" O.D. holes and strontium 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 separate 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 ThO2 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 ThO2 target at 77% theoretical density in separate process tubes. Determinations of k0 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)


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


III. REACTIVITY MEASUREMENT

1967


Critical Assemblies— perturbation measurements in ZPR-3, use of fusible materials for; perturbation measurements in ZPR-3, use of boron and tantalum for; flux measurements in ZPR-6, adjoint; Doppler effect measurements in ZPR-9.


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

In order to treat simultaneously a large number of experimental data relative to a given material, a program (CALI) has been written which determines a new set of “effective” multigroup constants consistent with the experiments considered and as close as possible to a reference set obtained by best-value criteria. Proper allowance is given to the uncertainties of the reference cross sections and to the errors associated with the integral data themselves. This allowance removes most of the difficulties encountered when correlating data obtained in assemblies with nuclear 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 thus 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 than 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, (b) dynamic method, in which steam bubbles were created in the reactor moderator. The effects of the voids on the reactor reactivity were measured and the void coefficients were evaluated. The first method, which simulates better the case of uniformly distributed air in the moderator, gives accurate results. The proposed second method, which simulates better the case of real steam bubble formation in the moderator, is simple and gives satisfactory results for fair approximation. (auth)

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


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


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.I.S.)


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 EHG. For some materials the reactivity change as a function of sample size has also been determined experimentally. The core of Assembly 1 consisted only of U enriched to 20% whereas the core of Assembly 3 was diluted with 30% graphite. The results have been compared with calculated values obtained with a second-order transport-theoretical perturbation model and using differently shielded cross sections depending upon sample size. Qualitative agreement has generally been found, although discrepancies still exist. The spectrum perturbation caused by the experimental arrangement has been analyzed and found to be rather important. (auth)


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


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)


The change in the Doppler coefficient due to crystalline binding effects was studied for UO2 in a UO2 lattice for fast reactor systems. The "weak binding," or the short compound nucleus lifetime (SCN) approximation was used, where the 238U atoms in the UO2 lattice are treated as a free gase with an effective temperature. Calculations were made to determine the dependence of effective temperature versus temperature using the experimental phonon distribution of Dolling, et al. Comparisons were also made with effective temperatures obtained assuming a Debye model for the crystal. It was found that for temperatures above 300°K the effective temperature dependence was well represented by the Debye model (a < 620°K). The temperature dependence of the reactivity was calculated for the ANL critical assembly, ZPR-I, 47, with the result that incorporation of the crystalline binding effects caused the calculated Doppler coefficient (dR/dT) to diverge from the 1/3T rule in the low temperature region. In qualitative agreement with the experimental results of Reynolds and Stewart. (auth)

The effect of Pu isotopic composition on the Doppler coefficient is examined in fast reactors having different chemical compositions of the fuel and different core volumes. It is found that for a given core volume and chemical composition the absolute value of the Doppler coefficient increases with increase of the amount of high Pu isotopes $^{239}$Pu, $^{240}$Pu, and $^{241}$Pu. (auth)

1968


Central reactivity contributions of gram-sized samples of $^{144}$Cm, $^{239}$Pu, and $^{235}$U have been obtained in a fast critical assembly of bare $^{239}$Pu in a spherical geometry. Resulting values are: $^{144}$Cm = (1276 ± 6)% cents/g at; $^{239}$Pu = (1393 ± 3)% cents/g at; $^{235}$U = (701 ± 2)% cents/g at. From these data, the critical mass of a bare sphere of $^{144}$Cm is estimated to be (27.7 ± 2.5) kg at a density of 13.5 g/cm$^3$. (auth)


Small BeO-modulated 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)
1967

A short description is given of the zero power facility ECO, its reference fuel elements UI/19/12, and a summary of the results obtained during the initial start-up experiments. Results of reactivity measurements on control elements and on certain perturbed core configurations are given, together with a list of the measured buckings 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 ±25% for relatively small homogeneous cores. The effective addition of reflector and geometric size as measured by the first method was less accurate than that measured by the second method. The values obtained by both methods coincided within the limits of the experimental errors. The method of measuring neutron flux distribution was not applicable in this case. (TTT)


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), TOPEX (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 DAMNET, a code for rearranging and altering the mode of the standard BCD ENDF/B library tape; ETONE, a code for preparing as MC library tape; and MC3, a code for generating multigroup constants from microscopic neutron data. Calculational results have been compared with experiment as well as results obtained using other nuclear data libraries. (auth)

Reactivity Measurement


Prompt neutron decay constants have been determined for unreflected and unmoderated subcritical cylinders of enriched uranium (93.15 wt % U235) by the Rosel-0 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 1% with those measured by the pulsed neutron method; the comparison with the results of S1 transport theory calculations showed disagreements as large as 20%. Reactivities as much as 33 dollars subsicritical were determined from the prompt neutron lifetime with cylinder height calculated by S1 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 % U235) by the Rosel-0 technique. The cylinder diameters were 17.77, 27.93, and 38.09 cm and the heights varied from 10.184 to 2.548, 8.431 to 5.399, and 7.602 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 S1 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 S1 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


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


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


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

The reactivity effect of thermal expansion of 239U metal has been measured and calculated in a series of neutron energy spectra. The correction of the measured temperature coefficient of reactivity for this effect to obtain the Doppler effect is shown for one of these spectra. The results of temperature coefficient measurements with 239U and Th in these spectra are given. (auth)


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

1968


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 1 describes the evaluation of the cross section data, testing of the data by comparison of calculations with integral experiment measurements, and an examination of methods used for critical assembly calculations. Neutron cross sections important for the sodium analysis have been evaluated as modifications to the ENDF/B data file. Calculations of criticality, reaction-rate ratios, and material worths using both the ENDF/B and modified ENDF/B data have been made for ZPR-III Assemblies 48, 48B, 49, and ZPR-VI Assemblies 2 and 3. The modified ENDF/B data are found to be over-reactive for these assemblies by 0.2% to 0.4% while the ENDF/B data are under-reactive by 1.3% and 0.7% for the 235U and 239Pu-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)


The second of a two-volume report on an analysis of sodium reactivity measurements in fast reactor critical assemblies is presented. Volume II presents the sodium void analysis. In Vol. I, emphasis is placed on cross section evaluation and data testing conducted 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. For the heterogeneity analysis, cell-homogenized cross sections are obtained by flux-weighting over the CCF using transport theory calculations of the spatial flux distributions. Overall calculations of the void reactivities with the MENDF/B data show better agreement with experiment, particularly for the 239Pu-fueled assemblies. For centrally voided regions the MENDF/B data yield the following qualitative agreement with experiment: 25% less positive than experiment for Assemblies 48 and 48B, 15% less negative than experiment for Assembly 2, and 25% more negative than experiment for Assembly 3. Regions with large leakage contributions tend to be consistently underestimated by about 15% with maximum discrepancies of 25% for all assemblies analyzed. Heterogeneity and transport theory effects on the void reactivities are found to be typically large for Assembly 48. (auth)
1968


Thesis.

A 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 $^{239}$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 $^{237}$U capture and $^{237}$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 $^{235}$U was detected by counting the $^{237}$Np γ-activity of irradiated foils using the γ-X-ray coincidence technique, and the fission ratio in $^{239}$U was obtained from the counting rate of a small fission chamber. The experimental results were reproducible to within about 1%. Systematic errors due to the heterogeneous core loadings and other effects add another 1 to 2% to the net uncertainties. The measured values obtained at the centers of the cores are in good agreement with results of multigroup calculations. (auth)
IV. NEUTRON FLUX SPECTRA

1967


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


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


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

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

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


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


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


1967

40346 (RPI-328-87, pp 22-59) FAST NEUTRON SPECTRUM PROGRAM. (Rensselaer Polytechnic Inst., Troy, N. Y.), Information is given on the design and development of a spherical lead electron target for use as the neutron source in fast assemblies. Measurements were made of the neutron spectrum from a 44 x 40 x 38 in. 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 $^{137}$Cs- Na2 detector. An effort was made to formulate the adjoint fast reactor spectrum, and an application is given for a representative oxide core. A variational principle is developed for the determination of decay constants in pulsed fast neutron assemblies. A group constants averaging procedure for few-group importance-function and reactivity calculations in fast reactors is presented. A six-group importance function calculation is given in graphical form for GODIVA using importance-averaged and flux-averaged group constants. (S.F.L.)


The difficulties inherent in spectrum determinations in reactors at positions remote from the core center are considered, and the suitability of $^{14}$Li and $^{10}$Be semiconductor sandwich spectrometers for this work is discussed. Spectra obtained in many positions in the DAPHNE reflector are compared with calculated spectra. (auth)

35687 (ANL-7326, pp 481-5) A COMMENT ON THE COMPARISON OF THEORETICAL WITH EXPERIMENTAL NEUTRON SPECTRA IN FAST CRITICAL ASSEMBLIES. Travalli, A. (Argonne National Lab., Ill.). The experimental data obtained by Bennett in the measurement of central neutron spectra in fast critical assemblies have a resolution of 15 to 20% in energy over the energy range, which extends approximately from 1 keV to 1 MeV. The high resolution, combined with the small statistical error of the data, makes it possible to measure, in addition to the macroscopic behavior of the neutron spectrum, some of the microscopic variations of the spectrum at 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 0.5 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 (dioxide reflector of BR-1 reactor) are presented. The neutron spectrum in the large UO2 block is investigated by the time-of-flight method and sandwich techniques. The results of the measurements are compared with the calculations of the neutron propagation in UO2 by 26-group sets 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 fission spectra can be explained on account of the interaction of neutrons with the materials of the active zone. With transmission coefficients up to $10^{-2}$ to $10^{-4}$, the total cross section was measured for water, carbon, and lead in the energy interval from 1 to 3 MeV with errors of 2.5 to 3% at $E_n < 0.4$ MeV and 1 to 2%, at $E_n > 0.6$ MeV. The total cross section for carbon and water at $E_n > 2$ MeV satisfactorily agrees with published data. For lead and water below 2 MeV, the sizes of the cross sections are lower (in comparison with the data for thick samples), which is explained by the passage of neutrons through the minimum value of the cross section. For lead in the energy range above 3 MeV, a tendency to irregularities was observed. The dependence of the luminescence of the stilbene crystal on the energy made it possible to determine the scattering from carbon samples. (tr-auth)
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 0.8 to 24 MeV region by means of threshold detectors. Fission chamber determinations and activation studies were also made. It was found that below 3 MeV, the reactor spectrum differs from the fission spectrum; this is due to the low energy transitions of neutrons caused by inelastic processes. Above 3 MeV, the spectra are fairly similar. The evaporation model characterizes the fission neutron spectrum up to 24 MeV; it remains to be determined how far this range extends. (TTT)


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-504 lithium glass detector for intermediate energy neutrons. A 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)


Construction of the Subcritical Time-of-Flight Spectral 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


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.)

39548 (RPI-328-123, pp 18-25) FAST REACTOR PHYSICS EXPERIMENTAL. (Rensselaer Polytechnic Inst., Troy, N. Y.). The analysis of time-of-flight measurements of fast neutron spectra in depleted uranium was continued, and computer programs were developed for analyzing time-of-flight data on fast neutron spectra. Preliminary results of measurements of position-dependent fast neutron spectra in iron are also reported. (D.C.W.)

48424 (RPI-328-133, pp 35-55) FAST REACTOR PHYSICS: EXPERIMENTAL. (Rensselaer Polytechnic Inst., Troy, N. Y.). A series of position-dependent fast neutron spectrum measurements in rectangular assemblies of Fe and depleted U was performed; comparisons were made with theoretical predictions. Preliminary studies of an 27Ag 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.)

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V. NEUTRON CROSS SECTIONS

1. Sources of Data

1967


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


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

39149 (BNL-50061A) SIGMA CENTER, NEUTRON CROSS SECTION COMPILATION CENTER SCIRS NEWSLETTER. PART A, REFERENCE AND BIBLIOGRAPHY. (Brookhaven National Lab., Upton, N. Y.), June 1967. 139p. Dep. CFSTI.

A listing by reference of the literature incorporated into the 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 ≤ A ≤ 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-405, 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.)
1967


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,f) 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 (so far as they refer to specific target nuclei. (S.F.L.)

35546 (ANL-7330, pp 3-14) STATUS OF BASIC NUCLEAR DATA REQUIRE FOR FAST BREEDER REACTOR DEVELOPMENT. Cox, S. A. (Argonne National Lab., Ill.).

The present precision and availability of nuclear data are discussed. Fission cross sections are shown in the light of recent 235U measurement at A.W.R.E., Aldermaston. Since the 235U 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 235U fission cross section, but also on the 96B(n,a) cross section and measurements of spherical shell transmission. Each normalization procedure is discussed in detail. It is concluded that much of the existing disparity in results for fission and capture cross sections would be greatly reduced by a critical renormalization of all of the data to currently acceptable values of the 235U fission cross section and the 15B(n,a) cross section. Recent re-evaluations of measurements of spherical shell transmission also tend to reduce the disparity. Neutron scattering cross sections from 0.3-1.8 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 ~3-5 MeV. More data are needed from 1.5-5 MeV. Above 5 MeV the optical model can be used to interpolate between measurements rather widely spaced in energy. Recent P measurements confirm the nonlinearity of P vs E, and also suggest the presence of an anomaly at ~6.4 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)

1968

11932 (TID-2357(Suppl.1)) CINDA: AN INDEX TO THE LITERATURE ON MICROSCOPIC NEUTRON DATA. (Division of Technical Information Extension (AEC), Oak Ridge, Tenn. ENEA Neutron Data Compilation Center, Gif-sur-Yvette (France)). Oct. 15, 1966. 327p. (EANDC-66(U); CCDN-Cl-13). Dep. mn. CFSTI $3.00 cy, $0.65 mn.

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


The heavy 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,2n), (n,3n)). An analysis of 235U and 238U for which considerable data is available has justified the usefulness of this approach to evaluation. The non-spherical optical model permits the evaluator to separate the elastic and inelastic components of a measured angular distribution for these isotopes. Combined with an appropriate compound-nucleus model, it is possible to differentiate between conflicting sets of 235U capture data within the range of 10 to 100 keV. (auth)


NEUTRONS, FAST—reactions (n,a) with plutonium-239 and ~240 at 5 to 150 keV, cross sections for (E)

NEUTRONS, FAST—reactions (n,a) with uranium-235 at 30 and 64 keV, cross sections for (E)

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

Uranium-235—neutron fission cross section at 50 and 64 keV (E)

Uranium-235—neutron fission at 7 to 14 MeV, neutron emission in (E)

Uranium-235—neutron fission at 7 to 14 MeV, neutron emission in (E)

Neutron Cross Sections:

1. Sources of Data


The spherical shell method for investigating inelastic scattering cross sections was used in a fast-reactor core environment. The changes in 238U/235U, 238U/239Pu, and 239Pu/235U fission ratios caused by placing shells of graphite, sodium, aluminum, lio, 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 Betha, Bueter, and Carter. Comparisons of the experimental data with values calculated using two multigroup cross-section sets show clearly where these data sets are accurate and where they are in error. (auth)
1967

25670 (ANL-7320, pp 80-7) EVALUATION OF BASIC CROSS-SECTION DATA BY ANALYSIS OF FAST CRITICAL ASSEMBLIES. Fillmore, F. L.; Specht, E. R.; Vernon, A. R.; Ottewitte, E. H. (Argonne National Laboratory, Canoga Park, Calif.). Calculations of the properties of a number of fast critical assemblies are being carried out as part of a continuing program for testing and evaluating multigroup cross sections for fast calculations. The experiments included in the present analysis have been chosen to cover a wide range of fuel materials and reactor spectra. This increases the ability to identify errors according to the material and energy range in which they occur. The assemblies studied include a number of $^{238}$U-fueled ZPR-III assemblies; ZPR-III-48, ZPR-III-46, and ZPR-III-45, which were mainly used for Doppler measurements; the SEFOR mockup, ZPR-III-47; the Pu-fueled assembly, ZPR-III-48; ZEBRA Core 3; the small fast assemblies POPPY and JEZEBEL (fueled with metallic Pu), and GODIVA (fueled with metallic Pu). The properties examined include critical mass, Doppler effect, fission ratios, and the reactivity worths of various material replacements. The properties of the small fast assemblies depend only on the high-energy data; they are practically independent of cross sections below 50 keV. However, the larger assemblies require accurate data down to about 100 eV. All of the multigroup cross sections have been constructed using basic microscopic cross-section data and have not been adjusted to force agreement with the critical-assembly data. For the large, soft-spectrum assemblies, the cross sections were weighted by a line-group spectrum calculated for each assembly. A list of 25 references is included. (auth)


The breeding gain of plutonium-fuelled fast reactors is strongly influenced by the capture-to-fission ratio $q_0$ of $^{239}$Pu. In the softer spectra associated with a large dilute fast reactor, the uncertainty in $q_0$ is of the order of ±25%. To reduce this uncertainty, two new techniques are being developed for use in zero-energy fast reactor lattices. In the first method, measurements are made in a lattice which is arranged to have an infinite multiplication constant $k_x$ near to unity, so that $k_x$ can be determined by a null reactivity technique without introducing significant systematical errors. All the important neutron fission and capture rates, except for the capture rate in $^{239}$Pu, are then measured in this modified lattice; and $q_0$ 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 $q_0$ is known. Some preliminary results obtained by the first technique indicate that current nuclear data sets underestimate $q_0$ significantly in dilute fast reactor lattices. (auth)


1968


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 $B_n$ 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)


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


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. (J.A.T.S.)

Neutron Cross Sections:

1. Sources of Data
Neutron Cross Sections:

1. Sources of Data

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

Preliminary results of a program to measure capture cross sections for fast reactor materials and to measure and calculate capture-to-fission ratios for $^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{235}$U, $^{236}$Pu, and $^{238}$U 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.)


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

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

48414 (GA-8773) INTEGRAL NEUTRON THERMALIZATION.

A number of theoretical studies completed during this period are discussed. The final conclusions resulting from the first principles calculation for beryllium are summarized. A theoretical scattering law for UO$_2$ was completed; however, some additional numerical studies remain to be done before this work can be incorporated in the ENDF. A new model for polyethylene is also described, and comparisons between results of this model and experiment are presented. The lattice dynamical model for beryllium oxide was used to calculate a frequency spectrum and a scattering law in the incoherent approximation. A calculated total cross section for DeO, including coherent elastic scattering, is presented as part of this work. Some recently completed work on multiple scattering in double differential experiments is also described. This work relates to problems involved in the use of specially constructed samples designed to reduce multiple scattering. Reports on work in progress include an outline of efforts being made to improve capabilities for computing coherent inelastic scattering. Also some preliminary work is reported concerning efforts to broaden significantly the scope of the theoretical analyses underlying ENDF scattering laws by relating the temperature dependence of the frequency spectra to 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


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


A bibliographic guide to experimental and theoretical information on neutron cross sections, resonance parameters, thermal scattering laws, fission parameters, and other related quantities is presented. A one-line format includes the element, isotope, or compound studied; the quantity studied; the type of investigation; the type of references; 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.)


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


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

20578 (WASH—16789) REPORTS TO THE AEC NUCLEAR CROSS SECTIONS AND RESERIES. (NCSTI). Dep. CFSTI.

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


The activities of the center are summarized. Recommended values of the thermal neutron capture and fission cross sections and resonance integrals of $^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{242}$Pu, $^{243}$Pu, $^{244}$Pu. which are based on a systematic treatment of available data, are presented. The random matrix theory of nuclear cross-section fluctuations was extended to include the possibility of time reversal violation in nuclear forces. Optical-model parameters that provide a good fit to the cross sections of $^{201}$Pb, $^{202}$Pb, and $^{203}$Pb at 1 MeV were obtained. Resonance parameters for 46 resonances of Ho below 35 keV were used in fitting total cross sections on to a Breit—Wigner multilevel scattering and single-level absorption formula. (D.C.W.)


The results obtained from a comprehensive experimental study of elastic and inelastic neutron scattering are reported. The incident neutron energy interval was 0.3 to 1.5 MeV and scattering from 50 elements extending from Be to U was investigated. Fast neutron time-of-flight techniques including a multi-angle detector system and fully automated computer control were utilized to achieve a good scattered neutron resolution. Differential elastic and inelastic scattering cross sections were determined at eight or more angles at incident neutron energy intervals of 50 keV or less. The elastic angular distributions are expressed as Legendre expansions of up to six terms. The observed differential inelastic cross-sections are integrated to obtain the respective inelastic excitation cross sections. The experimental results are compared with optical-model Hauser—Feshbach calculations; and it is shown that interpolations of experimental values, based on the model, are valid. Experimental evidence for intermediate resonance structure, with fluctuation effects, and nuclear deexcitation is presented. The influence of each on calculation is illustrated. (auth)
1968


Some recent measurements of differential cross sections and y spectra for the reactions $^{14}$N(n,t)$^{14}$N, $^{14}$N(n,p)$^{14}$C, and $^{14}$N(n,α)$^{11}$B; the neutron total cross section of $^{14}$N; spectra from resonance and thermal neutron capture by $^{16}$Ce, $^{18}$K, $^{19}$K, $^{19}$Re, $^{19}$Re, $^{20}$Tl, $^{21}$Tl, $^{19}$Sn, $^{19}$Sn, and $^{11}$Sn; and capture and fission cross sections for $^{239}$Pu from thermal to 30 keV and for $^{233}$U from thermal to 1 eV are summarized. Calculations of the elastic and inelastic scattering cross sections for $^{56}$Fe are also reported, as is the status of the electron linear accelerator. (D.O.W.)


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 $^{238}$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 also interpolation on theoretical grounds. Within the framework of contracted research with the Association EURATOM-Karlsruhe on fast reactors, an evaluation was made of basic nuclear data for the high Pu isotopes. Some pertinent parts and aspects of the evaluation are summarized. Resonance parameters and cross sections are presented for $^{239}$Pu, $^{241}$Pu, and $^{242}$Pu in the form of experimental and recommended data. Complete sets of parameters include the first 43, 61, and 20 resonances of these isotopes, respectively. Average parameters are derived from these sets, to be used at higher energies where either the parameterization 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)

46 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 are still several uncertainties in these parameters, a compilation has been made of the most detailed cross-section data available and reaction products for the four major elements in tissue (i.e., H, C, N, and O). The compilation is for neutron energies below 15 MeV, but the energy interval requiring the most study and analysis was that from 2.5 to 15 MeV. Particular attention was directed to the nonelastic reactions [e.g., the C(\text{n},\text{n}0) \text{3a} reaction]. Average values for the energies of the various charged particles as a function of the energy of the incident neutron have been computed. These values were compiled to provide a basis for revision of the dose-distribution functions for neutron exposures of man and of animals used in radiobiological studies. An analysis of the results of various measurements are compared with calculated values based on these cross sections and with the values listed in NBS Handbook 63. (auth)


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 \text{4Li} and \text{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 AWRE data file originated by K. Parker of Aldermaston. Values for \sigma_{\text{LAB}} and \xi, along with Legendre coefficients for the elastic scattering angular distributions, were received from H. Alter of Atomics International. Plots of the original AWRE cross-section data converted to the ENDF/B format are presented. The ENDF/B listings for the \text{4Li} and \text{7Li} data, as they appeared on the first version (approximately February 1967) of the ENDF/B tape, are shown. (auth)


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 \text{244Pu} and \text{248Am} is reviewed. An analysis of Tweed abundances is made to obtain capture-to-fission ratios for the odd-A plutonium isotopes through A = 263. 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 \approx 159, one needs, in this formalism, an accurate way of predicting shell corrections or nuclear masses. (auth)


Some measurements of the capture and fission cross sections of \text{234U} in which the target foils were exposed to a moderated flux from a nuclear explosion are summarized. Relation fission cross sections are presented for energies of 20 to \text{10^6} eV, as well as information on the capture-to-fission ratio between 21 and 63 eV. The results of resonance analyses using both single-level and multi-level formalisms indicate that the fission of \text{234U} takes place through several channels. (D.C.W.)
Neutron Cross Sections:

2. ENDF/B Tapes ...


27568 \textbf{THE }^{239}\text{U} \textbf{ FISSION AND CAPTURE CROSS SECTIONS AND THEIR ANALYSIS AT LOW ENERGIES. Bergen, Delmar W. (Los Alamos Scientific Lab., N. Mex.). Dec. 1966. Contract W-7405-eng-36. 76p. Dep. CFSTI. Thesis, Submitted to Univ. of New Mexico (Albuquerque). The }^{239}\text{U} \textbf{ fission and capture cross sections were measured using a nuclear-device neutron source and time-of-flight techniques. Cross section data are presented from 20 to } 10^6 \text{ eV for fission and from 20 to } 63 \text{ eV for fission + capture. The resonance region ( } 25 \text{ eV to } 63 \text{ 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 Sections:
2. ENDF/B Tapes...


From IAEA Conference on Nuclear Data, Paris, France. In an experiment in Nevada in June 1965 a nuclear device with a yield equivalent of 1.2 kton of TNT provided the neutron source for time-of-flight measurements over a path of 185 m in vacuo. To exploit the combination of high flux and high energy resolution, new recording techniques were required. Because more than a million data points are acquired in any one exposure of a set of targets, the general problems of data retrieval and processing required special attention. Measurements of fission cross sections of the nuclides $^{235}$U, $^{238}$U, $^{239}$Pu, $^{241}$Pu, $^{242}$Pu, $^{244}$Pu, $^{241}$Am, and $^{237}$Np are reported. In addition, capture-to-fission ratios of $^{232}$U and $^{239}$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)

Dep. mn.

USING A LARGE LIQUID SCINTILLATOR. Kompe, D. (Kernforschungszentrum, Karlsrube (West Germany). Institut fuer NTl1Ta, 18zW, lUW, lMWO AND lUW CROSS SECTION DATA 4274 1(GEMP448) EVALUATION AND COMPILATION Nuclear Materials and Propulsion Operation). Nov. 11, 1966. $0.65 mm

Contract AT(40-1)-2847. 91P. Dep. mn. CFSTI $3.00 CY.


The nineteen-group beryllium cross sections have been revised with intc~mal mvapurcd dain: and from the discrqmn,+w 11 1s


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of the data, and the error matrix resuitlng fm.f.m the hmit squarra

calculatin. (D. C.W.)

21410 NEUTRON CROSS SECTIONS OF HYDROGEN IN THE TNP~Ky RANGE 0.0001 eV TO 20 MeV. Horsley, A. (Atomic Weapons Research Establishment, Aldermaston, Eng.,). Nucl. Data, Sect. A, 2: 243-62 (May 29, 1967). 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 spontaneous-fission neutron spectrum of 238U was measured from 0.003 to 15.0 MeV by time-of-flight techniques. A hydrogenous liquid scintillator was used as a detector at the higher energies, while a 1.1-loaded glass scintillator was used at the lower energies. The measured spectrum has an average energy of 2.318 MeV. A Maxwellian distribution, N(E) = E^3 exp(-E/1.56), fits the data well for 0.5 < F < 10.0 MeV. Below 0.1 MeV, N(E) has a F dependence but with values ~25% larger than those predicted by the extended Maxwellian spectrum. The results are interpreted in terms of a simplified evaporation model. (auth)


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.)

31810 235UF FISSION NEUTRON SPECTRUM FROM 0.003 TO 15.0 MeV. Meadows, J. W. (Argonne National Lab., III.). Phys. Rev., 157: 1075-82 (May 1967). The spontaneous-fission neutron spectrum of 235U was measured from 0.003 to 15.0 MeV by time-of-flight techniques. A hydrogenous liquid scintillator was used as a detector at the higher energies, while a 1.1-loaded glass scintillator was used at the lower energies. The measured spectrum has an average energy of 2.318 MeV. A Maxwellian distribution, N(E) = E^3 exp(-E/1.56), fits the data well for 0.5 < F < 10.0 MeV. Below 0.1 MeV, N(E) has a F dependence but with values ~25% larger than those predicted by the extended Maxwellian spectrum. The results are interpreted in terms of a simplified evaporation model. (auth)


The available resolved resonance data for the isotopes of zirconium are evaluated, with the view to compile a file of best tabulated neutron cross-section data. These data are also compared with integrally measured data; and from the discrepancies it is concluded that there is a need for better measurements of the radiative widths of the resonances for all zirconium isotopes, especially for the s-wave resonances of 92Zr; 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, Fe, Mo, Ni, O, 195Pt, Na, 185U, 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 241Am and 248Am 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 241Am. The fission cross section of the doubly odd nuclide 249Am is about twice that of 248Pu 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 as is discussed, is also data processing for single-crystal fast-neutron scintillation spectrometry. (D.C.W.)
16002 (GA-6133) NEUTRON CROSS SECTIONS FOR NIO-
BIUM. Allen, M. S.; Drake, M. K. (General Dynamics Corp., San Di-

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 238U FISSION CROSS

The neutron-induced fission and capture sections of 238U were measured by time of flight with a nuclear detonation as the neutron source. Cross-section data are presented from 10 to 10^5 eV for fission and from 30 to 63 eV for the capture-to-fission ratio. 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-Moore 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 239U FISSION AND CAPTURE CROSS SECTIONS

The 239U fission and capture cross sections were measured using a nuclear-device neutron source and time-of-flight techniques. Cross-section data are presented from 20 to 10^5 eV for fission and from 20 to 63 eV for fission + capture. The resonance region (20 to 63 eV) was fitted with both a single-level function consisting of a sum of Breit-Wigner levels and the Reich and Moore multilevel function based on R-matrix theory. The resulting resonance parameters are listed and discussed. In order to establish the validity of the resonance parameters derived from the multilevel fit, a study is presented of the cross section derived from two and three hypothetical resonances under various conditions and of the cross sections obtained from randomly generated resonances. (Diss. Abstr.)

18228 NEUTRON-INDUCED FISSION CROSS SECTION OF

The neutron-induced fission cross section of 241Am was measured from 0.02 eV to 6 MeV by the time-of-flight method at the Livermore 30-MeV linear electron accelerator. The data are normalized at 0.0253 eV to a value of 800 b measured in a reactor thermal-neutron flux. The cross section at 0.2 eV is 4700 b; at 1 eV it is 540 b; and at 4 MeV it is 2.1 b. The data are analyzed to obtain values for the neutron strength function (S^2/D) of 1.4 X 10^3, the level spacing D = 1.2 eV, and the quantity 2πS^2/D = 2.5. All three quantities are quoted per spin state. The high cross section at low energies can be attributed to the unusually high value for 2πS^2/D, and to the existence of a very large resonance at 2.173 eV. The fission cross section of 241Am was also measured in the MeV region and found to be 1.96 ± 0.2 b at 2.5 MeV. (auth)

24873 SELECTED FISSION CROSS SECTIONS FOR 232Th,

The fission cross sections of 232Th, 233U, 235U, 237U, 239U, 239Pu, 241Pu, 242Pu, and 244Pu 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-5224) THE AVERAGE NEUTRON TOTAL
CROSS-SECTION OF 10B FROM 100 eV TO 10 MeV AND ABSORPTION

The transmission measurements to be described were made on the 126-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 oE/T = (610.3 ± 3.1) E^-7 + (1.95 ± 0.10) barns. The deviation of oE/T from this value above 40 keV is due to an increase in elastic scattering cross section. Subtraction of the latest scattering cross sections measured by Mooring et al. yields an absorption cross section that is proportional to E^-7 up to at least 250 keV. (auth)

16001 (GA-7462) NEUTRON CROSS SECTIONS FOR 231Pa,
S)-167. 85p. Dep. CFSTI.

A survey was made of the available experimental cross-section measurements for 231Pa, 233Pa, and 235U. 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 Cross Sections:

2. ENDF/B Tapes...

1968


A survey was made of the available experimental cross-section measurements for 238U and 239U. Sets of recommended neutron cross sections are presented for neutron energies from 0.01 eV to 10 MeV. Resonance parameters are also included. (D.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 of the secondary neutrons are given. Also, gamma-ray production cross sections were obtained as well as energy and angular distributions of the secondary gamma rays. In general, the recommended data were based on experimentally measured data. However, where no experimental data were available, the recommended cross sections were obtained using model calculations. (auth) (USGRDR)


The present (April 1967) Evaluated Nuclear Data File is (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 (U.K.) is maintained wherever practical. In the case of neutron 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 monochromatic beam of fast neutrons produced by a nuclear reaction in a "white" spectrum and a pulsed beam time-of-flight technique was used to measure neutron total cross sections in the 2 to 10 MeV region. Total cross sections for the elements Mg, Al, Ca, V, Cr, Pd, Ag, and Pb were measured at 1% average uncertainty in steps of 0.08 nm in energies from 0.001 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. (Disscr. Abstr.)


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


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 Petoled nuclear explosion. The total radiation width, $\Gamma_N$, was determined for $2.1 \pm 0.0$ levels with $\Gamma_N = (19.1 \pm 0.6$ (stat.) $\pm 1.4$ (syst.)) x $10^{-4}$ eV. There appears to be a significant variation in the value of $\Gamma_N$ from resonance to resonance. Approximately 200 weak resonances were identified which cannot be assigned to discrete levels. Analysis of these weak resonances, assuming $\Gamma = 1$, 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 $1.8 \pm 0.3 \times 10^{-4}$. (auth)


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


A cubic spline curve fitting method and a statistical theory of unknown systematic errors are combined to give a practical computer-orientated method of evaluating neutron cross sections. Particular attention is paid to reconciling sets of discordant data. The input data and evaluated curve can be displayed on a CRT graphical display unit. Among program output examples given are evaluated curves for several cross sections of $^{127}$I. (auth)
Neutron Cross Sections:

2. ENDF/B Tapes...


Work performed under United States-Britain 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, Al, H, He, Fe, Mo, Ni, O, 228Pu, Na, 235U, and 238U. (D.C.W.)


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. (autth)


The energy dependence of the neutron fission cross section of 241Am was investigated from 0.02 eV to 6 MeV. The measurements were normalized to a value for the fission cross section of 6600 barns at 0.0253 eV. Below 3.7 eV and above 300 keV, the cross section was corrected for 241Am in the sample. The data are presented in terms of mean values and resonance integrals over the GAM group structure. The resonance integral above 0.5 eV is 1570 ± 110 b. The data below 3.5 eV were analyzed in terms of a sum-of-single-levels fit. For the six resonances below 3.5 eV, the average fission width is 0.45 eV. (D.C.W.)


The evaluation of 59Na neutron cross section data was carried out for the ENDF/B file. Data were evaluated from 10^-6 eV to 15 MeV for the following neutron reactions: total, elastic scattering including Legendre polynomial expansions of the angular dependence, nonelastic, inelastic including resolved levels, (n,p), (n,a), and (n,2n). Graphs of the evaluated data are compared with experimental data. (autth)
1968


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


Evaluated libraries of cross sections were prepared for natural hafnium and its isotopes 177Hf, 178Hf, 179Hf, 180Hf, and 181Hf. The libraries contain total, elastic, capture, inelastic, (n,p), and (n,2n) cross sections and elastic scattering Legendre moments below 15 MeV. The most recent experimental data were used in the evaluation; and, whenever data were not available, theoretical calculations were made. (auth)

48416  (KFK–120(Pl.1)) NEUTRON CROSS SECTIONS FOR FAST REACTOR MATERIALS. PART I: EVALUATION. Schmidt, J. J. (Kernforschungszentrum, Karlsruhe (West Germany), Institut fuer Neutronenphysik und Reaktortechnik), Feb. 1966. 1400p. (ENDFB(E)–55(U(PL.1))).

A comprehensive documentation and critical review of the available experimental and theoretical information on microscopic neutron cross section data are presented, and the procedures that were used to reduce this information into cross section sets in graphical and tabular form are described. The neutron energies considered range from 0.01 eV to 10 MeV, with emphasis on fast and resonance neutrons. The materials covered include C, Cr, Fe, H, Mo, Ni, O, 235U, 241Pu, and 238U. (D.C.W.)


Total, capture, and fission cross sections for $n + 235U$ were simultaneously least squares fit over the energy range 0.4 to 61.4 eV using the single-level Breit–Wigner formula. Fifty-three resonances were found in this range. Good agreement was obtained between integral data and this fit to differential data for the resonance integrals of 235U. Resonance parameters are given, along with brief statistical analyses of them. A multigroup cross section library from 0.001 eV to 10 MeV is presented. (auth)


Some recent experimental data and evaluations of activation cross sections for several nuclei are analyzed; and new GAM, TNS, and BfT libraries based on these analyses are presented. The cross sections for the isotopes 240Pu, 241Pu, and 238Pu, for natural Pb and Zr are analyzed. A preliminary analysis of the 116mNb(n,γ) cross section is also summarized. (D.C.W.)
V. NEUTRON CROSS SECTIONS

3. Wide Ranges in Energy


The neutron total cross section of 238Pu was measured from 0.001 to 6500 eV. These data give single-level Breit-Wigner parameters for resonances below 200 eV. The observed total cross section at 2200 m/sec is 588 barns. A value of 552 barns was calculated for the effective (equivalent 1/ν) thermal absorption cross section. Parameters of individual resonances below 200 eV and average parameters at higher energies give a resonance absorption integral of 164 ± 15 barns, and a value of 1.10 ± 0.20 x 10^{-4} for the s-wave neutron strength function (1/2D). (auth)


Measured values of η for 239Pu and 239U are given for the energy range 0.04 to 11 eV, together with fission and total cross sections for 239U. Average values of η were calculated and also, in the case of 239U, 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 239Pu 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, 1966), 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,3n reaction. Two standard neutron sources, a 238Pu/Ra-Be source and the AWRE 238Pu spontaneous fission source, have now been calibrated at the National Physical Laboratory, Teddington, England, and in the boron pile. The count rates of the standard sources in the boron pile can be used to obtain a second independent value of the pile efficiency and hence give information on the correctness of the boron pile 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 244Cm, 239Pu, and 241Pu 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 232Th(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 232Th target in the energy region 0.75 to 1.0 MeV shows up clearly in the energy dependence of the fission cross section. The disagreement between the known thermal neutron fission cross section 0.06 ± 0.02 mb and the value extrapolated from α_f at higher energies is discussed. (auth)
were simultaneously fit with a single-level Breit-Wigner equation when weighting proportional to the inverse of the cross section used in the least-squares analysis, and another set was obtained to 6 Mcv. The theoretical cross sections are obtained by optical-model calculations including spin-orbit interaction and by Hauser-Feshbach theory. Transport cross sections are calculated both from the experimental and the theoretical differential cross sections. Calculations of inelastic 11-group cross sections are performed using the continuum model for the compound nucleus in the region above 1.4 MeV and assuming five excitation levels below 1 MeV. Between 1.0 and 1.4 MeV, the cross-sections are obtained by taking a mean value between the results from the continuous and the discrete level theory.

### Neutron Cross Sections:

#### 3. Wide Ranges in Energy

**32931** (ANL-7320, pp. 16-21) \textit{Fission Cross-Section Measurements of }^{235}U, ^{239}Pu, ANd ^{241}Pu \textit{in the Energy Range from 1 to 25 keV.} James, G. U. (Atomic Energy Research Establishment, Harwell (England)).

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


The effective resonance integral and Doppler coefficient of \(^{235}U\), \(^{239}Th\), and \(^{241}Pu\) 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.


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Neutron Cross Sections:

3. Wide Ranges in Energy


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


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1967

21525 TOTAL NEUTRON CROSS SECTION OF $^{133}$Pa.

Transmission measurements on $^{133}$Pa were taken with the Materials Testing Reactor (MTR) fast chopper. The total cross section was calculated in the energy range from 0.01 to 10,000 eV. These measurements were made on 700 mg of chemically separated $^{133}$Pa in an oxide form. The protactinium was produced by irradiating 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 Breit–Wigner resonance parameters were obtained for the resonances below 16 eV. The average parameters give a value of $0.75 \pm 10^4$ for the s-wave neutron strength function $f_0/D$. Weighting the level spacings inversely as $2J+1$ gives the average observed level spacings per spin state of 1.10 and 1.84 eV. A second-order polynomial least-squares fit to the $\sigma(E)$ data between 0.01 and 0.10 eV gives a 2200 m/sec total neutron cross section of 56 ± 3 b, superceding a value of 57 b given previously. The resonance-absorption integral for neutrons with energies above 0.4 eV was calculated to be 501 ± 45 b. (auth)
Neutron Cross Sections:
3. Wide Ranges In Energy


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

The fission cross-section ratios $^{235}$U/$^{239}$U, $^{239}$Pu/$^{239}$U, and $^{239}$Pu/$^{237}$U were measured with pulsed, monenergetic neutrons in the energy range 1.0 to 5.0 MeV with time-of-flight background discrimination. Vacuum evaporated fissile deposits (~0.5 mg/cm$^2$) were placed back to back between two 9.5 cm$^2$ surface barrier detectors. Slow and fast output signals were obtained simultaneously from each detector by means of separate electronic systems. Slow, linear pulses which exceeded a lower bound set to reject alpha particles were identified as fission events. The fraction of fragment pulses below this bias, determined from an extrapolation of the pulse-height spectrum, was ~1.2%. Only those fission events which occurred during and a few nanoseconds after the neutron burst were recorded. This time interval, determined by the measured time resolution (1.3 nsec FWHM) and the time walk of the smallest pulses (~2 nsec), was typically 6 nsec. Fission events induced by scattered neutrons which occur at later times were excluded. Characteristics of this detector system and preliminary data were reported earlier. Present results include additional data on $^{235}$U/$^{239}$U and $^{239}$Pu/$^{239}$U and new data on $^{235}$U/$^{237}$U.


Neutron total cross-section measurements were made between 100 eV and 1 MeV on nuclei near the mass-100 and mass-240 p-wave resonances using the Harwell "booster" pulsed neutron source and the 120-m and 300-m spectrometers. The s-wave strength function $s$ and distant level parameter $R^c$ 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 $^{60}$Nb, $^{90}$Mo, $^{108}$Mo, $^{128}$Rh, $^{137}$Th, $^{132}$U, $^{235}$U, $^{239}$U, and $^{239}$Pu.

35295 (TPM-RFR-698) $^{239}$Pu EVALUATIONS. Wallin, Marie (Aktiebolaget Atomenergi, Stavdikt (Sweden)). Dec. 7, 1967. 11p, Dep.

The updating of the Speng library for $^{239}$Pu to reflect recent measurements and evaluations of the cross sections and resonance parameters for $^{239}$Pu is discussed. (D.C.W.)
Neutron Cross Sections:

3. Wide Ranges in Energy


From Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany. See STI/PUB-165 (Vol. II); CONF-871043 (Vol. 2).

Integral values of $\sigma$ for $^{239}$Pu and $^{235}$U have been deduced from experiments in two different cores (median $^{239}$U fission energy 50 and 180 keV, respectively) in the IRO reactor. The method of measurement is the same as that used for instance at ZEHA.

The experiment includes reactor measurements of the sample material and of a standard, $^{195}$Pt, and as well as an absolute determination of the fission rate in $^{239}$Pu and $^{235}$U and the capture rate in $^{195}$Pt. The fission and capture rates are measured for the hard spectrum core measurements, both for $^{239}$Pu and $^{235}$U. The measured value for $^{235}$U in the soft core is slightly higher than the calculated one. The discrepancy for $^{239}$U is large and hitherto unexplained. (auth)

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

URANIUM—neutron cross sections for, crystalline effects on Doppler-broadened resonance integrals for, crystalline effects on (M.L. S.)

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

From Conference on Physics of Reactors, Milan. See CONF-66-469.

Use of a generalized optical model for analysis of the $^{235}$U cross section is described. Differential cross sections are shown for $E = 0.650$ MeV, 1.1 MeV, 2.5 MeV, 4.1 MeV, 7.0 MeV, and 14.1 MeV. The equations used for the analysis are given. Total and differential inclinative cross sections are shown. Behavior of the Legendre polynomial expansion coefficient is determined as a function of energy. (M.L. S.)


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 \( \nu \) for fissile and fertile isotopes are reviewed. The absolute determination of \( \nu \) 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 \( ^{235} \text{U}, ^{239} \text{Pu} \), and \( ^{241} \text{Pu} \). The available data relating the dependence of \( \nu \) on incident neutron energy are tabulated, and straightline fits to the data are made by the method of least squares. These results are of direct value in reactor calculations and the related evaluation of nuclear cross sections. Although there are instances where the data indicate structure of a more complicated nature, no attempt is made to analyze these situations. For practical purposes, a reasonably satisfactory fit to the data can be made with not more than two straight lines. (auth) (UK)


From 14th Annual Meeting of the American Nuclear Society, Toronto, Sept. CONF-650601.


Recommended cross sections for \( ^{239} \text{Pu} \) and \( ^{241} \text{Pu} \) 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)


From IAEA Conference on Nuclear Data, Paris.

The energy dependence of the fission cross section \( \sigma(E) \) usually shows a complex structure in odd fissionable nuclei near the threshold. Characteristics of the lower fission channels can be derived by comparing the observed fission cross section \( \sigma \) 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 succession of lower channels of the transitory \( ^{237} \text{Th} \) nucleus, which were excited in the \( ^{237} \text{Th} \) (n,f) reaction by neutrons having an energy \( E_n < 1.6 \text{ MeV} \): \( \frac{1}{2} ^{1} \), \( \frac{1}{2} ^{2} \), and \( 
\frac{1}{2} ^{3} \). The new channels with \( K = \frac{3}{2} \) explain the inflexion of \( \sigma \) in the region of neutron energy \( E_n = 1.1 \text{ MeV} \). (TTT)


The available experimental data for the value of \( \nu \) for fissile and fertile isotopes are reviewed. The absolute determination of \( \nu \) 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 \( ^{235} \text{U}, ^{239} \text{Pu} \), and \( ^{241} \text{Pu} \). The available data relating the dependence of \( \nu \) on incident neutron energy are tabulated, and straightline fits to the data are made by the method of least squares. These results are of direct value in reactor calculations and the related evaluation of nuclear cross sections. Although there are instances where the data indicate structure of a more complicated nature, no attempt is made to analyze these situations. For practical purposes, a reasonably satisfactory fit to the data can be made with not more than two straight lines. (auth) (UK)


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Neutron Cross Sections:
3. Wide Ranges in Energy

3. Wide Ranges in Energy


The capture cross section of $^{239}$U was measured absolutely at a neutron energy of 30 keV using kinematically collimated neutrons from the $^7$Li$(p,n)^7$Be reaction near threshold. Activation techniques were used to determine both the number of capture events and the number of neutrons that occurred during the irradiation. The result of the $^{239}$U capture cross section measurement is $479 \pm 14$ mb at 30 keV. In addition, the shape of the $^{239}$U capture cross section was measured for neutron energies from 25 to 500 keV using neutrons from the $^7$Li$(p,n)^7$Be reaction. The capture reactions in the $^{239}$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 $^{239}$U capture cross section as given by other authors. (auth)


Relative fission cross sections of $\sigma_f/\sigma_t$ ($^{238}$U to $^{239}$U) and of $\sigma_f/\sigma_{\text{Pu}}$ ($^{239}$Pu to $^{238}$U) were determined over a neutron energy $E$ of 0.3 to 2.5 MeV at a relative accuracy of 1 to 2% by measuring the number of fissions in a double ionization chamber containing layers of the isotopes which were to be compared. Since the fission cross section for the $^{238}$U isotope is well known with high accuracy at $E_0$ = 0.7 MeV, a number of measurements at various neutron energies were made to derive more accurate data on the cross sections of $^{238}$U and $^{239}$Pu. The results were compared with the data compiled by Davey, and it was found that the two sets of data were in good agreement at $E_0 < 0.7$ MeV, but deviated from each other by 7 to 10% at higher values of $E_0$. The results are in good agreement with the data of Lappinere. (TTT)
1968


From IAEA Conference on Nuclear Data, Paris. See STI/PUB-140(Vol.2); CONF-661014-(Vol.2).

The need and requirements for cross-section evaluations is discussed; and the evaluations that were available on June 1, 1968, 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 $^{121}$In 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 $\eta(\text{Pu-239})/\eta(\text{U-235})$ and $\eta(\text{Pu-241})/\eta(\text{U-235})$ were determined from reactivity measurements in ARMF-I and ARMF-II. Results for 2000 m/sec and Maxwellian average values are given. (auth)


The Doppler effect of an enriched (93.2%) $\text{UO}_2$ target in a 1/2" 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 $\sigma$, 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, $\sigma$, 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 U-233 and U-235. Two capsules containing a known amount of highly enriched U-235 and four capsules containing a known amount of highly enriched U-233 were irradiated in the MTR and subsequently analyzed for total and isotopic uranium concentration. Based on these data, tentative effective $\sigma$ values (at a Cd ratio for Co = 15) and the computed 2000 m/sec values are: 0.0997 ± 0.0016 and 0.0984 ± 0.0016 for U-233, and 0.1768 ± 0.0015 and 0.1716 ± 0.0015 for U-235, respectively. (auth)


The graded exposure of 4 Pu-Al alloy, 19 rod clusters of fuel elements, and the subsequent destructive sampling of the elements, and the subsequent destructive sampling of the elements have provided experimental data showing the variation of Pu isotopes with irradiation. Irradiations were conducted in the heavy-water-moderated and -cooled Plutonium Recycle 1st Reactor. Using $^{135}$Cs as a fission indicator, the depletion of the initial Pu to 50.4 ± 1.1% is determined. Reactor effective cross-section ratios for the Pu isotopes are derived from the data, and results show that the capture-to-fission cross-section ratio for $^{239}$Pu is 0.426 ± 0.019. (auth)
An experimental method for the determination of the spectral average of the capture-to-fission ratio \( \alpha \) for materials inserted in a low-flux reactor is described. The procedure involves a comparison of reactor response to oscillated samples of a fissile material, an absorber, and a spontaneous fission neutron source, plus an experimental determination of fission rate for the fissile material and capture rate for the absorber. In addition, it is necessary that the neutron source be calibrated. These experimental results, combined with a knowledge of the number of neutrons per fission for the fissile material, yield a value of the quantity \( 1 + \alpha \). This method has been tested in Hi-C Core 10, a critical assembly of \( {\text{\textsuperscript{235}}} \)-enriched-\( \text{U}_2 \) fuel pins, moderated and reflected by light water, in a lattice spacing which yields a \( \text{H} \)-to-\( \text{\textsuperscript{235}} \)U atom ratio of \( 2:51 \). The oscillator and absolute counting data yield a value of \( 0.217 \) for the central capture-to-fission ratio of \( \text{\textsuperscript{235}} \)U, with a standard deviation of \( \pm 0.015 \). This agrees well with values derived from a combination of measured \( \text{\textsuperscript{235}} \)U fission cadmium ratios and calculated thermal and epithermal values for \( \alpha \). (auth)

**1967**


**1968**


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


From IAEA Conference on Nuclear Data, Paris. See STI/PUB-140(Vol.2); CONF-661014-(Vol.2).

Eta values and total and fission cross sections were measured using the Harwell 45-MeV electron linear accelerator neutron time-of-flight spectrometer. The 3.49-m flight path, with a resolution of 7.2 ns/m, was used for the energy range 10 eV to 1 keV; and the 97.5-m flight path, with a resolution of 2.5 ns/m, was used for the energy range from 50 eV to 30 keV. The experimental methods employed were similar to those of Brooks et al., in which the fission neutron yield and the transmission of a number of samples of different thicknesses are measured together with the shape of the incident neutron spectrum, all the measurements being made with the same resolution. In the yield measurements, fast neutrons from fission events were detected in liquid scintillators, and pulse-shape discrimination was used to reject events caused by gamma rays. The transmission measurements were performed with a lithium glass scintillator, and a 1/V detector was used to measure the spectrum. Backgrounds were measured using the "black resonance" technique. The measurements at 34.3 m were normalized to the 34.9-m measurements in the region of 50 eV.

Results were corrected for multiple scattering in the samples.

Capture cross sections were derived from the total and fission cross sections by assuming values for the scattering cross sections. A direct measurement of alpha (the ratio of the capture to fission cross sections) is in progress over the energy range 10 eV to 30 keV. This will lead to a better estimate of the capture cross sections since alpha measurements are not so sensitive to the magnitude of the scattering cross sections.


The neutron fission cross sections and capture-to-fission ratios of $^{238}$Pu and $^{239}$Pu were measured in the neutron energy ranges 5 eV to 23 keV and 0.15 eV to 30 keV, respectively, using the time-of-flight method. Fission events were registered by delayed coincidence between prompt gamma-ray pulses and slowed fission neutrons. The counting rate of capture events was determined by the anticoincidence. The capture-to-fission ratios and the fission cross sections are presented as functions of energy. Values of the capture and fission resonance integrals are tabulated. (D.C.W.)


An analysis of the feasibility of integral measurements of $\gamma$ for $^{238}$Pu in the Physical Constants Test Reactor (PCTR) as a function of the carbon-to-uranium ratio in plutonium oxide-uranium oxide-carbon mixtures is summarized. The experiments proposed will use 8 to 14% PuO$_2$ of low $^{239}$Pu in UO$_2$ mixed with graphite. It is concluded that integral values of $\gamma$ for $^{238}$Pu can be measured in the PCTR with an uncertainty less than 0.05. (D.C.W.)
A special liquid scintillator detector has been developed for the purpose of measuring alpha(E) for 235Pu in the energy range 10 eV to 30 keV using the Harwell Linear Accelerator time-of-flight spectrometer. Alpha(E) is the ratio of capture to fission cross sections as a function of incident neutron energy. The detector has two outputs, one responding to gamma-ray interactions and the other to fast neutrons. The efficiency of the detector for gamma rays is arranged to be proportional to the gamma-ray energy. This property is achieved by utilizing an improved Moxon-tube 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)


Instrumentation for measurement of u (the ratio of the neutron capture cross section to the fission cross section) was designed, assembled, and calibrated, and measurements were made for 239Pu, 238U, and 235U 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 5.5 x 10^{-9} sec under operating conditions. The energy resolution at the 2.5-MeV sum peak of 60Co was 35%. Neutron energies were measured by time-of-flight techniques using a flight path of 1 meter. Fission events were distinguished from capture events by the detection of thermalized fission neutrons following the primary events. Measurement of the relative efficiency of the large scintillator tank for detecting capture and fission events was made by simultaneously compiling the background and foreground pulse-height spectra for both the capture and fission events. Typical spectra for the neutron time of flight for capture and fission events along with the background and foreground spectra or both types of events are shown. The techniques and adjustments used to obtain the timing resolution on the large tank are discussed. The logic necessary for distinguishing fission events and the foreground and background pulse-height spectra associated with both types of events are described. (auth)
V. NEUTRON CROSS SECTIONS

5. Resonances

1967


From IAEA Conference on Nuclear Data, Paris.

The results are presented of measurements on the radiative capture of neutrons in the separated silver isotopes $^{107}$Ag and $^{109}$Ag in an energy range up to 1000 eV. A number of previously unreported levels were found. It is shown that the marked discrepancy between the level-spacing distribution previously reported for Ag and the Wigner distribution for various superposed level systems is only apparent. It is also shown that there is no correlation between $^{107}$Ag and $^{109}$Ag levels within the limits of statistical accuracy obtained. The values of the strength functions $S_p$ for $^{107}$Ag and $^{109}$Ag are found to be $0.43 \times 10^{-4}$ and $0.83 \times 10^{-4}$, 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)


Resonance integral calculations are done for $^{232}$Th infinite dilute, $^{107}$Ag metal rod, and $^{232}$ThO$_2$ rod systems. Doppler effect calculations are performed for $^{232}$ThO$_2$ rod systems for temperatures up to 2000 K. The resolved resonance integral for rod systems at each temperature is evaluated by Monte Carlo calculations and the resonance overlap effect between the two resonances of $^{232}$Th at 21.78 and 23.45 eV is taken into account. The unresolved s- and p-wave contributions were computed by standard methods. The data describing the resolved resonance parameters up to 3 keV ($\lambda \sim 25.0$ 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 $1.83 \times 10^{-4}$ (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)


From IAEA Conference on Nuclear Data, Paris, France.

Transmission measurements of the neutron total cross sections of Fe, $^{10}$Fe, $^{11}$Mn, and $^{11}$V at 20 to 200 keV were made using the time-of-flight method. A multilevel analysis of the data for $^{11}$Mn and $^{11}$V yielded resonance parameters for these isotopes, and an analysis of the data on Fe yielded resonance parameters for $^{11}$Fe. (D.C.W.)


The intensities of resonance averaged $\gamma$ 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 70 eV, all the 20 resonances known to exist below 370 eV in the total cross section have a measurable fission width with an average (r) = 0.008 MeV. The second and third groups are centered at 8.33 keV and 13.8 keV and extend over about 1 keV. The area under the fission cross section curve $\sigma_f$ is 97.7b eV for the 29 resonances below 740 eV, 52 b eV for the group at 8.33 keV and 32 b eV for the group at 13.8 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. This evidence for a grouping of sub-threshold fission resonances is similar to that already found in $^{239}$Pu and $^{237}$Np and according to Weigmann and Lynn is a result of the existence of the second minimum in the fission potential barrier as predicted by Strutinsky. The well depth corresponding to a level density of 7 keV indicates that the second minimum lies at 3 MeV above the ground state of $^{234}$U. (auth)

39537 (RPI-328-123, pp 1-17) Neutron Cross Sections.
(Rensselaer Polytechnic Inst., Troy, N. Y.).

The average capture cross sections of $\text{W, } ^{238}$W, $^{237}$W, and $^{235}$W in the energy range from 1 to 100 keV were determined. The $p$-wave strength functions for $^{54}$Fe and $^{95}$Ni were extracted from capture data, and the radiative width of the 4.6-keV resonance in $^{42}$Ni was determined. The total cross section of $^{235}$U was determined by transmission measurements on four $^{235}$U samples over the energy range from 0.008 to 200 eV; neutron resonance parameters were obtained for energies up to 129 eV. (D.C.W.)


Neutron radiative capture in natural Mo was studied in the neutron energy range from 10 eV to 25 keV. An area analysis was applied to part of the observed resonances which yields $g_\alpha$. For small resonances or $I_\gamma$ large resonances. Above 1 keV neutron energy, the observed capture rate was averaged to give the mean capture cross section which is in good agreement with earlier measurements. (auth)
V. NEUTRON CROSS SECTIONS

6. Doppler Effects

1967


A series of experiments has been carried out in a fast-neutron spectrum, characterized by a median fission energy of 62 keV, in order to measure the Doppler coefficient and other related temperature effects for a variety of materials which are of particular interest in fast power reactor technology. Special emphasis has been placed on the 235U isotope which has been investigated in several chemical forms amenable to utilization in high-temperature, high-efficiency fuel elements. Changes in the size and chemical composition of samples of this isotope and other heavy-element isotopes have been made in order to evaluate the effects on the Doppler coefficient of changes in the surface-to-mass ratio and changes induced by the addition of C or O to form carbides and oxides. In addition, the effects of localized spectrum perturbation on the Doppler coefficient of Th have been studied by surrounding the sample with "blankets" consisting of heavy resonance absorbers, structural materials, and several types of scatterers, including Na. (auth)


The results of Doppler experiments in ZPR-3(Adab) are presented. The experiments were done on two cores: one which served as a fixed mockup 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.)

6. Doppler Effects

The Doppler effect was measured in 235U capture and 239Pu fission in a fast neutron spectrum. Experimental results were obtained for two 239Pu foil thicknesses and one 235U 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 foil activity with temperature/room temperature foil activity), \( R-1 \), was a factor of 2 higher than that calculated using a neutron energy spectrum derived from “nominal” material cross sections. Presently available cross sections in the energy range of interest are sufficiently uncertain so that it is possible to infer from them “hard” or “soft” neutron energy spectra such that the value of \( R-1 \) varies by a factor of two. The measured values for 235U agreed quantitatively with those found from the “soft” neutron energy spectrum. Within the precision of the measurement no 239Pu Doppler effect was observed. The calculated 239Pu Doppler effect was smaller than the sensitivity of the experiment, thus within its precision (±0.002) the measurement confirms the theory. (Disser. Abstr.)

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**4. Theoretical Analysis**

The theoretical analysis of the Doppler effect in 235U capture and 239Pu fission in a fast neutron spectrum is based on the neutron energy spectrum derived from the U-235 and Pu-239 cross sections. The calculated Doppler effect is expressed as the ratio of the change in foil activity with temperature to the room temperature foil activity, \( R-1 \). The experimental results were compared with the calculated values for various conditions, such as different foil thicknesses and temperatures. The agreement between the calculated and experimental results within the experimental uncertainties indicates the validity of the theoretical model. (Disser. Abstr.)
1967


The measurements reported were designed to study various aspects of the Doppler effect in a spectrum relevant to a fast power reactor. The method used was based on measuring the reactivity changes resulting from the in-pile heating of small samples in a zero-power assembly. After proof-tests of various kinds, the measurements concentrated on the $^{239}$U Doppler effect as a function of sample size and chemical form; the $^{238}$U Doppler effect, with particular attention paid to evaluation of the extraneous effects of thermal expansion; and the Doppler effect of mixtures of $^{239}$U and $^{235}$U. A consistent intercomparison of the most pertinent results, and a comparison of the results with a consistent set of calculations using current theory are presented. A list of 11 references is included. (M.L.S.)


1968

7920 (ANL-Trans-540) INVESTIGATION OF THE DOPPLER EFFECT IN $^{238}$U IN A SHIELD MADE OF URANIUM OXIDE. TULL, C. E.; Lewis, R. A.; Pond, R. B. (Argonne National Lab., Ill.).


The measurements reported were designed to study various aspects of the Doppler effect in a spectrum relevant to a fast power reactor. The method used was based on measuring the reactivity changes resulting from the in-pile heating of small samples in a zero-power assembly. After proof-tests of various kinds, the measurements concentrated on the $^{239}$U Doppler effect as a function of sample size and chemical form; the $^{238}$U Doppler effect, with particular attention paid to evaluation of the extraneous effects of thermal expansion; and the Doppler effect of mixtures of $^{239}$U and $^{235}$U. A consistent intercomparison of the most pertinent results, and a comparison of the results with a consistent set of calculations using current theory are presented. A list of 11 references is included. (M.L.S.)


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

Measurements have been carried out on the Doppler effect in $^{238}$U heated to 750°C in a large block of depleted $^{235}$U oxide $^{238}$U. The Doppler effect is measured by inserting small samples of uranium dioxide samples in a temperature range of 200 to 2000°C. The effect of the sample dimensions on the measurement results was established, as also was the effect of surrounding the sample by a layer of scattering material (heater material, container material, etc.). The accuracy of the results obtained was 5 to 6%. It is shown that in the temperature dependence the proportional increase in the $^{238}$U(n,y) reaction rate is close to the asymptotic spectrum of the medium being studied. The reaction rate for the cold sample irradiated at the same position in the reactor, gamma rays energy 74 keV accompanied by beta decay of $^{238}$U was measured. Measurements were carried out in the temperature range 300 to 2000°C. The error in calculating the Doppler effect is less than the error in the experimental data. (auth)


Measurements of the Doppler effect on the $^{198}$U(n,y) reaction were made on uranium metal samples, heated to 770°C, with neutrons of different energies in the range 0 to 54 keV from the $^{198}$U(n,y) reaction. The measurements made in one particular energy range (0 to 10 keV) were extended to temperatures up to 1960°C using uranium dioxide samples. The results obtained were compared with computer calculations. In the uranium metal samples the Doppler effect at different neutron energies was found to agree, within the rather large experimental errors, with the theoretical calculations. In the uranium oxide samples the measured Doppler effect over the temperature range 290 to 1960°C agreed with that calculated, except for the first 200°C of the range where the measured effect was greater than that calculated. (auth) (UK)


From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.
Neutron Cross Sections:
6. Doppler Effects


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. These samples were heated to different temperatures for this purpose, and the capture γ-rays were detected by proportional counters. Because hot-to-cold capture ratios are measured a knowledge of the neutron flux is not necessary, therefore a direct comparison of calculated and measured temperature-dependent cross-sections is possible. A theoretical analysis of the experimental data for uranium is given. (auth)


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


Measurements of the Doppler effect in $^{238}$U capture and $^{235}$U fission were made by activation technique in three different neutron spectra in the fast critical assembly FR-O. The experiments involved irradiation of thin uranium metal foils or $^{235}$U disks, which were heated in a small oven placed in the core center. The measurements on $^{238}$U were extended to 1780°K and on $^{235}$U to 1470°K. A core region surrounding the oven was homogenized to facilitate the interpretation of results. The reaction rates in the uranium samples were detected by gamma counting. The experimental method was checked with regard to systematic errors by irradiations to a thermal spectrum. The data obtained for $^{238}$U capture were corrected for the effect of neutron collisions in the oven wall, and were extrapolated to zero sample thickness. In the softest spectrum (core 3) a Doppler effect (relative increase in capture rate) of 0.256 ± 0.018 was obtained on heating from 343 to 1780°K, and in the hardest spectrum (core 3) the corresponding value was 0.050 ± 0.003. An appreciable Doppler effect in $^{235}$U fission was obtained only in the softest spectrum, in which the measured increase in fission rate on heating from 320 to 1470°K was 0.007 ± 0.003. (auth) (Sweden)


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

The Doppler broadening of the $^{238}$U capture and the $^{235}$U fission cross-sections in fast reactor spectra has been studied by irradiating heated foils or plates of the isotopes of interest in a small furnace at the center of the zero-energy fast reactor FR-O. 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 $\gamma$-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 FR-O assemblies. The neutron spectra of the cores were broadly similar to those of current steam-cooled fast reactor concepts. The experiments include a study of the spatial distribution of the reactivity coefficient of polythene. The results are in reasonable agreement with calculated values. The latter are sensitive to small changes in the absorption cross-section in the low neutron energy range. Most of the calculations have been made with a one-dimensional diffusion theory program using 16 energy groups but a two-dimensional code and a transport theory code have also been used. Additional measurements have been made on vertical polythene rods, 2.2 and 4.6 cm thick, inserted in the central fuel element. The measured distributions of the fission rate in $^{235}$U and the capture rates in $^{135}$Mn, $^{115}$In and $^{197}$Au inside and around the rods have been compared with results of multigroup calculations. (auth)
VI. LABORATORY SUMMARY REPORTS AND MISCELLANEOUS

1967


Summaries of progress are presented concerning the EBR-II operation and development, physics experiments in ZPR-3 and ZPHT, burnup measurements for fast reactors, Na technology, fuel development and processing, reactor physics development, fuels and cladding development and fabrication, heat transfer and fluid flow studies, mechanics of materials studies, fluoride volatility process development, liquid metal direct conversion generator research, AARR design and development, and research on nuclear safety. (J.R.D.)


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 boron 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 ZPR were completed. The floor slab and foundation for the vault-workroom and service floor of the support wing were poured. Detailed reports of measurements made in ZPR-3 during the last four months on Assembly 48, a large, clean, plutonium-fueled core, were compiled and a summary is presented. The AARR Title I report was submitted to the AEC for approval on October 7, 1966. The AARR core design was changed to a HFIR type, and all research and development work on the stainless steel cermets 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.)


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.)
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-worth type for use in making reactivity adjustments in the reactor are also reported. Experiments were continued in ZPR-3 Assembly 48, a Pu-fueled critical assembly with a distribution of materials similar to a large carbide reactor. This assembly is being used to make measurements relevant to the FFTF design. Heterogeneity measurements were made using thin $^{235}$U, $^{239}$Pu, and manganese foils in several configurations. Relative $^{235}$U capture rate was measured at seven positions near the core midplane, using a method developed at ZPR-3. Excavation of the ZPPR reactor pit is complete. The six reactor pit posts have been steel capped and the concrete tops for 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 present 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.)

Regular and special fuel pins were removed from the EBWR for analysis of specific isotopes. The fuel in the plutonium zone had an average exposure of 1300 MWD/tonne. A detailed study to determine swelling in the uranium and correlation of this to exposure time and temperature was made. Neutron flux wire irradiation was conducted at low powers to obtain distribution rates of fission and plutonium production in the radial blanket. Radial blanket subassemblies of the FFtF and ZPPR reactors were examined for external and internal change in physical properties. Foil irradiations were used to measure the relative $^{239}$U capture and fission rates and $^{235}$U fission rates in ZPR-3. Further studies were also made to determine the effect of the distortion of the flux distribution on the linearity of $k_{eq}$ subcritical measurements in reactors with a softened neutron spectrum. In ZPR-6 the effects of Na voiding on the fission densities in $^{239}$U and $^{235}$U and the capture density of $^{238}$U were investigated by foil measurements. Material sample worths and Doppler coefficients were measured for $^{235}$U and $^{239}$Pu in the ZPR-9; these results are listed in tabular form. Construction activities are detailed for the ZPPR. Three systems of fission-gas pressure transducers for the FFTF are discussed. Failed fuel detection methods, sensor leads for fluid signals, in-core flowmeters, and fuel-pins are some component systems which are briefly discussed. Irradiation of U-Pu alloy fuels is discussed; ceramic, cermet, and mixed-carbide fuel irradiation is also discussed. A method for the preparation of (U,Pu) C fuel is given. Development of refractory-metal alloys for service in oxygen contaminated systems is detailed; corrosion of jacket and structural materials by Na in the reactor is examined and results presented. Microhardness tests were made on Ni-base alloys; results are presented. A relatively detailed outline for fast reactor fuels is given. In the area of general research and development the following topics are discussed: fast reactor core-parameter study, uses of $^{235}$U in a fast-reactor era, neutronic investigations, thermal design studies, and fuel cycle costs. An oscilloscope display for an on-line computer is described; changes in the cross section computer programs are given; reactor parameter calculations are discussed in some detail; fuels and materials development are discussed at length; radiation damage to structural materials is outlined; techniques for fabrication and testing of fuels are presented. Various phases of engineering development are discussed in some detail. Results of corrosion studies for the Argonne Advanced Research Reactor secondary coolant loop are tabulated. A general discussion of the reactor physics measurements and experiments in the AARR is given. A survey of work done in nuclear safety is given; the following subjects are covered: linkage of heat transfer and two-phase coolant flow nodules, Na expulsion, superheat, critical flow, electron-bombardment heater tests, convective instability, thermophysical properties of Na, component dynamics, transfer in pile loop experiment with EBWR-II Mark I Fuels, summary and analysis of single-pin results, seven-pin-cluster Na loop experiments, vacuum-inert gas glowbox. TREAT operations, metalwater reactions, and Pu volatility safety. (M.L.S.)
The EBWR was operated at the maximum permissible power level of 70 MW. During the approach to power, pile-oscillator measurements of the reactor transfer function were made at 28, 38, 49 and 57 MW. In preparation for Run No. 25, several zero-power reactor runs were made to obtain measurements of reactivity worth of the stainless steel subassemblies in Rows 7 and 8, which have replaced subassemblies of depleted U. An extensive program of analysis was initiated to determine the sources and extent of Cu found in the reactor primary Na system. Removal of the Cu electrodes associated with the auxiliary primary pump showed that the exposed Cu ends were severely pitted and eroded and had indeed lost Cu to the primary Na. The exposed Cu ends of the electrodes were sheathed in stainless steel and the electrodes were reinstalled. The pump was checked out and is operating satisfactorily. Analysis for further sources of Cu in the primary system is continuing and results are being evaluated.

Postirradiation examination of radial-blanket subassemblies from the 7th, 8th, 9th, and 12th rows of the reactor revealed that length, diametral, and density increases were greatest for the innermost rows, decreasing along the radius from the core, with no significant increases noted for the 9th or 12th row subassemblies. Concrete for the ZPPR cell floor and pit and for the blanket storage room floor was poured. Experiments are in progress with ZPR-3 Assembly 46B, a reactor with a two-zoned Pu core and with a 12 x 15-in. central region that contains Pu with 22% $^{235}$Pu substituted for 4.5% $^{239}$Pu. Critical mass was determined after control-rod calibrations were made as well as measurements of the worth of core-edge material and of fuel spiking of the safety rods. In the high-Pu-content central region, measurements were made of Na substitutions, fission ratios, perturbation reactivity measurements with small samples, and of fine flux variations across the cell. $^{233}$U, $^{234}$U, $^{235}$U, $^{236}$U, $^{238}$U, and $^{239}$Pu. Reaction rate traverses were made in a radial direction at the core midplane and reactivity traverses have been made in the radial direction using stainless steel, Ta, $^{135}$Ba, $^{237}$Ta, $^{239}$Pu, and $^{241}$Pu. Work done to define more narrowly the limits of precision of ZPR-3 experiments by refinement of measurements related to the ZPR-3 gap interface is reported. (J.R.D.)
1967


Brief descriptions of work on the following subjects are presented: fission properties and cross section data including fast neutron scattering studies, elastic neutron scattering from elements of intermediate weight, elastic neutron scattering from Li and Si, fast neutron scattering from nuclei in the mass region A - 95-130, the interaction of fast neutrons with the 162, 184, and 186 isotopes of W, a search for fluctuations in the fission cross section of 129U, neutron flux measurements in the 10-200 keV region, (n,d) stripping reactions, fast neutron total cross sections using a monoenergetic source and an automatic facility, fast neutron total cross sections using a monoenergetic source and an automatic facility, fast neutron total cross sections using a monoenergetic source and an automatic facility, neutron energy degradation through the fission process, unitary models of nuclear resonance reactions, the 233U fission neutron spectrum, the 235U fission neutron spectrum, the 239Pu fission neutron spectrum from 0.003-15.0 MeV, direct and absolute measurements of average yield of neutrons in the thermal fission of 235U and spontaneous fission of 239U, spontaneous fission half-lives of 129I and 129Xe, thermal reactor physics including Hi-C neutron conversion rate, critical experiment, Hi-C uniform lattice calculations, critical calculations of the EBWR Pu recyle program, measurement of capture-to-fission ratios of 129Pu and 129Pu 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 (ARR) 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--analyses of hypothetical accidents; fast reactor physics including the neutron energy spectrum in a dilute UC-fueled fast critical assembly, neutron spectra in a depleted U, calculations of Ne-void coefficients in large fast neutron carbide cores in assemblies No. 2 and 3 of ZPR-6, calculations of the effect of this slab heterogeneity on the non-leakage reactivity component of Ne voiding, non-linearity in the spectral component of Ne void effect as a function of Ne content, effect of parameter uncertainties on Ne void effect and critical size of fast reactors, Doppler-effect measurements on a dilute carbide fast assembly--2-PZ-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-6 assembly No. 47, standard deviation of ion chamber current measurements in ZPR-6 assembly 47, measured reactivity removal rates in ZPR-6 assembly No. 42, the Argonne National Laboratory of ZPR-3 assembly No. 44, critical assembly comparison calculations using new cross section data, comparative neutron characteristics of metal, oxide and carbide EBR-II driver fuel, effect of fuel and blanket changes on the LIR-II flux, PARET Core I fuel irradiation program and reference design, twenty-six group cross section set for W rocket reactor systems, physics measurements in fast W rocket reactor critical experiments, measurements of space-dependent material worths in several ZPR-9 assemblies, rocket critical assemblies analysis, physics measurements in an operating fast breeder power reactor, further neutronics studies of the 1000 MWe metal-fueled fast breeder reactor, reactor physics calculations for a 10,000 MWe metal-fueled fast breeder reactor, fast breeder reactors for water desalting, criteria for the density of monitoring points in large reactors; fast reactor safety including capabilities of the present TREAT facility core as a fast flux loop meltdown facility, meltdown experiments using the Mark I integral Na loop, analyses of single pin loop meltdown experiments, properties of irradiated UO2 pins prior to TREAT facility transients, photogaphic fast reactor safety experiments on irradiated oxide in the TREAT facility, transients--initial plume on UO2-W cermet rocket fuel samples, design of the Mark II integral TREAT facility Na loop, calibration mockup for the large loop test section for the TREAT facility, transient response of stand-off pressure transducer assemblies on the TREAT facility integral Na loops, extensible multi-purpose vacuum glove box, experimental results and improvements in the fast neutron hodoscope, the exact three-dimensional solution for thermoelastic stresses and displacements in finite and infinite tubes, transient vaporization of Na in reactor coolant channels, convective heat or mass transfer with phase changes, theoretical prediction of thermodynamic and transport properties of metal vapors, equation of state of reactor materials at high pressures and temperatures, a modified equation of state for hydrodynamic calculations in the AX-1 numerical program, properties of refractory ceramics at extremely high temperatures (UC liquid expansion), modification of the high temperature W filament furnace, failure pressures of thick-walled doubly-reinforced concrete cells, maximum permissible body burdens of Pu isotopes and resulting release criteria, fast reactor meltdown accident analysis code, PREX; experimental physics techniques and facilities including a low geometry counting chamber, absolute determination of fission rates in 235U and 238U and capture rates in 238U by radiochemical techniques, precision fission rate measurements by fission track 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 235U 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 Dn 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 inelastic neutron scattering, a multi-angle fast neutron time-of-flight system, multiple angle detector apparatus for neutron elastic scattering and polarisation measurements, multiple scattering correction, automated computer control of a fast neutron laboratory; reactor computation methods and theory including the Argonne Reactor Computation (ARC) system, the ARC system glossary, the Multigroup Constants Code (MC1),

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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, quasistatic treatment of space dependent reactor transients, space dependent kinetic calculations using the WIGLE code, reactor systems analysis and hybrid computers, computation of the coupled error function by continued fractions, treatment of source discontinuities in the solution of the diffusion equation, revision of the bulk shielding code MAC for the CDC-3600 computer, codes for analysis of elastic scattering angular distributions, multilevel cross sections for a fissionable isotope, the effect of interference on the resonance integral mixtures of Th and U, the effect of randomness on group cross sections, the chemical binding effects on the resonance line shapes of $^{238}$U in a UO$_2$ lattice, equivalence between homogeneous and heterogeneous resonance integrals in cylindrical geometry, effect of the fluctuations in collision density on fast reactor Doppler effect calculations, an approximate calculation of space dependent flux using a variational principle, neutron-wave analysis; miscellaneous including energy spectrum of fast cosmic-ray neutrons near sea level, a UO$_2$ system for direct conversion of nuclear energy to coherent laser light, theory of plasma oscillations—generation of thermionic RF energy and interactions with DC, circulating shield reactor for space power, and improvised shatter design for the JANUS reactor.

A total of 609 references is listed throughout the report. (M. L. S.)

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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 PFR, radiation effects on

REACTOR EXPERIMENTAL FACILITIES—design parameters for FFTF; heat transfer analyses for FFTF rabbit capsules

FAST TEST REACTOR—core for, parametric studies on; cooling systems for, analysis of; tube materials for, stress analysis for; core for, hot channel factor determination for; core for, radioactive 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

NUCLIDEs—flux distributions in FFTF, evaluation of postulated effects of skewed

THERMOCOUPLES—time response of ungrounded, analysis of

REACTOR SAFETY—scram requirements for single channel fuel module

REACTOR MODERATORS—effects on FFTF neutron physics parameters of beryllium oxide

URANIUM ISOTOPES U-235—Doppler coefficients for, effects of crystalline binding on

REACTION—worths of control rods for FFTF, two-dimensional transport models for calculation of

REACTIVITY CONTROL ELEMENTS—reactivity worths of FFTF, two-dimensional transport models for calculation of

SHIELDING—calculations for FFTF


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 releases; high temperature and radiation chemistry; electronic structure of metals and alloys; and radiation damage in crystalline solids. (P. C. H.)
1967


Developments are reported for studies on: cross section analysis; integral reactor-physics experiments; basic-theory improvement; fast spectrum Doppler measurements; boiling studies for sodium reactor safety; measurement of Doppler coefficients; reector dynamics SIMS, and Program S-XIII codes. A comparison of the results shows the RBV 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 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 contains 24 wt % Pu. A calculational study of thermalization in PuO₂ - UO₂ fueled 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 with a mixed lattice of Pu-Al fuel and thorium targets in alternating cells. New techniques were developed to adapt the PCTR type of measurement to this lattice array of supercells. The results include values of kₚ 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 saling, neutron flux intensities for various thermal column conditions. The addition of a polyethylene reflector to the graphite stack was found to improve the thermal neutron intensity and to reduce the fast neutron component of the total flux. A small cavity in the center of the thermal column did not appreciably reduce the neutron flux gradient in the standard fuel 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 fissile fuel fission rates are in reasonable agreement with experiment, whereas the computed 235U/239U fission rates are consistently higher than experimental values. A new group cross section "collapse algorithm" has been devised which makes use of a pseudo absorption cross section in each group.


The development of the reactivity technique for determination of effective 235U concentration in samples of borated D-O is described. The results of critical experiments in the PRCF are given. Results from an extensive set of reactivity measurements of neutron absorbing rods in PuO₂ - UO₂ - H₂O lattices are summarized. The worth of the HTI/TR control rods are given. Comparisons of unzoned and two-zoned concepts for the FTR are detailed. Crystalline binding effects on the FTR Doppler coefficient are discussed. FTR fuel heterogeneity effects are examined. Fast reactor investigations for the FTR are presented. Reactivity measurement techniques using pulsed neutrons are discussed. Analyses of critical experiments in hydrogenous media are given. Crystal spectrometer studies of atomic and molecular motions of water at 268K and 299K are presented. Cross section data for low neutron inelastic scattering in H₂O and D₂O are presented. (M.L.S.)
1967


PRESSURE VESSELS—testing of high temperature gas cooled reactor prestressed concrete.

REACTORS, GAS-COOLED—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; fuel cycle for high temperature, economics of.

GAMMA RADIATION—buildup in heterogeneous media of; calculation of flux calculations using transport theory, effects of quadrature on.

IN-PILE LOOPS—fuel element irradiated in GAIL, postirradiated testing of.

REACTOR FUELS—radiation effects on coated particle, tabulation of GAIL IV; cycle economics for high temperature gas-cooled; costs for, development of barrier; divert feeds on.

Uranium Isotopes U-238—tritium absorption as a function of lattice parameters, calculation of.

Fission Products—deposition on stainless steel specimen from GAIL IV fuel element; deposition in stainless steel pipe damaged from GAIL III; burnup in berillium fuel compact of steady-state gascooled.

FI TIB—radiochemical analyses of GAIL IV main loop, results of.

Reactor Fuels Elements—reprocessing of high temperature gas-cooled, effects of fuel element design on; reprocessing of high temperature gas-cooled, economics of.

COATINGS—behavior of silcon carbide reactor fuel, effects of strongly corrosive atmosphere on; fissiion product release from barrier.

Neutron Cross Sections—evaluation of uranium 238.

Plutonium—diffusion studies on.

Graphite—radiation effects at high temperatures on; thermal expansivities of isotopic and anisotropic (M.I.S.)


For Southwest Atomic Energy Associates.

A series of critical experiments were performed on a mockup of SEFOR at ZPR-3. Analyses of these experiments and the application of the results to the SEFOR design are discussed. Critical mass values were determined for 1-, 2-, and 3-segment SEFOR fuel designs to help establish the Pu atom fraction in the SEFOR fuel. Reactivity effects of axial fuel expansion were measured which led to selection of the 2-segment design for SEFOR. Measurements of the reactivity worth of the radial reflector established the adequacy of the SEFOR reflector control system. The Doppler coefficient was measured. The calculated 235U Doppler coefficient was in agreement with the experiment; the measured 233U 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 (nG) 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. Flux ratios, fission and burnup transverses, 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; fissile densities in, sodium void effects on absolute; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in.

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, sodium void 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; 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; core for ZPR-6 and ZPR-9, measurement of central fission ratio in; core for ZPR-6, measurement of fission neutron density in; design parameters for ZPR-9 assemblies 11 and 12; core for ZPR-6 and ZPR-9, comparison of calculated and experiment flux at centerline of; Doppler effect measurements in ZPR-9 assemblies 11 and 12; core for ZPR-6, neutron spectrum comparisons for zoned and homogeneous fast; sodium void reactivity calculations for ZPR-6, heterogeneity effects on; neutron density in uranium-238; sodium void coefficients for; effect on core calculations of; Doppler reactivity measurements in ZPR-9, calculation of; Doppler reactivity measurements in ZPR-9; pulsed neutron techniques for; neutron decay constant measurements in ZPR-9, use of pulsed neutron technique for prompt; reflectors for fast ZPR-3, savings properties of inconel-sodium; core for ZPR-3, symmetry effects of embedded reflectors in blanket around; reactivity worth of, relation of reciprocal flux to; reactivity worths in ZPR-3, comparison of measured and calculated; plutonium fission rates in, calculation of.

Reactivity—effects of uranium-238 on; fast; material expansion in fast reactors, calculational procedure for determining; measurements in fast ZPR-9 assemblies 13-17; effects of temperature on; measurements in ZPR-6, pulsed neutron techniques for; determinations for fast converter reactors, effects of sodium voiding on.

Uranium Oxides UO3—fuel pellets, temperature effects on; radial expansion of axial constrained.

Uranium Isotopes U-235—expansion effects on; fast ZPR-6 reactivity; fission densities in, sodium void effects on absolute; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3.

Molybdenum—Doppler effect measurements in ZPR-9 assemblies 12-17; reactivity measurements in large uranium carbide fast ZPR-6; reactivity measurements in ZPR-9 assemblies 11 and 12; reactivity worth determination in ZPR-3.
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24925 (ANL-7127) REACTOR DEVELOPMENT PROGRAM

CRITICAL ASSEMBLIES—neutron spectrum in ZPR-3 Assembly 51, central (E); neutron spectrum in ZPR-6 Assembly 8 with and without sodium, central, (E); 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-7131) 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
URACITY — 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

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37401 (ANL-7457) REACTOR DEVELOPMENT PROGRAM.

This monthly progress report includes information on EBWR, LMFBFR, ZPR-3, ZPR-6, ZPR-9, ZPPR, EBR-II, general reactor technology, and reactor safety including TREAT operations. For detailed information, see the annual reports from the following ANL Divisions: Chemical Engineering, Metallurgy, Reactor Engineering, and Reactor Physics. 25 references. (D.C.C.)

42158 (ANL-7460) REACTOR DEVELOPMENT PROGRAM.

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 MWd in a run with a full complement of fueled experimental subassemblies and a reference run with reflector subassemblies of stainless steel rather than depleted uranium. Nearly normal production was resumed in the Fuel Cycle Facility hot line. Experiments continued with the ZPR-3 Assembly 52, the second core of the FTR Phase-B critical program. Installation of the ZPPR reactor assembly and associated equipment was completed, and seal door and containment structure testing is in progress. Nuclear safety studies, including TREAT operations, and chemical separations activities are discussed. (H.D.R.)

50699 (ANL-7487) REACTOR DEVELOPMENT PROGRAM.

Highlights of project activities are summarized for the month of August 1968. Fuel development programs for the LMFBFR are described. Operations of the ZPR-3 Assembly 51 and 52 for LMFBFR physics developments during August 1968 are described. Loading and startup programs of the ZPPR are presented. Operations for the EBR-2 during August 1968 are described. Nuclear safety studies, including TREAT operations, and chemical separation activities are discussed. (D.C.C.)

35422 (ANL-7445) REACTOR DEVELOPMENT PROGRAM.

This monthly progress report includes information on ERR-II, ZPR-3, ZPPR, LMFBFR, general reactor technology, and reactor safety. For detailed information, see the annual reports from the following ANL Divisions: Chemical Engineering, Metallurgy, Reactor Engineering, and Reactor Physics. 13 references. (J.M.J.)
32940 (ANL-7310, pp 31-135) THERMAL REACTOR
PHYSICS. (Argonne National Lab., Ill.)
CRITICAL ASSEMBLIES — cadmium ratio measurements in high conversion uniform lattices, 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
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REACTIVITY — neutron cross section variation effects on, uranium-233 and uranium-238; temperature effects on AARR reactor; insertion accidents in AARR, analysis of; transients in AARR, summary of reactor responses to
EXPERIMENTAL BOILING WATER REACTOR — poisoning for, reactivity worth of boric acid; control rods for, reactivity worth of; temperature coefficient of reactivity for, measurement of uniform plutonium; plutonium recycle experiment in, power operation history of; plutonium fuel loading in, measurement of capture-to-fission ratio of plutonium-239 and plutonium-241; cadmium ratio measurements in, test conditions for
NEUTRON CROSS SECTIONS — variations in uranium-235 and uranium-238, effects on reactivity of; determination of AARR internal thermal column effective group

30155 (NAA-SR-12296) AEC UNCLASSIFIED PROGRAMS,
NEUTRON CROSS SECTIONS — evaluation for fast reactor physics analysis, (E/T)
NEUTRONS — spectra in coolant and structural material of sodium-cooled reactors, (E); spectra distribution in fast reactors, analysis of energy and spatial, (T)
REACTORS, FAST — reactivity variations in, Doppler measurements on
REACTORS, LIQUID METAL COOLED — stability of sodium-cooled, effect of boiling and two-phase flow on, (E/T); cooling system of, behavior of fission products released in sodium, (E); coolant flow in, energy and fission product release in sodium, (E/T)
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 flow in sodium-cooled reactors, energy and fission product release in, (E/T)
REACTIVITY — Doppler coefficient of, method for determination of, (E/T)

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27396 (AEC-12638) AEC UNCLASSIFIED PROGRAMS,
NEUTRON CROSS SECTIONS — SCORE evaluation system for, description of
THORIUM ISOTOPES Th-232 — neutron cross section for, use of optical model for evaluation of
URANIUM ISOTOPES U-238 — neutron cross section for, use of optical model for evaluation of; reactivity worth measurements in fast spectrum reactor
REACTORS, FAST — core for, temperature coefficient of thorium metal
THORIUM — reactivity worth measurements in fast spectrum reactor
PLUTONIUM — Doppler effect of, effects of isotopic composition on
REACTIVITY — worth measurements of thorium and uranium-238 in fast spectrum reactor; temperature coefficient of, effects of plutonium isotopic content on; worth measurement of uranium-235, effects of mass on

27340 (RSW-1647-7) PHYSICS VERIFICATION PROGRAM,
CRITICAL ASSEMBLIES — description of B & W Critical Experiment Laboratory uranium oxide (UO2)-fueled water-moderated multi-rod
CRITICAL STUDIES — reactivity measurements for aluminum-clad uranium oxide (UO2) multi-rod water-moderated, analysis of
REACTOR FUEL ELEMENTS — assemblies of aluminum-clad uranium oxide (UO2), criticality of multi-rod circular
RADIATION DETECTORS, PROPORTIONAL — calibration of B & W Critical Experiment Laboratory multi-rod critical assembly, description of
RADIATION DETECTORS, FISSION CHAMBER, RADIATION DETECTORS, IONIZATION CHAMBER — calibration of B & W Critical Experiment Laboratory multi-rod critical assembly, description of
NEUTRONS — radiation measurement of, effects of concrete shielding thickness on B & W Critical Experiment Laboratory multi-rod critical assembly
REACTIVITY — measurement of aluminum-clad uranium oxide (UO2) multi-rod water-moderated critical assembly, analysis of
RADIATION DETECTORS, ACTIVATION — measurement of neutron flux distribution in aluminum-clad uranium oxide (UO2) multi-rod water-moderated critical assembly, analysis of; design of rhodium wire, description of; performance of rhodium wire, description of; calibration of aluminum-dysprosium wire, description of; performance of aluminum-dysprosium wire, description of
ALUMINUM — use for neutron flux measurement, description of
ALUMINUM ALLOYS AND SYSTEMS — Al-Dy, use for neutron flux measurement, description of
REACTOR CONTROL ELEMENTS — reactivity worth of aluminum-cadmium-indium, measurement of (D.C.C.)
1968


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.)


Measurements to obtain adjoint weighted excess 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 intrinsic medium neutron multiplication factor (k.) was obtained for each lattice. The calculated values agreed with the measured to 1% accuracy. The value of k was compared with the corresponding value obtained from measurements for the same lattice. Analysis of the CaF2-Phoenix water reflected core with 17 energy groups yielded a calculated reactivity within 1% of experiment using diffusion theory and 3% using transport theory. Pre-experiment calculations on the PRCF-Phoenix core revealed that k dropped as more core-reflector interface detail was added. Final critical loading estimates were less than 9% low. Placement of flux suppressors for the MITI-Phoenix core was investigated to avoid having the far-out flux increase outweigh the close-in flux decreases. A full core, banked-shim critical loading was achieved in the PRCF-Phoenix fuel experiment, with the shims 65% withdrawn. Power distribution measurements in the shimmed core have shown the expected power peaking in the fuel followed just below the bottom of the core. Compilations have been issued of the burnup data obtained by destructive analysis of Al-1.8 wt % Pu and Al-2.6 wt % Pu fuels which were irradiated in PRTR. Analysis of the burnup data from the Aluminum-Plutonium fuels irradiated in PRTR was completed. The isotopic concentration data were processed using multivariable regression analysis to obtain a unique set of cross section ratios. An empirical formula was derived for use in fitting critical mass data from cylindrical arrays of rods, moderated with H2O. The formula provides a method of accurate interpolation to obtain the fuel-to-moderator ratio for minimum critical mass. Relative rod power measurements were made in PRTR to determine the power match between fuel rods in elements in the Batch Core and fuel rods in several possible test configurations in the Fuel Element Rupture Test Facility (FERTF). The relative rod powers were determined by gamma scanning. The reactivity changes associated with loss of coolant from the PRCF were also measured for several fuel compositions and test configurations. A theory-experiment correlation study of ratios of effective cross sections was performed. A result of the study is that the calculated ratios agree best with the experimental ratios using the Leonard normalization for the 239Pu and 241Pu thermal cross sections. Calculations of the photonutron production in D2O moderated reactor systems fuelled with 19-rod clusters of Al-Pu, UO2-PuO2, and UC were made. A calculational study was made comparing various correction methods to obtain the effect of resonance integrals and reactivity were also determined. Calculations were made of power peaking 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 241Pu content. (auth)

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12088 (BNWL-472, pp 4.1-43) FAST REACTORS. (Batelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

A survey was performed to investigate the effect of BeO on PTR neutronics. Items considered included the sodium coefficient, Doppler coefficient, critical mass, and flux spectrum dependence on BeO volume fractions. A detailed analysis of PTR 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 PTR 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 PTR 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 caused the Doppler coefficient to diverge from the 1/T dependence in qualitative agreement with the experimental results. The two-dimensional perturbations code, 2-D PERT, is not operational. This code will be used for driver fuel and test management studies. The Phase-A critical experiments in ZPR-III were completed, and analysis of the experiments was started. The attenuation characteristics of a number of alternate fuel arrangements were compared analytically to provide a basis for the conceptual design of the PTR. Estimates of gamma intensity at selected locations within two different closed loop cell concepts were also made. (auth)
1968

14251 (BNWL-534) REACTOR PHYSICS DEPARTMENT
(Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

NEUTRON CROSS SECTIONS—analysis of, use of optical model for
theoretical
COMPUTER PROGRAMS—description of fast reactor FCC-IV.

REACTOR FUEL ELEMENTS—reactivity worth of PRTR, measure-
ment of; power distributions in Phoenix, measurement in
PRCF of; neutron multiplication factors for polonium, com-
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PLUTONIUM RECYCLE TEST REACTOR—neutron fluxes in,
ratios of axial peak-to-average; fuel element recycle test facility
for, measurement of fuel element reactivity worths in
BOHR—reactivity worth of, PRTR hatch core measurement of

REACTIVITIES—lattice parameters for uranium dioxide 19-
cluster, analysis using HBW Monte Carlo code for; fuel lattices
for, temperature coefficient calculations for uranium-233 oxide
(UO2)–thorium-232 oxide

PLUTONIUM RECYCLE CRITICAL FACILITY—reactivity
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HIGH-TEMPERATURE LATTICE TEST REACTOR—Doppler
coefficient measurements for, procedures for

FAST TEST REACTOR—physics characteristics of, analysis of
split-cone; control rods for, reactivity worth of, accident conditions in, fuel pin center-line temperatures un-
der; control rods for, calculations of re-activity worths of;
components for, heat generation rates and neutron damage
rates in

REACTOR SAFETY—FTR fuel pin centerline temperature un-
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CRITICAL ASSEMBLIES—Doppler coefficients for ZPR-3,
comparison of measured and calculated values for; neutron
fission rates in ZPR-3, uranium-233 and plutonium-239;
nutron spectral measurements in ZPR-3 and ZPR-6,

URANIUM ISOPTES U-239—neutron fission rates in ZPR-3
PLUTONIUM ISOPTES Pu-239—neutron fission rates in
ZPR-3

REACTOR CONTROL ELEMENTS—reactivity worth calcula-
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URANYL NITRATE—critical uranium-239 enrichment for so-
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REACTIVITY—worth of uranyl nitrate as function of Uranium-
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PHYSICAL CONSTANTS TEST REACTOR—reactivity worth
measurements in, determination of critical uranium-239 en-
richment for uranyl nitrate, using

NEUTRONS—monochromatization of, use of multiple Bragg
reflection for high-resolution

CHARGED PARTICLES—relativistic motion of, description of
computer program for calculating and plotting eight-
dimensional phase-space electromagnetic plane wave-driven

1967

20779 (BNWL-624, p. 3.1-159) REACTOR PHYSICS.
(Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

Experiments were conducted in the Critical Approach Facility
and the Plutonium Recycle Critical Facility to investigate the de-
tailed physics properties of plutonium-fueled reactor systems.
Measurements have been made using UO2–2 wt % PuO2 rods in
H2O moderator. Experiments were conducted at lattice spacings
covering the range of moderator to PuO2–UO2 volume ratios
of 0.61 to 3.7 and the plutonium contained either 9, 16 or 24% 239Pu.
The results included critical masses, 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 UO2 and for obtaining
additional data on water lattices not obtainable from other fa-
cilities which are required for further evaluation of analysis meth-
ods. 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 % 239Pu). The design of the first experiment is the water
tank is complete and will contain PuO2–UO2 pellets of a 0.1 in.
diameter pitch. A large scale burnup experiment has been initiated
in the DO–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 expansion
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 UO2, and UO2–PuO2 fuels. In addi-
tion, the fuels which contained plutonium differed in their
composition. 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 fission and fuel concentration and isotopic com-
position. From these data effective cross section ratios are
derived in use in evaluating burnup analysis methods. New tec-
hniques for this purpose using multi-parameter, non-linear regres-
sion methods have been developed and applied to data from four
sets of irradiated Al–Pu samples. The PNL Gamma Scanner has
been improved and continues to be used for the nondestructive
analysis. Information on the physics characteristics of pressurized
water power reactor systems which were obtained during the Sax-
ton plutonium program is compared to results obtained with zero,
one, and two-dimensional diffusion theory methods. The conclusion
reached is that a one-dimensional cylindrical model of the reactor
system is adequate from the standpoint of reactivity calculations for
single lattice units. Analytical survey studies have been performed
to provide information on the physics characteristics for various
reactor systems of interest. It is shown that the UO2–4 wt %
PuO2–H2O cores which were studied would be undermoderated
and that the characteristic parameters such as θ, f, and p play an im-
portant role in the physics behavior of these types. A survey study
was conducted to determine the approximate magnitude of various
reactivity coefficients for UO2–PuO2 fueled light water reactors.
Calculations were also performed to determine the reactor character-
stics of thorium loadings in 10% moderator. The feasibility of the
thorium loading in the PRTR was determined either as a batch
core or as a driver region for a UO2–Pu4H, core. The physics
characteristics of metallic fuel and of ceramic fuels were investigat-
eg along with the variation in rod size in thorium enrichment.
Slow neutron inelastic scattering cross sections have been reported
for H2O and D2O. The double differential cross section and corre-
sponding Eqs. 2, scattering law have been obtained. Measurements
for room temperature H2O and D2O and for H2O of five de-
grees below its freezing point using the Battelle Rotating Crystal

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time-of-flight Spectrometer. In addition, results of measure-
ments for \( \kappa_{\text{eff}} \) at \( 25^\circ \)C using the Battelle Triple-Axis Spectrom-
eter have been reported. An extensive evaluation of the RBU Monte
Carlos Code has been completed. Results of this evaluation indi-
cate that the RBU Monte Carlo code is free of gross program er-
rors and can be used reliably for reactor physics calculations.

Knowledge of the Legendre moments of moderator scattering cross
sections is of particular importance in the production of the thermal
neutron spectrum of plutonium fueled reactors. Based on the
Egami-Schroeddel formalism, two methods for calculating scatter-
ing moments for moderators have been developed and programmed
on the computer. A model for water is being evaluated using avail-
able experimental data. Several computer codes have been adapted,
modified, or improved for use in the physics programs. The RBU
Monte Carlo code has been thoroughly tested and appears to be
working satisfactorily. The ZODIAC-2 burnup code was modified
and adhered to increase its burnup capabilities. The HRG (Stan-
ford 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. Adaptation of the ENDF/B system to the local
UNIVAC 1108 is in progress. Theoretical studies have been de-
directed toward developing a mathematical physics model which ac-
curately predicts the observed physics behavior of power reactor
systems. A measure of the validity of the mathematical model is
obtained by comparing measured and calculated integral param-
eters such as reactor multiplication and effective cross sections.
Comparisons have been performed for numerous plutonium and/or
uranium fueled \( H_2O \) lattices. Comparisons of flux and power den-
sities, reactivity coefficients, kinetic parameters, and effective
cross section ratios, have also been made. The results of the
comparisons show that in many cases the calculational techniques
need to be refined. (auth)

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Technical Activities Quarterly Report, December 1967-February
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PLUTONIUM OXIDES \( \text{PuO}_2 \) — \( \text{PuO}_2 \)-\( \text{UO}_2 \) radiation effects on,
analysis of neutron; \( \text{PuO}_2 \)-\( \text{UO}_2 \) fuel loading for PRCF re-
tector; physics measurements for critical, (\( E/T \))

CRITICALITY STUDIES — measurement of plutonium oxide
(\( \text{PuO}_2 \))-uranium oxide \( (\text{UO}_2) \), effects of plutonium enrichment
on, (\( E/T \))

REACTIVITY — measurement of PRCF uranium oxide \( (\text{UO}_2) \)-
field core excess, analysis of; measurement of effects of
boron concentration in water on \( T/T \), analysis of, (\( E/T \))

PLUTONIUM RECYCLE CRITICAL FACILITY — core loading
for, reactivity measurements for uranium oxide \( (\text{UO}_2) \) and
plutonium oxide \( (\text{PuO}_2) \)-uranium oxide \( (\text{UO}_2) \)

URANIUM OXIDES \( \text{UO}_2 \) — fuel loading for PRCF reactor of,
physics measurements for critical, (\( E/T \))
Fabrication of the 260 homogeneous fuel rods containing 2% 235U and 2.7% 239Pu is completed. Fabrication has started on the making of 250 heterogeneous type rods containing 2% 235U and 2.7% 239Pu. 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 swelling of the rods reached several per cent and there is a difference between the homogeneous and heterogeneous rods in composition and volume of gas collected. Two irradiation tests of three Zircaloy 2 clad fuel rods have been effected in hydraulic conveyor at a power rate in the region of 770 W/cm (HR-1 runs 9 and 10). Following the 9 and 10 HR-1 runs, three sets of three stainless steel clad fuel rods were irradiated up to a power of 1250 W/cm. The dimensional measurements effected before and after irradiation showed that the homogeneous rods had swelled by about 1 to 2%, while the heterogeneous rods had retained their initial dimensions. The maximum power attained in CMR-4 is 800 W/cm. The burn-up ratios are 22500 MWD/MT in the homogeneous rod and 18500 MWD/MT in the heterogeneous rod. The PANTHER code has been improved by proceeding with the treatment of the heterogeneity in the fast field, the broadening of the resonance of 239Pu 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 4/1 fuel. The critical mass has been determined, together with the axial and radial buckling. The reactive effects and the power distributions have been measured in the presence of a number of perturbations (water films, aluminum plate, absorber rods). Analysis of the data from the sub-critical VENUS-Vulcan tests has continued. A supplementary program of sub-critical tests on 4/1 fuel with an M tip 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 UO2-PuO2 area with 3.6 wt % of PuO2. (J.C.W.)