The Data of Nuclear Reactor Physics: A Bibliography
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THE DATA OF NUCLEAR REACTOR PHYSICS:
A BIBLIOGRAPHY

INTRODUCTION

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

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

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

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

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

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


   and by Bernard et al.:


   An extensive "Calculations of Neutron Cross Sections Using a Local Optical Potential with Average Parameters" has been published as LA-3530-MS (1966) by Ferne P. Agee and Louis Rosen.

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

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

1955

1. 7927

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

The experimental determinations of $\nu_{\text{eff}}$ in the intermediate neutron energy region show that the prospects of using $^{239}$Pu and $^{235}$U hydrogen-moderated reactors for breeding purposes are poor.  

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

1956

2. 9859 CF-54-2-165
Oak Ridge National Lab., Tenn.

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

3. 4470 CF-1627
Clinton Labs., Oak Ridge, Tenn.

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

4. 3151 WAPD-128
Critical Experiments 5-10

Contract AT-11-1-GEN-14.

Critical experiments with a highly enriched homogeneous reactor have been performed in the range of H:U\(^{235}\) atomic ratios from 1633 to 1776. The cylindrical core has a fixed diameter of 36 in. and heights between 23 and 51 in. An aqueous solution of UO\(_2\)(NO\(_3\)) with a U\(^{235}\) enrichment greater than 90% was used. An 8-in. radial H\(_2\)O reflector was available. Measurements were made of critical heights and of the reactivity worth per inch at these heights as functions of the H:U\(^{235}\) atomic ratio. U\(^{235}\) critical masses are listed. Additional measurements were made to determine the control rod worths, the temperature coefficient of reactivity and the effect of the radial reflector. (auth)

5. 5399  MonP-402
[Clinton Labs., Oak Ridge, Tenn.]  

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

6. 3230  MonP-357
[Clinton Labs., Oak Ridge, Tenn.]  

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

7. 7309  AECD-4044
[Oak Ridge National Lab., Tenn.]  

The experimental facilities and operating procedures which have been used in the conduct of critical experiments to study the characteristics of small enriched reactors have been described in MonP-357. The purpose of the present report is to describe briefly certain improvements that have been made in the experimental facilities and to present the results of further measurements on critical mass and spatial neutron flux distributions in Be-reflected reactors having square and thin slab geometries. The critical mass of a reactor having the usual U-Al-H\(_2\)O core and no reflector was measured and preliminary experiments to determine the fast neutron flux in these reactors were carried out. (auth)


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


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

10. 6452  MonP-48
[Clinton Labs., Oak Ridge, Tenn.]  

Experiments are described on the critical masses of assemblies of fluorinated, hydrogenous mixtures containing 24% enriched uranium. A hydrogenous reflector was used. The critical masses of the uranium as a function of the amount of hydrogen in the mixture under the experimental conditions of density and effective molecular weight are tabulated. The effect of Cd and B shielding between the U mixture and the surrounding reflector was also studied, and an experiment was made on varying the shape from a cubical to an elongated geometry. An attempt is made to interpret the criticality results in terms of pure UF\(_4\), leading to the result that at this isotopic concentration, and a density of 4.5, about 36 kg of UF\(_4\) would be critical if lumped surrounded closely by hydrogenous material. (auth)
11.
3700 CF-51-7-106
Oak Ridge National Lab., Tenn.

1957

12.
13147 K-126

Using uranium enriched with 30% U235 fabricated with fluoroplastic into cubes having nuclear properties of UF6, a study was made of factors affecting the mass assembled at criticality. The effect of intermixed hydrogen, type of reflector, density of U235, homogeneity, and geometric shape are considered. (auth)

13.
13855 K-343

The conditions under which U235 (93.4%), contained in aqueous solutions of UO2F2, becomes critical in cylindrical aluminum and stainless steel reactors were examined. (L.M.T.)

14.
13938 KAPL-M-IB-13
Knolls Atomic Power Lab., Schenectady, N. Y.

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

15.
11006 K-406

Some exploratory experiments are reported describing the conditions under which enriched uranium, contained in aqueous solutions of uranyl fluoride, becomes critical in two right cylindrical reactors having parallel axes. Values of the height of solution and the critical mass have been obtained for reactors with diameters ranging from 5 to 20 inches. Data were obtained at four chemical concentrations corresponding to H:U235 atomic ratios varying between 30 and 330 and with reactor separations up to 50 cm. In some of the experiments, the reactors were submerged in a water bath while others were done with no reflector. The critical mass of an unenclosed two reactor system was found to depend upon the distance between the components when they were separated even as much as 50 cm; however, the mass in each was then more than 90% of that required to make it singly critical. Interclosure of water between two reactors reduced their interaction due to the attenuation of the neutron flux by hydrogen. Two water enclosed reactors which could be made singly critical when approximately equidistant were found to be effectively isolated when separated 15 cm or more. Two which could not be made individually critical were isolated by a separation of a few centimeters. Those which were singly critical at heights large compared to their diameters showed apparent interaction at more than 20 cm spacing. The effect is attributed to the equivalence of a few interaction neutrons in the two component system and the relatively large quantity of uranium which must be placed at the end of long reactors to produce small increases in reactivity. The smallest mass accumulated at criticality in these experiments was 680 gm U235 contained in each of two water enclosed reactors ten inches in diameter with sides in contact, each filled to a height of 15.9 cm. The chemical concentration of the fuel corresponded to an H:U235 atomic ratio of 325. (auth)

Critical Experiments 11-16

16.
11007 K-643

An investigation has been made of the conditions under which aqueous solutions of enriched uranyl nitrate become critical in right cylindrical reactors. A comparison was made of the neutron reflectivity of stainless steel, bismuth subcarbonate both dry and as a water slurry, aqueous solutions of natural uranyl nitrate and phosphoric acid. The materials tested, with the exception of dry bismuth subcarbonate, were about as effective reflectors as water. Dry bismuth subcarbonate was considerably less effective. The free nitrogen content of the nitrate solution and the addition of phosphoric acid and metallic bismuth to the reactor core were among the variables studied. Nitric and
Critical Experiments 17-22

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

17.
10293 UCRL-18980
California, Univ., Livermore. Radiation Lab.
CRITICAL MASS MEASUREMENTS ON GRAPHITE U\textsuperscript{238} SYSTEMS. James E. Carothers. Apr. 11, 1957. 15p. Contract W-7405-eng-48. $3.30(ph OTS); $2.40(mf OTS).

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

18.
1313 AERE-R/R-2051

Critical masses of U\textsuperscript{235} and U\textsuperscript{238} in the form of UO\textsubscript{2}F\textsubscript{2} in aqueous solution have been measured in cylindrical geometry for a range of concentrations H/U\textsubscript{F}\textsubscript{5} = 250 to 850. A cylinder radius of 15.24 cm was used with an effectively infinite radial water reflector. Where possible, estimates were made of the critical masses for the system unreflected and also for the system reflected but with the Cd shut off abeam between the core and reflector. Flux measurements were made through the system in axial and radial directions at all concentrations. The effect of change of temperature on the system over the range 15 to 90°C was observed for all concentrations. An assembly for each isotope in a very concentrated solution H/U\textsubscript{F}\textsubscript{5} = 60, H/U\textsubscript{F}\textsubscript{5} = 32 was also made in a 6.35 cm radius cylinder but was not brought to criticality. (auth)

19.
9865 ORNL-1726
Oak Ridge National Lab., Tenn.

A series of experiments was performed with aqueous slurries of UO\textsubscript{2}, enriched to 93% in U\textsuperscript{235}, as a preliminary study of the nuclear stability of critical and near-critical suspensions. The slurries were contained in a 12" diameter cylinder and were mixed by a centrally located variable speed stirrer in a manner giving small surface disturbances. Critical masses were measured at concentrations ranging from 40 to 200 g U/L at several stirrer speeds.

The dependence of the critical parameters of well mixed slurries on concentration is essentially the same as found earlier with solutions. The critical masses of incompletely mixed slurries were measured to be 5% to 10% less than those of well mixed ones under otherwise similar conditions, an amount not strongly dependent on the chemical concentration. Neutron activity excursions of corresponding magnitude were observed in near critical quantities of the slurry following stopping the stirrer in a well mixed system and, in some instances, after starting agitation of one which had settled. Although the activity changes lagged the alterations of the stirrer speed, in most cases requiring a few tens of seconds to develop, the magnitudes of the excursions were sufficient to warrant consideration of this effect in the safety features of reactors designed to use slurry fuels. The critical mass of a (UO\textsubscript{2})\textsubscript{2} solution in the same vessel increased monotonically as the stirrer speed was increased over the same range, the total change being about 35%, due in part at least, to some vortex formation at the higher speeds. It is concluded that the activity excursions in the slurry result from changes in the U distribution more complex than those which occur in settling alone. An analysis of the distribution, relating it to the neutron activity, has not been made so the results of the experiments are not completely interpretable. (auth)

20.
12211 AECD-4245
Los Alamos Scientific Lab., N. Mex.

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

21.
13911 CF-56-4-29
Oak Ridge National Lab., Tenn.

Combination of critical radius and critical concentration (in grams U\textsuperscript{235}/cm\textsuperscript{3}) are given for a bare reactor having UF\textsubscript{4} dissolved in NaZrF\textsubscript{4}. (L.M.T.)

22.
8689 HW-25614
Hanford Works, Richland, Wash.
CRITICALITY CONDITIONS FOR 1.75 PER CENT EN-
Critical Experiments 23-30

stacking 10.5 in. diameter oralloy and graphite plates. In the relation Oralloy critical mass = constant x (fraction of oralloy/in the core volume)^n values for the exponent, n, in the neighborhood of 0.70 were obtained. (auth)

28.
13166 LAMS-786
[Los Alamos Scientific Lab., N. Mex.]
SAFETY TESTS ON HAND STACKING OF U-235 CUBES.
Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

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

29.
13894 CF-53-1-294
Oak Ridge National Lab., Tenn.

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

30.
12632 CF-54-8-221
Oak Ridge National Lab., Tenn.
A REPORT ON CRITICAL DIMENSIONS OF CYLINDERS.

Curves showing the relationship between the critical heights and critical diameters of stainless steel cylinders having wall thicknesses of 1/4 in. and containing aqueous solutions of UO_{2}F_{2} and enriched U have been plotted for several values of H to U^{235} atomic ratios. A critical size nomograph is presented for simple geometries. If the dimensions of one critical shape are known, then the dimensions for a series of similar shapes using the same assay material may be determined from the nomograph. Criticality relationships are also considered for cylindrical annuli and cylinders with small axial poison rods. (M.F.G.)
Critical Experiments 31-37

31. HW-42463

[Hanford Atomic Products Operation, Richland, Wash.]

NUCLEAR SAFETY OF RIGHT ELLIPTIC AND RIGHT ANNUAL CYLINDERS. Norman Ketlach. June 1, 1956. 9p. $1.80 (ph OTS); $1.90 (mf OTS).

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

32. K-1260

Oak Ridge Gasous Diffusion Plant, Tenn.


A theoretical method of calculating the critical mass of unreflected hydrogenous systems containing U235, based on a spherical harmonics expansion for hydrogen moderation and on age theory slowing down for all other materials, is developed and set up for machine computation on the IBM 607. The calculated critical masses for high assay systems agree with experimental values within 13% for hydrogen to U235 ratio from 40 to 750, and agree within 12% for a single calculation at 4.9% U235 and a hydrogen to U235 ratio of 145. The calculating method appears to yield results which are directly applicable in the evaluation of the nuclear safety of uranium processing equipment. (auth)

33. LA-1653

Los Alamos Scientific Lab., N. Mex.


Neutron detector traverses of the untamped U235 (Godiva) and the U235 or Ni-tamped U235 (Topsy) metal critical assemblies have been obtained by counting γ-activity of U238 and U235 fission products, fission fragments of Np237 in a spiral chamber, and β-activity of Au and S. At a few positions within the assemblies, and for the U235 fission spectrum, cross section ratios of a number of pairs of fissionable isotopes were determined by means of a comparison fission chamber. At the center of Godiva, qf (25)∕cf (235) = 6.2, and this ratio for the Topsy U235 - U235 assembly ranges from 6.8 at the center to 76 at a radius of 8.4'. Results of radiochemical analyses for fission and other reaction products are listed. (auth)

34. WAPD-169


EXPERIMENTAL AND THEORETICAL STUDY OF CRITI-


Data are presented from critical slab experiments with a thin core of enriched U235, H2O, and Zr which were surrounded by reflectors of different compositions and arrangements. The reflectors studied were composed of water, Zr-water, and Al-water. In some experiments the cores were bisection by water gaps. Each critical assembly was analyzed by one-dimensional, group diffusion theory using digital codes. The calculated neutron multiplications from a four-group scheme predict criticality in most of the experiments within 1%, and from a two-group scheme within 3%. Calculated neutron flux distributions are fitted with experimental traverses of subcadmium and epicadmium activation (of Mn). The accuracy of the calculations with regard to input data and method is discussed. (auth)
Reactivity changes resulting from the introduction of foreign materials into the hydride assemblies are discussed. Apparent regularities with respect to Z and qualitative interpretations of variations with radius are pointed out. From data for various radial positions, changes in critical mass corresponding to small changes in composition and density are computed. (auth)

38. 11489 AECID-4243
Los Alamos Scientific Lab., N. Mex.

39. 13953 LA-1614
Los Alamos Scientific Lab., N. Mex.

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

40. 1322 GAT-189
Goodyear Atomic Corp., Portsmouth, Ohio.

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

41. 12815 Y-533
Carbide and Carbon Chemicals Corp. Y-12 Plant, Oak Ridge, Tenn.

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

42. 9838 MonP-454
Clinton Labs., Oak Ridge, Tenn.

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

43. 11827 IIO-16172

Critical mass measurements were made for cylinders of $^{235}$U where the cylinder was bare, $H_2O$ reflected, $\text{H}_2\text{O}$ reflected with a $\text{Cd}$ interface, and bare with a concentric $\text{Cd}$ shell. The method of two-group, spherical geometry was used. (D.E.B.)
Critical Experiments 44-48

44.
13107 LWS-24712
California Research and Development Co., Livermore, Calif.
CRITICAL MASS STUDY FOR CYLINDRICAL GEOMETRY AS A FUNCTION OF RADIUS TO HEIGHT RATIO.
The results are presented of a study on the manner of variation of the critical mass of Pu and the peak-to-average power production ratio in a bare cylindrical reactor core. The particular case considered was that of a Pu, U, Fe, and Pb composite. Results indicate that increasing the ratio of critical radius to critical height from 0.5 to 1.0 would entail an increase in critical mass of 22%. (B.J.H.)

45.
13164 LA-1679
Los Alamos Scientific Lab., N. Mex.
The construction, operation, and typical uses of Topsy, the versatile, remotely controlled, critical assembly machine at Pajarito are described. Section 1 covers the mechanical design of the machine, and the hydraulic and electrical operation of the various components. Section 2 describes how Topsy is used for investigation reacting metal assemblies. Procedures for establishing a delayed critical configuration and operation at delayed critical are illustrated for the U235-U system. Also included are brief descriptions of U235-Ni, Pu-U, and low density and concentration assemblies that have been made on the machine. (auth)

46.
12703 ORNL-2332
Oak Ridge National Lab., Tenn.
A recent criticality study has indicated that the Thorex Pilot Plant is critically safe for the processing of thorium irradiated to less than 10,000 grams of U235 per ton. As the size of equipment exceeds the geometrically safe size and the total uranium inventory exceeds the minimum critical mass, minor equipment modifications and changes in operational procedures were necessary to provide safety under adverse operating conditions. (auth)

47.
14016 Y-881 (Del.)
Oak Ridge National Lab., Y-12 Area, Tenn.
An essentially bare graphite moderated critical assembly having dimensions of 51.0" × 44.11" and a critical mass of 52.48 kg of U235 was constructed at the ORNL Critical Experiments facility. Using a value for the buckling, B, of 0.0016828 cm⁻¹ and an extrapolation distance of 2 cm, a multigroup calculation for the assembly gave an effective multiplication of 0.9912. The self-shielding factor due to lumping of the fuel in 0.01 in. discs was measured and found to be 0.94 ± 0.04 compared to a calculated value 0.95. Values of 0.15 ev for the mean fission energy and 27.68% for the fraction of fissions in the thermal group were calculated. Core removal type control rods were calibrated by the "rod-drop" method and by the observation of stable reactor periods. The form of the control rod calibrations curves indicate a contribution due to neutron streaming in the void formed by withdrawing the rod as well as the expected cosine squared distribution. A comparison using a flat strip of cadmium as a poison rod, which left essentially no void when removed, showed good agreement with a cosine squared sensitivity curve. The loss in reactivity as a function of the separation of sections of the assembly showed a loss of multiplications of 0.00625 due to a gap 0.3 in. wide. Bare In and Cd covered In foils were exposed in various parts of the assembly to observe the flux distribution both macroscopically and microscopically. Power distributions were observed by means of Al catcher foils in contact with the U. Comparison between fission rates of bare, Cd covered, and Cd covered fuel discs gave values of 2.1 and 3.4 compared to a calculated average value was 2.98. Danger coefficients for Na, Fe, Ni, and Mo were calculated using the multigroup neutron spectrum and the known cross section data. (auth)
6966 LA-618 (델)
Los Alamos Scientific Lab., N. Mex.

The critical masses of enriched U-H mixtures under various conditions were measured, using two assembly structures embodying safety devices. A table of critical masses of UH2 in various tampers is given. The neutron density distribution inside WC and BeO tampers were measured with several detectors. In addition, a measurement was made of \( \frac{\nu}{\nu_f} \) for the UH2 spectrum of neutrons. The mean life of neutrons in a structure tamped with BeO was measured by means of the Rossi method. (auth)

10073 LA-2142
Los Alamos Scientific Lab., N. Mex.

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

Critical Experiments 49-54

49.

17715

Central-source neutron multiplication measurements have been made on small spheres of \( \text{U}^{233} \), \( \text{U}^{235} \), \( \text{U}^{238} \), and \( \text{Pu}^{239} \). Measurements were made on four sizes of spheres, varying approximately from 1 in. to 2.5 in. in diameter. In the case of \( \text{U}^{235} \), only one sphere, 1.245 in. in diameter, was available. The three neutron sources, mock-fission, Po-210, and Po-209, were 0.4-in. diameter spheres. They have widely different neutron spectra. Multiplications were measured with a flat-response long counter, and with \( \text{U}^{233} \), \( \text{U}^{235} \), and \( \text{Pu}^{239} \) spiral fission chambers, under conditions such that the error introduced by room-scattered neutrons was negligible. Results of the multiplication measurements are given. The measurements have been analyzed to obtain the leakage spectra from the spheres in terms of three velocity groups. The \( \text{Pu}^{239} \) and \( \text{U}^{238} \) fission chambers serve as threshold detectors defining the lower energy of the second and third velocity groups, respectively. (auth)

50.

10899 AECU-3606
Los Alamos Scientific Lab., N. Mex.

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

51.

4429
AECU-3105
Los Alamos Scientific Lab., N. Mex.

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

52.

18999 ABCD-4296
Cerado and Carbon Chemical Co. (K-85 Plant), Oak Ridge, Tenn.

Based on experimental data and on conservative theoretical considerations, an estimate has been made of the relation of the minimum U-235 critical mass to the hydrogen moderation for uranium material at the U-235 equilibria for which experimental information is available. These estimates have been used in indicating maximum safe amounts for conditions where definite moderation limits can be established. (auth)
Critical Experiments 55-61

55.
8757

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

56.
3200
AERE-RP/P-1810


57.
6550
ORNL-2367
Oak Ridge National Lab., Tenn.


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

58.
11786

Measurements of two independent types were made of the reactivity effect in a thermal test reactor of samples of U\textsuperscript{235}, Pu\textsuperscript{239}, and Pu\textsuperscript{241}. From these measurements average subcadmium values of eta (\eta) relative to \eta of U\textsuperscript{235} are obtained independently of other knowledge of the average absorption cross sections. Average absorption cross sections are also obtained from the measurements. Values of (\eta) for U\textsuperscript{235}, Pu\textsuperscript{239}, and Pu\textsuperscript{241} are respectively 2.231 \pm 0.034, 1.927 \pm 0.024, and 2.213 \pm 0.07. The corresponding value of (\eta 0.0253 ev) of Pu\textsuperscript{241} is found to be 0.025. A presentation of the method and results are given together with a comparison with previous work. (auth)

59.
4471
WAPD-P-695


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

60.
13434
TID-5345
Carbide and Carbon Chemicals Co. (K-25 Plant), Oak Ridge, Tenn.


$1.80(OTS); $1.60(OTS).

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

61.
15015
A/CONF.15/P/593
Oak Ridge National Lab., Tenn.

EXPERIMENTAL AND THEORETICAL STUDIES OF UNREFLECTED AQUEOUS U\textsuperscript{235} CRITICAL ASSEMBLIES. R. Gwinn, D. K. Trubey, and A. M. Wehnberg. 28p. $0.50(OTS).


The empirical kernel method for treating bare critical systems is discussed and is utilized to predict the material buckling of aqueous U\textsuperscript{235} bare reactors in an effort to clarify the age discrepancy. These results are compared with experimental critical and kinetic data which are also presented. In the comparison, it is found that the uncertainty regarding the proper experimental extrapolation distance arises and prevents an unambiguous comparison. (auth)
62.

15276 K-1380(Pt.F)
Oak Ridge National Lab., Tenn.
CRITICAL MASS DATA APPLICABLE TO NUCLEAR
SAFETY PROBLEMS. R. Owin and J. T. Thomas. Pt. F
[cf STUDIES IN NUCLEAR SAFETY. Lectures
Presented at the Nuclear Safety Training School Con­
ducted by Union Carbide Nuclear Company, June 3-14,
Representative data obtained at several laboratories
are summarized graphically. These include homogeneous
solutions, solids, effects of reflectors, and mention of
preliminary information on interacting systems. (T.R.H.)

63.

686 HW-51168
General Electric Co., Hanford Atomic Products
Operation, Richland, Wash.
PROGRESS REPORT ON EXPERIMENTS TO DETERMINE
INFINITE MULTIPLICATION FACTORS OF ENRICHED
$1.80(ph OTS); $1.80(mf OTS).
Experiments to determine maximum U°° enrichment for
UO₂-H₂O and UO₂(NO₃)₂-H₂O mixtures which will be sub­
critical independent of the H/U ratio and the volume are
described. (T.R.H.)

64.

662 LA-2141
Los Alamos Scientific Lab., N. Mex.
BERYLLIUM-REFLECTED, GRAPHITE-MODERATED
CRITICAL ASSEMBLIES. G. E. Hansen, J. C. Hoogterp,
J. D. Orndoff, H. C. Paxton, P. G. Koontz, W. H. Ronch,
$1.75(OTS).
Included are data on the properties of three sets of
 cylindrical Be reflected, graphite-moderated critical
assemblies that have been set up in the Honeycomb
machine. The first set was primarily to establish charac­
teristics as functions of C/O₂ atomic ratio of a core with
nearly constant reflector thickness. Fission rates were
mapped with U°° foils, and reactivity contributions of
some foreign materials were measured. The second set
consisted of three assemblies with fixed cores to deter­
mine the effect of redistributing reflector from the ends
to the cylindrical wall. Flux distributions in the basic
fully-reflect assembly were mapped extensively with
bare and with Cd-shielded foils of Cu. The third set
was to establish the minimum-volume core at C/O₂
~ 350 that could be made critical with available Be.
Experimental critical data converted to equivalent spherical
systems are compared with results of S₈ calculations.

65.

4428 AECU-3604
Los Alamos Scientific Lab., N. Mex.
CRITICAL ASSEMBLIES OF GRAPHITE AND ENRICHED
URALUM WITH BERYLLIUM REFLECTORS. G. E.
Hansen, J. C. Hoogterp, J. D. Orndoff, and N. C. Paxton.
20p. Contract W-7405-eng-36. $4.80(ph OTS); $2.70
(mf OTS).
Data are given on properties of three sets of cylindrical
Be reflected, graphite-moderated critical assemblies.
The first set was primarily to establish characteristics
as functions of C/O₂ atomic ratio of core with nearly
constant reflector thickness. Fission distributions were
determined. The second set consisted of three assemblies
with fixed core to determine the effect of redistributing
reflector from the ends to the cylindrical wall. This
series was done to provide the Los Alamos Scientific
Laboratory's Theoretical Division checks for two-dimen­
sional diffusion code. Flux distributions in the uniform­ly
reflected assembly were mapped extensively with bare
and Cd-shielded foils of oralloy, Au, and In. The third
set was to establish the minimum-volume core at C/O₂
~ 350 that could be made critical with available Be.
Experimental critical data converted to equivalent spherical
systems are compared with results of S₈ calculations.

66.

12893 LA-2203
Los Alamos Scientific Lab., N. Mex.
CRITICAL MASSES OF ORALLOY IN THIN REFLEC­
ATORS. G. E. Hansen, H. C. Paxton, D. P. Wood, K. W.
W-7405-eng-36. $6.30(ph OTS); $3.00(mf OTS).
Critical masses were measured for 5.25 in. diameter
O₂ cylinders in 0.5 and 1 in. thick reflectors of Be,
graphite, Mg, Al, Ti, mild steel, Cu, W alloy, Th, Ni,
Co, Mo, Al₂O₃, MoC, and polythene. These results
were converted to the equivalent spherical critical
masses of O₂ and compared to yield consistent trans­
port cross sections for the reflector materials. In addi­
tion, critical masses of O₂ spheres in ~ 2 and ~ 4 in.
and thick spherical reflectors of W alloy, Fe, Ni, Ni-silver,
Cu, Sn, Th, Be, BeO, C, and Th were determined. (auth)
Critical Experiments 67-72

67.
15017 A/CONF.15/P/592
Los Alamos Scientific Lab., N. Mex.
PROPERTIES OF ELEMENTARY FAST-NEUTRON CRITICAL ASSEMBLIES. G. E. Hansen. 15p. $0.50
(OTS).
Typical properties of elementary fast-neutron critical assemblies studied at the Los Alamos Scientific Labora-
tory are tabulated. The kinetic, spectral, and perturbative properties of Godiva, Topsy, and their Pu-
analogs were studied in detail. Results of critical measurements are supplemented by data from ura-
nium-metal exponential columns. The influences of shape and material composition on critical size are discussed.
The properties of spherical U and Pu assemblies are covered. (M.H.H.)

68.
9745 K-1019(4th Rev.(Del.))
Oak Ridge Gaseous Diffusion Plant, Tenn.
BASIC CRITICAL MASS INFORMATION AND ITS APPLICATION TO OAK RIDGE GASEOUS DIFFUSION
The current minimum experimental values of the basic criticality control parameters for $^{235}$U assays
between 2% and approximately 90% are presented together with the basic criticality control methods cur-
rently in effect at ORGDP. The fundamental nuclear safety criteria remain essentially unchanged from
previous editions of the report with the exception of the neutron interaction specifications, which have been
extended considerably, and the approval, for the first time, of the limited use of water in cascade fire con-
trol activities. A chart of the organization for nuclear safety control at ORGDP and a glossary are also in-
cluded. (auth)

69.
11384 AECD-4286
Carbide and Carbon Chemicals Co. [K-25 Plant],
Oak Ridge, Tenn.
VARIATION OF CRITICAL PARAMETERS BETWEEN U-235 ASSAYS OF 4.9 PERCENT AND 93.4 PERCENT.
eng-26. (K-399(Del.)). $3.30(ph OTS); $2.40(mf OTS).
The minimum critical mass, volume, and cylinder diameter for U of 4.9% $^{235}$U assay were determined as
1.83 kg, 32.1 liters, and 10.7 ln., respectively. Curves have also been prepared by applying the same constant
correction to each minimum parameter curve that was used in establishing “always-safe” values for 93.4%
assay material. These reduction curves may serve as possible criteria for establishing “always-safe” values
over a wide range of $^{235}$U assays. (W.L.H.)

70.
3898 LA-1155(Del.)
Los Alamos Scientific Lab., N. Mex.
ORALLOY SHAPE FACTOR MEASUREMENTS. V. Joseph-
Decl. with deletions Dec. 17, 1957. 35p. Contract W-7405-
eng-36. $6.30(ph OTS); $3.00(mf OTS).
Measurements were made at the Pajarito remote control laboratory to determine the effect of change of shape on
system reactivity for oralloy cylinders. Systems tested include cylindrical configurations with various height-to-
diameter ratios ranging from slabs to rods. Each system reactivity is referred to that of a sphere in the same
tamper. Reactivity tests were made on bare (untamped) Cy
configurations, as well as on systems in tuballoy tampers
1.12, 1.87, and 8.0 in. thick. The amount of reactivity
change associated with a particular cylinder height-to-
diameter ratio is found to be a function of tamper thick-
ness, and is greatest for very thin tampers. (auth)

71.
15585 HW-55707(Del.)
General Electric Co. Hanford Atomic Products
Operation, Richland, Wash.
NUCLEAR SAFETY IN PLUTONIUM METAL DISSOLU-
$3.30(ph OTS); $2.40(mf OTS).
Experimental evidence as well as theoretical considera-
tions are presented which indicate that a plutonium metal-plutonium solution system can be more reactive
than either one alone. More experiments as well as further theoretical development are required to achieve
a better understanding of such systems. (auth)

72.
4770 UCRL-4937
California, Univ., Livermore. Radiation Lab.
SPHERICAL AND CYLINDRICAL PLUTONIUM CRITICAL
MASSES. Fred A. Kloosterstrom. Sept. 1957. 18p. Con-
tact W-7405-eng-48. $3.30(ph OTS); $2.40(mf OTS).
Experiments to determine critical masses of 6-phase
plutonium cylinders of three diameters with thin metallic
reflectors are reported. Critical reflector thickness
measurements were made with two spherical Pu cores;
the cylindrical and spherical data are combined to yield
shape factors for the spheres for $^{239}$Pu and Be reflection.
(auth)
Critical Experiments 73-76

Theoretical fuel distribution having a continuous concentration gradient. The theoretical fuel distribution was that given by a calculation method developed in a theoretical treatment of the problem by Goertzel, who demonstrated mathematically that the condition of minimum critical mass in a suitably chosen thermal reactor required that the thermal neutron flux be uniform everywhere in the core. The experimental results clearly establish the validity of the Goertzel theory. The experimentally determined critical height and mass, using the theoretically determined fuel loading, were within 2.5% of the corresponding calculated parameters. The importance function for the fuel was shown to be constant throughout the core as required for a minimum critical mass reactor. The measured thermal and non-thermal components of the neutron flux were in good agreement with theory, and the thermal flux, except for deviations produced by the stepwise approximation to the ideal fuel distribution, was uniform along a radius of the core. The longitudinal neutron flux behaved as expected; a measure of the Cd fraction showed the reactor to be essentially thermal, and in general, data obtained from the experimental reactor were compatible with the postulates and predictions of the Goertzel theory. An exploratory investigation to obtain experimental verification of the conditions of the theory made by Goertzel to predict how the critical mass of fuel can be minimized in a reactor of less than optimum radius met with little success. Although some lowering of the critical mass was produced by the theoretically determined fuel distribution, a discrepancy of more than 35% was found to exist between theory and experiment. (auth)
Critical Experiments 77-81

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

77. 2121 LA-2023
Los Alamos Scientific Lab., N. Mex.

Reflected cylindrical exponential columns of bare and natural U were constructed of stacked plates of oralloy (uranium, enriched to 93.4%) and tuballoy. Average concentrations of U\textsuperscript{235} ranging from 0.72% to 9.16% were investigated. Flux distributions, buckling values, and spectral indices were obtained. The indicated U\textsuperscript{235} concentration for infinite critical mass is about 5.5%, (auth)

78. 1139 LA-1958(Del.)
Los Alamos Scientific Lab., N. Mex.

Data on critical configurations of fissionable metals are summarized in a form emphasizing the influence of conditions commonly of concern in nuclear safety questions. Although the bulk of the data is derived from experiments with uranium metal enriched, usually, to about 90 wt. % U\textsuperscript{235} there are enough data for Pu and U\textsuperscript{232} to establish some general relations about their criticality relative to U\textsuperscript{235}. The specific information includes the influences exerted upon critical mass by various reflectors, by shape of the fissionable material, by variations in U\textsuperscript{235} concentration, by variations in material density, and by graphite and hydrogenous diluents. (auth)

79. 15123 A/CONF.15/P/2408
California, Univ., Livermore. Radiation Lab.
CRITICAL MEASUREMENTS AND CALCULATIONS FOR ENRICHED-URANIUM GRAPHITE-MODERATED SYSTEMS. H. L. Reynolds. 27p. (UCRL-6175). $0.50 (OTS).

Prepared for the Second U. N. International Conference on the Peaceful Uses of Atomic Energy, 1958. The experimental results on a variety of enriched-uranium, graphite-moderated systems, both with and without reflectors are presented. Also included are the results of the multi-group transport, and one- and two-dimensional diffusion calculations used to interpret the data. In general the University of California Radiation Laboratory systems were bare or one-dimensional in order to simplify the analysis. Efforts were made to obtain systems as close as possible to an idealized system containing only a homogeneous mixture of moderator and fuel with all extraneous factors removed. The Los Alamos Scientific Laboratory experiments were carried out with more complex geometries approaching more closely potential reactor systems. All of the assemblies utilized heterogeneous arrangements of moderator and thin uranium foils. The bare systems range in carbon-to-uranium atomic ratios from 300/1 to 2500/1. These systems are not truly thermal and are in a range where critical mass is extremely sensitive to size or buckling. Experiments were performed for these systems to determine accurately the effects of extraneous factors such as room return, control-rod void spaces, nonhomogeneity of fuel loading, moderator block porosities and poison content. The reflectors include graphite and beryllium in one-, two-, and three-dimensional arrangements. The constants used in the multigroup calculations are presented in tabular form. Eighteen energy groups were used. The same constants were used in the transport and diffusion calculations to allow comparison of the two methods of calculation. (auth)

80. 8226 UCRL-4975
California, Univ., Livermore. Radiation Lab.

The critical thickness of beryllium reflector was determined for oralloy (uranium, enriched to 93.17% U\textsuperscript{235}) spheres ranging from 10.765 kg to 32.624 kg. Four points which were determined by other experimenters were normalized to the data and are included in the curves. (auth)

81. 12087 AECO-4264

A study was made of the available high-assay criticality data for materials of 93.4% and 95.3% U\textsuperscript{235}, and minimum critical volumes were estimated for several conditions. The data, estimated for aluminum and stainless steel containers, water and air reflected, and with and without shielding, are tabulated. (J.S.R.)
Critical Experiments 82-87

82.
2176 Y-629
In a previous report a method of graph analysis was developed for obtaining the critical dimensions of semi-infinite U²³⁵ water reactors from measurements made on finite systems. The empirical treatment is applied here to analysis of experimental data. The cases considered are: (1) a comparison of critical conditions occurring in Al-walled and stainless steel-walled reactors, and (2) a comparison of critical conditions in isolated and interacting pairs of water-reflected cylinders at the moderation which minimizes the critical volume. (auth)

83.
11005 UCRL-5006
A system for the measurement of near-homogeneous carbon-U²³⁵ critical masses is described. Cores are constructed with thin, enriched-uranium foils spaced between graphite blocks. Fuel density is variable by use of different foil thicknesses and spacings. Reactivity is controlled by boron rods; standard reactor instrumentation permits critical operation at low power. Results of critical measurements on unreflected systems having atomic C/U²³⁵ ratios of 570, 1380, and 2350 are given. Thin reflectors of graphite and beryllium were also used. Corrections for self-shielding in the fuel foils and systematic errors are described. (auth)

84.
10078 ORNL-2499
Oak Ridge National Lab., Tenn.
This report contains a description of a series of two-group and multigroup calculations of the critical mass of two clean-geometry configurations of the BBR. It also contains a description of critical experiments that were done to determine the validity of the calculations. Comparison of the results indicates that the calculations described are capable of predicting the critical mass within about 2% of the measured critical mass under the favorable geometric conditions maintained in the present experiments. (auth)

85.
1642 KAPL-M-PPS-1
Kolls Atomic Power Lab., Schenectady, N. Y.
The critical mass density of U²³⁵ has been determined as a function of buckling for one region zirconium-water and stainless steel-water cores with and without U²³⁴. This functional relationship enables results to be applied to any type of geometry. Using reflector savings derived from multigroup calculations the above results were applied to a limited range of cylindrical cores with water reflectors. The critical mass as a function of the volume of stainless steel, the difference between critical masses hot and cold for a particular composition, and reactivity coefficients for the uranium mass are also shown. (auth)

86.
10086 WAPD-TM-100
Experiments were performed on light-water-moderated, highly enriched uranium-zirconium-aluminum cores in slab geometry. In one case, light water was used as a reflector on all sides of the core. In the second, a metal and water reflector containing natural uranium-niobium-aluminum was symmetrically added to a thinner, enriched uranium-zirconium-aluminum slab core. Experimental flux plots and machine calculated flux plots agreed within the experimental and calculational uncertainties. Few-group slowing-down models combined with spatial diffusion theory were used to predict criticality of the assemblies to within 2% in the worst case. The four-group slowing-down model, incorporating the effects of changes in the thermal flux spectrum, gave the best results. (auth)

87.
131
Critical measurements were carried out with solutions of UO₂ (93.6% U²³⁵) dissolved in 4.3 M H₃PO₄. A brief description of these measurements, their results, and an evaluation of some calculations are presented. (A.C.)
88.

5125 CF-58-12-36
Oak Ridge National Lab., Tenn.
CRITICAL CONCENTRATION DATA FOR HRT-TYPE
REACTORS MODERATED BY D_2O-H_2O MIXTURES.
eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Studies were made concerning the smallest core
diameter and the most appropriate D_2O-H_2O composition
to achieve criticality in an HRT-type reactor; the fuel solution should contain no more than
10 g U/liter at 280°C. For the present core diameter of
32 inches (two-region operation), it was found that an
addition of 10% light water to the moderator would re­
duce the critical concentration from 9.2 to about 7.8 g
U/liter at 280°C (concentrations in g total U/liter, based
on U of 93.4% enrichment). The smallest core diameter
for which the reactor would remain critical with less
than 10 g U/liter is about 29 inches. This occurred with
a moderator composition of about 86% D_2O. (auth)

89.

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

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

90.

2087 UCRL-5255
CRITICAL-MASS DETERMINATIONS OF LEAD­
REFLECTED SYSTEMS. Robert E. Donaldson and

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

91.

17458
REACTIVITY AND FLUX MEASUREMENTS IN HIGHLY
REFLECTED U^{235}-BISMUTH-GRAPHITE CRITICAL
EXPERIMENTS. T. C. Engelder, H. W. Giesler, and
J. P. Farrarr (Babcock and Wilcox Co.). Nuclear Sci.

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

92.

21043 ORNL-2143
Oak Ridge National Lab., Tenn.
CRITICAL MASS STUDIES. PART VIII. AQUEOUS
SOLUTIONS OF U^{235}. J. K. Fox, L. W. Gilley, and
eng-26. $1.00(OTS).

A series of experiments were performed to establish
the critical parameters of aqueous solutions of uranyl
nitrate and uranyl fluoride in which the uranium con­
tained 98.7% U^{235}. Solutions were made critical in both
spherical and cylindrical geometries with paraffin or
water as a neutron reflector and, in two instances, with
no reflector. The U^{235} concentration varied from 30 to
600 g/liter. The minimum critical mass observed was
590 g of U^{235} in the solution having an H : U^{235} atomic
ratio of 419 occupying a 10.4-in.-diam sphere. The
minimum measured volume was 3.66 liters in a 6.7-in.
equilateral cylinder containing a solution with an H : U^{235}
atomic ratio of 39.4. Extrapolated source neutron mul­
tiplication data indicate that a 5-in.-diam cylinder can
be made critical if reflected, but a 4-in.-diam cylinder
would be subcritical at all moderations. It was also
found that 2.02 kg of U^{235} in an unreflected 10-in.
equililateral cylinder is critical with a solution having
an H : U^{235} atomic ratio of 154. An unreflected sphere
12.6 in. in diameter is critical with 1.14 kg of U^{235} in a
solution with an H : U^{235} ratio of 381. Extension of the
data to geometries other than those used experimentally
was made by an empirical calculation. (auth)
ASSEMBLY OF URANIUM—BERYLLIUM OXIDE. C. Gourdon, J. Martelly, M. Sagot, and G. Wanner. Bull. Inform. ORXIDE. CRITICAL uranium-beryllium oxide lattices by the classical axis. neutron spectra. Cacimlum possibly to attain criticality under quite favorable conditions. The assembly has no racall reflector channels with a cross section of 5 x 5 cm². By charging the assembly with 700 kg of uranium, enriched to 1.3%, it was possible to attain criticality under quite favorable conditions. The assembly has no radial reflector and the axial reflectors were poisoned with cadmium to have a very small albedo and to avoid a perturbation of the neutron spectra. Cadmium tubes around each uranium rod were moved up or down to control the reactivity. The material buckling, photoneutron effect in BeO, and the temperature and pressure coefficients of reactivity were studied. (J.S.R.)


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


Details are given of a series of critical assemblies and flux measurements undertaken in HAZEL using 46% U²³⁵ as UO₂F₂ in D₂O in cylindrical geometry. The fissile concentrations investigated were in the range 1939 to 6722 D/U₂, and in the geometry considered (2 inch diameter steel cylinder), the critical heights were in the range 70 to 200 cms. The minimum critical mass was 1.768 kg of U²³⁵ occurring at a concentration of 4687 D/U₂. (auth)

A more encompassing, yet consistent and practical, method of calculating the critical mass of aqueous uranyl fluoride solutions was developed. Comparison of calculated results with experimentation gave a maximum of only 2.3% error in the effective multiplication factor and 2% error in the critical mass, with absolute averages of 0.7% and 3.4%, respectively, through the ranges from high moderation down to an H/U of 25 for 99% U²³⁵ assays or an H/U of 4 for 2% U²³⁵ assays. (auth)
Critical Experiments 100-104

100.

5475  HW-58049
General Electric Co. Hanford Atomic Products
Operation, Richland, Wash.
NUCLEAR SAFETY IN PROCESSING LESS THAN 5.0% U-235 ENRICHED REACTOR FUELS. [Norman]
Eng-32. $4.80(p OTS); $2.70(mf OTS).
Processing of reactor fuels in which the initial enrichment is less than 5% U\textsuperscript{235} is considered. The critical parameters for heterogeneous systems of fuel in water or in uranium solutions as well as homogeneous solutions of fuel and water are examined. In addition, nuclear safety in the use of boron poisoning to increase safe batch sizes and in the use of safe vessel geometries is discussed. A cartridge-type dissolver system for fuel elements is described, and it is pointed out that experiments to determine the neutron reflecting properties of enriched uranium solutions surrounding vessels are justified. (J.R.D.)

101.

5473  BNL-489(p.275-90)
California Univ., Livermore. Radiation Lab.
STUDIES OF ENRICHED URANIUM GRAPHITE REACTOR SYSTEMS. Albert J. Kirchbaum, Appendix:
A/CONF.15/P/2408 supersedes this information in this paper.
The results to date are obtained from studies of essentially homogeneous enriched uranium (93.5% U\textsuperscript{235}) graphite systems. Critical configurations for bare and graphite or beryllium reflected cores are given for carbon to uranium atomic ratios of 600:1, 1200:1, and 2400:1. The results of experiments to determine the systematic errors are given. This allows reduction of the critical size data to idealized geometries for comparison with neutron calculations. By use of a pulsed neutron source, data on the prompt neutron population relaxation time as a function of buckling were obtained. The experimental technique and resultant data are discussed. A comparison of the critical buckling and time behavior data with a simple modified Fermi age theory is made. This includes discussion of the prompt neutron lifetime effectiveness of the control and safety rod system, the bulk neutron properties of the graphite, and self shielding effects of the uranium. (auth)

102.

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

103.

22212  UCRL-5369(P.1)
California, Univ., Livermore, Lawrence Radiation Lab.
A series of critical measurements on unreflected systems, fueled by thin Oy foils, is described. Fuel density is varied by use of different foil thicknesses and spacings between foils. Five fuel densities were used which correspond to atomic BeO/U\textsuperscript{235} ratios from 246 to 7680. For these three ratios, the fuel foil thickness was varied to find effects of self-shielding and flux depression. (auth)

104.

17970  AECD-4285
Los Alamos Scientific Lab., N. Mex.
A critical metal assembly is described that has a cylindrical core of uranium with an average U\textsuperscript{235} concentration of 161/2% and a 3 inch reflector of natural uranium. The critical mass was determined to be 692 ± 4 kg of core material, from which a value for the critical mass of a bare sphere, having the same concentration, is calculated. Measurements of the prompt neutron decay constant, reactivity contributions of several materials, and radial and axial variations of fission rates of U\textsuperscript{235}, U\textsuperscript{238}, and Np\textsuperscript{237}, as well as ratios of fission cross sections for these isotopes, are reported. (auth)
A series of critical experiments with blocks of $^{235}U$-enriched $UF_4-C_2H_5Cl$ was initiated at the ORNL Critical Experiments Facility. Thus far assemblies with $H:UF_4$ atomic ratios of 195 and 294 were built in parallelepipedal and simulated cylindrical geometries, both reflected and unreflected. From the results the minimum critical masses for reflected spheres were determined to be 16.3 and 8.5 kg of $^{235}U$ for fuel mixtures with $H:UF_4$ atomic ratios of 195 and 294, respectively. The minimum critical masses for unreflected spheres of these two fuel mixtures are 24.3 and 12.7 kg of $^{235}U$, respectively. (auth)

A method of correlating the critical conditions of bare homogeneous reactors is presented. This method is applied to the results of an 18 group analysis for a series of bare, cold, homogeneous reactors comprised of pure $^{235}U$ and the moderators $H_2O$, $D_2O$, $LiH$, $Be$, $BeO$, and $C$ over a wide range of moderat or uranium mole ratios. The properties contained in the correlation parameters used are the buckling, the transport
Critical Experiments 111-115

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

111. BNL-463 (p.73-4)
Los Alamos Scientific Lab., N. Mex.
Data consisting of critical reflector thicknesses for 2.41 kg and 19.0 kg $U^{233}$ metal spheres are summarized. Six-group $S_4$ predictions are compared with observed results for the Goralley-enriched U and U reflected cases. (W.L.H.)

112. UCRL-5349
The critical thicknesses of Be reflectors for α-phase Pu spheres of 2.472, 3.217, 3.933, 4.664, and 5.426 kg were found to be 32.0 ± 4.0, 21.0 ± 1.0, 13.0 ± 0.1, 8.17 ± 0.05, and 5.22 ± 0.02 cm, respectively, for Be density of 1.0 g/cc and Pu average density 19.25 g/cc. A description of measuring techniques and apparatus is given. (T.R.H.)

113. TID-3533
Technical Information Service Extension, AEC.
A compilation of 1122 references on criticality studies is presented including references to experimental and theoretical studies of conditions for criticality with various materials and configurations, as well as references on the safe handling of fissile materials, both during critical experiments and in plants for the chemical and metallurgical processing of fissile materials. (W.D.M.)

114. ORNL-2201
Oak Ridge National Lab., Tenn.
Two unreflected critical assemblies using beryllium as the moderator and 93.4% enriched uranium metal as the fuel were built to provide a basis for the evaluation of certain reactor calculational procedures. Control and safety rods of the core-element-removal type were used in order that the final assemblies would not be complicated by strong absorber rods. In the first assembly (CA-1), which had outside dimensions of 21.0 × 21.0 × 23.3 in, the 0.01-in.-thick uranium disks were separated by 1-in.-thick blocks of beryllium, which gave a Be:U$^{235}$ atomic ratio of 390 and a fuel loading of 18.08 kg of U$^{235}$. The extrapolated value of $k_{eff}$ for the system was 1.0054. In the second assembly (CA-16), which had outside dimensions of 24.0 × 28.4 × 24.1 in, the fuel disks were separated by 4-in.-thick blocks of beryllium, which gave a Be:U$^{235}$ ratio of 1560 and a fuel loading of 7.65 kg of U$^{235}$. For this assembly the extrapolated $k_{eff}$ value was 1.0020. The observed uranium cadmium fractions in the two assemblies were 0.46 and 0.86, respectively. A number of multigroup calculations were made to evaluate the effects of various corrections and assumptions. It was concluded that the calculated neutron multiplication is very sensitive to the competition between leakage and slowing down at high energies, a range where fundamental data are uncertain. Without resolving the detailed neutron behaviors in this range, a reasonable selection of data within experimental uncertainties will give satisfactory values for such quantities as critical size. (auth)

1960

115. BAW-1173
The cores to be studied are described, and the status of the project is given. The results of measurements on core #35-58 are discussed in terms of critical mass, perturbation of M/W, thermal disadvantage factor, buckling, and cadmium ratio experiments. (W.D.M.)
Critical Experiments 116-122

116.

22263 RFP-180
Dow Chemical Co. Rocky Flats Plant, Denver.

Contract AT(29-1)-1106. OTS.

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

117.

810 AERE-R/1-2731


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

118.

19691

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

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

119.

16186 GAT-DM-769
Goodyear Atomic Corp., Portsmouth, Ohio.

SAFE GEOMETRIES AND MASS AT ASSAYS BELOW FIVE PER CENT U235. J. L. Feuerbach. May 18, 1959. 9p. OTS.

Investigations were made to determine the most reliable method of estimating safe geometries at assays below 6% U235. Conservative methods of estimating safe mass at the assays considered are also examined. Data on safe geometrical and mass parameters are included. (J.R.D.)

120.

13125 HW-D4454
Sandford Works, Richland, Wash.


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

121.

6844 RFP-158
Dow Chemical Co. Rocky Flats Plant, Denver.


Neutron multiplication measurements were made on a number of cylindrical assemblies of Pu and graphite disks. S6 calculations were made on homogeneous mixtures of Pu and graphite with varying C/Pu ratios and varying reflector thickness. (auth)

122.

9165 ANL-6115
Argonne National Lab., Lemont, Ill.


Critically calculations for the TREAT reactor, assuming no slots and no test hole, and using a modified Fermi Age Theory which allows for epithermal absorption and fission, indicated a critical radius of 59 cm. At the time of startup TREAT became critical at a radius of 67.8 cm. The results of spectrochemical and chemical tests indicated a high boron impurity (an average of 7.6 ppm) in the core graphite, and a maximum of about 1.6 wt. % H2O in one sample from the permanent reflector graphite. The excess amount of boron impurity in the core graphite appears to be the major cause of the discrepancy. Values of the negative temperature coefficient of reactivity and of neutron lifetime were calculated for the critical reactor containing 7.6 ppm of boron, assuming no slots or test holes, and the results are found to be in fair agreement with measurements. The source of excess boron impurity was traced to the borated steel separators which were used during the baking of the fuel elements. (auth)
Critical Experiments 123-128

123.

12330

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

124.

16167 HW-24514(Del.)


The chain reacting conditions for plutonium nitrate in water solution were examined experimentally for a variety of sizes of spheres and cylinders. The effects on the critical mass of the displacement of hydrogen and the addition of poisons to the fuel were measured in water tamped and bare reactors. The data reveal that the absorption cross-section of Pu-239 is 925 ± 200 barns and the minimum critical mass of Pu-239 in water is 610 g at a concentration of about 33 g/liter. (auth)

125.

127.

7988

Assemblies considered consist of approximate spheres of enriched-uranium hydride composition (approximating UH3) in 8-in. thick normal uranium and nickel reflectors and in a uranium reflector with nickel liner. Data are of the following types: (1) critical sizes, (2) values of Roos alpha in the neighborhood of delayed critical, (3) activation rates of various internal neutron detectors, and (4) reactivity coefficients of a variety of elements. From the reactivity coefficients at various radial positions, changes in critical mass corresponding to small changes in composition and density are computed. (auth)

126.

20789 HW-63576(p.65-6)


The program of critical approach and exponential measurements of 3.053 per cent enriched uranium rods in light water was continued. Critical masses, buckling values, and some measured extrapolation length values were reported for three rod diameters (0.300, 0.600, and 0.925 inches) in previous quarterly reports. Measurements were made with rods of 0.175-inch diameter by 23.5 inches in length. These rods were ensceu with 0.025-inch wall Lucite tubes for insertion into hexagonal lattice frameworks. Measurements were carried out in the same manner as described previously. All lattices were moderated and completely reflected with light water. (auth)

127.

14250 CF-80-4-24
Oak Ridge National Lab., Tenn.

COMPARISON OF k_{eff} MEASUREMENTS IN A CRITICAL ASSEMBLY WITH k_{eq} MEASUREMENTS IN THE PHYSICAL CONSTANTS TESTING REACTOR. J. T. Mihalzo. May 3, 1960. 34 p. Contract W-7405-eng-28. OTS.

The infinite medium multiplication factor, k_{eq}, for a homogeneous mixture of 5% U-235-enriched UF_{2} in paraffin, was determined from a series of critical experiments in which known changes in the buckling were made and the resulting stable periods measured. The value determined, using a two-group model for the nonleakage probability, was 1.197 ± 0.09. Within the quoted errors this value is in agreement with an earlier value of k_{eq} = 1.216 ± 0.013 for the same material experimentally determined in the Physical Constants Testing Reactor. (auth)

128.

1103 LAMS-2288(Suppl. 1)
Los Alamos Scientific Lab., N. Mex.


Multigroup neutron diffusion and transport equations have been shown to correlate neutron cross sections and simple critical experiments for a wide variety of materials to approximately second order accuracy. This system was used for criticality survey work for the moderators H and C, giving critical radius for moderator to fissionable material atomic ratio and temperature. Critical radius dependence for D_{2}O, Be, and BeO was computed for U-235 only. Neutron group averaged cross sections supplementing and correcting previous listings are tabulated. These are consistent with the literature. (auth)

The value of $k_e$ for 2 wt. % U$^{235}$ enriched UF$_6$ paraffin moderated at a H/U$^{235}$ atomic ratio of 186 was determined in the PCTR. This material was returned to ORNL in order that critical experiments might be conducted to determine $k_e$. The two values of $k_e$ were then compared. The value of $k_e$ as determined in the PCTR was dependent upon the cross-section values which were chosen. In this experiment "effective" cross-section values obtained by averaging the cross section over the Wigner-Zirkens spectrum of a similar type of mixture were used. The value of $k_e$ determined in the PCTR was 1.216 ± 0.013. The value obtained from the work carried out at ORNL was 1.200 ± 0.011 from a one-group treatment of the critical experiments and 1.202 ± 0.012 from a two-group treatment of the work. A theoretical calculation of $k_e$ gives a value of 1.23. This is in good agreement with the experimental value.

Critical experiments done on this material at ORNL show that the minimum critical mass for a bare "square" cylinder contains approximately 28 kg of U$^{235}$. A calculated value of the amount of U$^{235}$ in a just critical bare square cylinder was obtained using the experimental $k_e$ value. Again there was good agreement with the experimental results. (auth)


The original correlation on highly enriched uranium-hydrogen critical systems is extended to all enrichments. By using three empirical equations and the one-group

buckling relations, the physical size of any uranium-hydrogen homogeneous critical assembly in simple geometry can be predicted. The predictions are compared with experiment where experimental data are available. The derived reflected spherical, infinite cylinder, and slab dimensions are shown. (auth)
Critical Experiments 135-140

135.

25020


A facility called "Hot Box" was constructed at Jackass Flats, Nevada, to be used in studies of high-temperature critical assemblies of simple geometry, in order to determine whether calculation procedures are adequate for estimating the effect of temperature changes. Uranium graphite-moderated bare-unreflected and reflected systems at temperatures up to 1200°F and should be useful for nuclear propulsion system calculations. (D.L.C.)

136.

9911 RFP-178

Dow Chemical Co., Rocky Flats Plant, Denver.


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

137.

16169 Y-939

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


The critical mass curves of an equilateral cylindrical reactor as a function of moderation were calculated by graphical methods for H: U$^{235}$ atomic ratios of 25 down to 1.4. The critical assemblies were built up of small cubes. These cubes were of two types, H-cubes and U-cubes. The H-cubes were small blocks of polyethylene (CH$_2$)$_2$ approximately 1" on the edge. The U plastic cubes (U-cubes) contained a mixture of UF$_4$ (U$^{235}$) isotopic concentration of 9.3% and polytetrafluoroethylene (CF$_2$)$_2$ pressed together to form a material having an over-all density of 4.73 g/cm$^3$. The infinite cylinder and slab dimensions as functions of moderation were approximated for the low values of H: U$^{235}$ atomic ratio. (C.J.F.G.)

138.

819 NP-8000

Lockheed Nuclear Products, Marietta, Ga.


One- and two-dimensional flux distributions are given for the Critical Experiment Reactor using two-group diffusion theory. One-dimensional flux distributions are shown for the two- and four-region reactor models along with two-dimensional iso-flux distributions. The calculated critical loading was found to be 18.6 fuel elements and is 0.9% lower than the experimentally-determined critical loading value. Values of K$_{eff}$ for a 30- and 32-element core, using a two-dimensional, two-group diffusion theory program, are 1.106 and 1.130, respectively. One- and two-dimensional methods for calculating reactivities of the CER agree within 1%. (auth)

139.

19727 RAW-1193

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


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

140.

22261 AHSB(S) Handbook I


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

Criticality data are presented which are intended for use by chemical plant designers and operators. The assumptions and definitions are based on operations and accidents that might reasonably be considered possible in such plants. Graphs are presented of the four commonly-encountered critical parameters: mass, volume, radius of an infinite cylinder, and thickness of an infinite slab. Curves are given for systems involving Pu$^{239}, 30-93\%$ U$^{238}, 5-30\%$ U$^{238}$, and less than 5% U$^{238}$. (W.D.M.)
144. **1961**


Three critical plutonium-enriched uranium composites with plutonium of different Pu-239 content are described. In these systems one gram of Pu-239 is equivalent to ~0.61 grams of Pu-239 and an analysis which translates this datum to $P(Pu^{239}, 2 MeV) = 3.32 \pm 0.14$ neutrons emitted per Pu-239 fission induced by a 2 Mev neutron. (auth)

**14420**

**2068** Y-901 (Del.)
Oak Ridge National Lab., Y-12 Area, Tenn.


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

**14430**


Certain conflicts arising from previous measurements of neutron flux parameters in the equilibrium spectrum of natural uranium were resolved. The parameters which were investigated are given along with "best" values as measured in this work: Material buckling, 0.0119 \pm 0.0005 cm$^{-1}$; Diffusion length, 0.17 \pm 0.18 cm; $U^{235}/U^{238}$ fission cross-section ratio, 239 \pm 7; $Pu^{239}/Pu^{238}$ fission cross-section ratio, 250 \pm 16; $Np^{237}/U^{238}$ fission cross-section ratio, 14.5 \pm 0.5; and $U^{238}$ inelastic scattering cross section, 2.00 \pm 0.04 barns. The experiment was performed at the Pajarito Critical assemblies facility utilizing two exponential columns of natural uranium, each 30.7-in. high, having diameters of 15 and 21 in. and excited by a small fast reactor. The system was outdoors, elevated some 11 ft above the ground level to reduce flux perturbations due to backscattering of neutrons. Perturbation corrected measurements in both columns made by several detection methods and with various sources spectra agree to within experimental error and are consistent with calculated values. (auth)


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

**14450**


The SNAP Critical Assembly is a spheroidal nuclear reactor with a fixed hydrogen moderator, 93.17% $U^{235}$ fuel, and a beryllium and graphite reflector. The core is made up of segments of cold pressed ZrH with 6% by weight $UO_2$ powder. The assembly is constructed in two hemispheres with horizontal faces which are brought in contact. Instrumentation and reactivity, activation, and intrinsic behavior measurements are discussed. (W.D.M.)

**14462**

Critical Experiments 148-152

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

148.

7018 (ORNL-2016 (p. 58-70)) SOLUTION EXPERIMENTS IN A FLUX-TRAP CRITICAL ASSEMBLY: PRELIMINARY STUDY FOR HIGH FLUX ISOTOPE REACTOR. J. K. Fox, L. W. Gilley, and D. W. Magnuson (Oak Ridge National Lab., Tenn.).

A study of a flux-trap critical assembly was made, using a solution of $UO_3+NO_2$ enriched to 95.5 wt. % in $^{235}$U and dissolved in mixtures of $D_2O$ and $H_2O$. The assembly was a preliminary mock-up of the HFIR, and the data were intended to aid in the establishment of design parameters for the HFIR. Critical parameters, relative flux distributions, flux ratios, and the effects of centrally located air- or aluminium-filled "voids" were obtained. (auth)

149.

2063 AERE-M-718


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

150.


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

151.


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

152.


Recent experimental criticality data with homogeneous and heterogeneous systems of interacting containers were used in evaluating an interaction principle for the safe storage and handling of dissimilar containers of fissile materials. The experimental data which included slab and cylindrical geometries, U-235 assays of 93.2%, and H/U-235 atomic ratios from 0 to 330, and which extend below the useful range of a two-group theory previously
used to evaluate interaction experiments, indicate that the principle is valid over the wide range of criticality parameters considered, and that a homogeneous system of interacting containers is, in general, more highly reactive than any corresponding heterogeneous one. An analysis was also made of the safety of cylindrical storage units where criticality control is based upon mass rather than upon geometric limitations. Calculations using a two-group interaction theory indicate that, for containers meeting ORGDPS safe interaction criteria, either uniform dilution or concentration of the fuel from an optimum \( H/U_{235} \) ratio of about 600 will result in a smaller container separation being required. (auth)

153.


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

154.


Two plutonium-metal critical assemblies were studied at the Pajarito site in Los Alamos. Part I describes Jezebel, the bare plutonium assembly, and gives its observed characteristics along with a few comparisons with enriched-uranium systems. Part II covers Popsy, a plutonium core in a thick normal-uranium reflector. As Popsy was relatively inflexible—intended only for a preliminary survey—its experimental program was much less complete than that of Jezebel. (auth)
159. (ORNL-2016(p.73-61)) CRITICAL PARAMETERS OF BARE AND REFLECTED 93.4 wt. % U\textsuperscript{235}-ENRICHED URANIUM METAL SLABS. J. T. Mihalco and J. J. Lynn (Oak Ridge National Lab., Tenn.).

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


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


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


This report supplements LAMS-2415. Correlations of computed and experimental critical data are discussed. The scope is limited to fissile systems that may be approximated by simple descriptions. The methods used to adjust data to uniform conditions are outlined. The requirements for computation where the values are intended to substitute for experimental data in nuclear safety guidance are described. Computational methods using DBN and multigroup diffusion techniques are discussed. Both computed and experimental data are given for the following: critical masses of bare spheres of uranium with various moderators, critical masses and volumes of homogeneous water moderated uranium spheres, core-density exponents for water reflected-water moderated U\textsuperscript{235} or Pu\textsuperscript{239} spheres, critical diameters of infinite cylinders of homogeneous water moderated uranium, critical thicknesses of infinite slabs of homogeneous water moderated uranium, critical masses of delta-phase plutonium mixtures with water and Plexiglas, influence of Pu\textsuperscript{239} on critical mass of water moderated plutonium spheres, critical diameters of infinitely-long cylinders and critical thickness of infinite slabs of homogeneous water moderated Pu\textsuperscript{239}, critical volumes of \(\text{U-}^6\text{H}_2\text{O}\) systems, and critical masses of bare spheres of uranium diluted with other metals and graphite. (M.C.G.)
The thickness of reflectors required for critical configurations with spheres of U₂³⁵ and Pu₂³⁹ were estimated from multiplication measurements of nearly critical assemblies. Reflectors employed were uranium enriched in U₂³⁵, normal uranium, beryllium, and tungsten alloy. Correction of the experimental data was attempted to give "idealized" dimensions, i.e., a solid core in intimate contact with its reflector material. (auth)

166. 2065 LA-267
Los Alamos Scientific Lab., N. Mex.
An experiment is described in which a source of neutrons is surrounded by a sphere of active material. The resulting counts when the sphere is in place and removed are compared. The ratio of the two rates is a measure of the reactive properties of the sphere. Results are compared with those obtained by other workers. (auth)

Twenty-three small, cylindrical, UO₂F₂ aqueous solution, water-reflected, critical assemblies were analyzed utilizing few-group diffusion theory with the one-dimensional code WANDA. The fast-group constants were obtained using the MUFT code which computes the flux spectrum in a material of prescribed composition and buckling. The cross sections are then averaged over this spectrum to obtain the MUFT constants. Maxwellian thermal spectrum constants for the slow group were obtained with the SOFOCATE code. Also, three large, UO₂(NO₃)₂ aqueous solution, cylindrical critical assemblies were analyzed for comparison. The purpose of the analysis was to make several direct comparisons of critical experiments with analytical models in order to determine whether these models can be utilized to calculate criticality for assemblies that are not complicated with heterogeneous structural materials. The general use of these analytical models for fuel handling criticality calculations was evaluated. (auth)


Critical Experiments 165-172

A compilation is presented of critical extrapolations of subcritical neutron multiplication measurements made on assemblies of plutonium metal tamped with Plexiglas. In addition to this compilation, recent data are reported on a 20- by 20-in. slab of metal tamped with Plexiglas. A simple empirical equation of the form \((\frac{1}{K_a} - \frac{1}{K_d}) (h-b) - C\) was found which fits the data and predicts the infinite slab and cylinder dimensions. (auth)

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

A study is made of the criticality conditions at the fuel reprocessing plant of the Canada India Reactor. The various stages in the fuel processing operation, whose purpose is to concentrate and recover the Pu₂³⁹ produced in the reactor, are described. The criticality aspects of each stage are considered. (T.F.H.)

About 25500 fuel disks of 90% enriched UO₂-graphite for the Semi-homogeneous Experimental Facility in JAERI were fabricated. In preparing these disks, satisfactory quality controlling with respect to homogeneity and accuracy was carried out. Special care was also taken to prevent the generation of radioactive dust, which resulted in success from the viewpoint of accountability and safety. These fuel disks were charged into the critical assembly at JAERI, and criticality was successfully attained with little deviation from the calculated value. (auth)

Critical mass, spectrum, breeding ratio, and coolant removal coefficients were calculated for a series of large Pu-U\textsuperscript{235}-fueled sodium-cooled fast-breeder power reactors, using a new 16-group cross-section set based in part on recent microscopic cross-section measurements. The parameters studied include reactor size, plutonium isotopic content, and type of structural material. Reactors cooled with Pb-B\textsubscript{12} eutectic and those containing U\textsuperscript{235}--Th fuel were examined. (auth)


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


Critical parameters for slightly enriched uranium rods in light water are presented. The parameters include minimum critical mass, critical slab thickness, critical radius, total thickness, and minimum critical mass per unit area. The variables are water-to-uranium volume ratio, uranium rod diameter, and U\textsuperscript{235} enrichment. The calculations are based on a semi-empirical method for calculating material fissioning derived from a correlation of theory and experimental measurements. The calculated results at 2 and 3.065% U\textsuperscript{235} are within 3% of the experimental measurements. (auth)


Experimental data obtained for the purpose of determining criticality parameters of fissionable materials outside reactor environments are surveyed, with emphasis on U\textsuperscript{235}, U\textsuperscript{238}, and Pu\textsuperscript{239}. The data on U\textsuperscript{235} are mainly in the enrichment ranges <5 and >90%. The data on U\textsuperscript{238} and Pu\textsuperscript{239} are restricted to metallic units and aqueous solutions within a small range of chemical concentrations. The assemblies studied are assumed to be spherical. The operating conditions assumed are those expected in chemical and metallurgical processing plants. (T.F.H.)


The critical sizes were determined experimentally and compared with the calculated results. The effect of non-fissile structural materials was studied. (auth)
Critical Experiments 181-187


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


Critical mass experiments were continued with plutonium concentrations ranging from 36.3 g/l at 3.3M HNO₃ with a 4-in. concrete reflector to 238.0 g/l at 5.21M HNO₃ with a 1.0-in. paraffin reflector. The corrections for reflector savings and nitrate ion concentration were derived from the experiments, and the experiments reduced to equivalent, bare critical systems. Data for evaluating the reflector savings of stainless steel were obtained. The worth of stainless steel as a reflector in combinations of stainless steel and paraffin were estimated. A coefficient was derived relating the nitrate--ion concentration to the critical volume. (M.C.G.)


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


A comparison was made of the measured and calculated critical masses for a water-reflected core, a graphite-reflected core, and a graphite-reflected core with an internal water thermal column. Measured values are given of the worth of the absorber rods in the first of these assemblies, and of one rod over a limited range in the second assembly. Values are also given for the void coefficients and reactivity effects of a fuel assembly and graphite expellers obtained. (auth)


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

The influence of temperature on the critical buckling of bare graphite assemblies with various carbon-to-uranium-238 molar ratios has been measured. A range from 1188:1 to 21,690:1 was covered, for 45 to 1200°F. Preliminary results indicate that the fractional rate of change of critical buckling with core temperature varies monotonically with C/U^{235} ratio by a factor of five over the factor-of-eighteen range in gross C/U^{235} ratio. This quantity appears to approach asymptotically a value near 2%/100°F at very high C/U^{235} ratios. (auth)

3856 (ORNL-31376(131-2)) CRITICAL PARAMETERS OF SOLUTIONS OF U^{235}-ENRICHED URANYL NITRATE IN CYLINDRICAL CONTAINERS. J. K. Fox (Oak Ridge National Lab., Tenn.).
The critical dimensions of cylindrical Al vessels containing enriched solutions of UO_2(NO_3)_2 are determined. At 93.15 wt % U^{235}, the solvent is a D_2O-H_2O mixture containing 70.1 wt % D_2O. At 92.6 wt % U^{235}, the solvent is H_2O. The cylinder diameters vary from 24 to 51 cm, and the critical heights from 19 to 72 cm. The data are taken for unreflected, B-D_2O-reflected, and H_2O-reflected cylinders. (T.F.H.)

Critical studies were performed with a metallic, fast reactor core designed to investigate the effects of replacing highly absorbing U^{238} diluent with high-scattering, low-absorbing sodium diluent. The fuel was 15.7 wt % enriched U^{235} and the core contained 18.2 vol % sodium and 12.68 vol % stainless steel. The experimental program was designed to measure the effect of the material replacement on spectral indices, which consisted of the standard fission ratios, foil irritations, and a large number of central reactivity coefficients. Other measurements included the Rossi-8, radial fission traverses, and edge reactivity worths of a few samples. (auth)

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

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

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

To understand the reactor physics in enriched-fuel, graphite-moderrated thermal reactors theoretical and experimental investigations were performed. A critical assembly was used to obtain the criticality data. An approach
Critical Experiments 195-200

195.


A series of critical experiments was performed on the system in which heavy water reflector surrounded heavy water solution of uranyl sulfate with a U235 enrichment of 20%. Heavy water molecule to U3H atom ratio in the solutions ranged from 3600 to 800 depending on the core diameter. Space dependences of thermal neutron spectra in these systems were studied by the integral method with La. Deviations of epithermal neutron spectra from 1/E distribution were also investigated by the cadmium ratio method with In, Au, Pd, and Co. In theoretical analysis of these systems, leakage of fast neutrons from the core and competition of the leakage with resonance absorption in the core are important factors. Therefore the resonance escape probability was defined rigorously and a multigroup model was applied. The group constants were determined by averaging over spectra which were calculated by the Greuling Goertzel approximation. Agreement between theoretical and experimental results are satisfactory except perturbation terms. Discrepancies in the effective multiplication factors do not exceed 1/2. Spatial distributions of the thermal, in-resonance, and fast neutron fluxes are well reproduced by the present theory. An agreement of the theoretical cadmium ratio in the core with the experimental values indicates that the leakage of fast neutrons from the core is treated adequately. (auth)

196.


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

200.


Experimental data were collected on over 70 light water moderated, fully enriched uranium, stainless steel, critical cores. An equation for the critical mass of cores with a buckling of 0.007 cm$^{-2}$ that is linearly dependent on stain-
Critical Experiments 201-204

less steel volume fraction and grams of B\textsuperscript{10} was compared with available critical experiments and found to yield reasonable results. A correlation method, relating buckling to \(\left(\mu_{n}/\mu_{o}\right)\) was found to fit the available experiments.

201.

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

Critical mass studies were made of plutonium nitrate solutions in a 14-in. dia stainless steel sphere. The Pu concentrations ranged from 30 to 231 g/l with acid molarities varying from 1.1 to 5.5 molar. The wall thickness of the spherical vessel is 0.044 in. Measurements were made with the vessel bare and reflected with \(1/2\) in. of paraffin, 1 in. of paraffin, 4 in. of concrete, and 10 in. of concrete. Four measurements were made with the sphere covered with 0.030-in. Cd sheet backed with reflector. Evaluations were made of the control rod worth at various concentrations. The data from the experiments are summarized. The additional experiments with paraffin reflectors were used to obtain improved estimates of the reflector savings of this material. Estimates for the reflector savings of the various reflectors were obtained by first correcting the critical volumes of the bare spheres for the effect of the stainless steel shell by 14 ml/ml of stainless steel; then by comparing equivalent spherical radii, a reflector savings for the vessel wall and added reflector could be calculated from the measured volumes. The equivalent sphere volume for those cases in which the vessel was subcritical and full was taken as the extrapolated critical volume; the equivalent sphere radius for a partially filled sphere was estimated on the basis of equal surface-to-volume ratios for equivalent systems. These reflector savings are given for three concentrations of Pu. Control rod worth was evaluated during the critical approaches by means of multiplication measurements. The control rod consists of a stainless steel tube which is 1 in. in OD with a 0.065-in. thick wall; this tube can be inserted directly into the solution to within about 2 in. of the sphere bottom. A plot of control rod worth versus Pu concentration is included. The worth varies from about 60 ml of solution at high concentrations to about 560 ml at low concentrations. The control rod worth in terms of Pu appears to have a nearly constant value of about 15 grams over the entire range of concentrations measured, essentially independent of the H:Pu ratio.

202.


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

203.


204.


Two types of integral experiments undertaken to provide experimental checks on nuclear data and methods of calculation are described. A cylindrical assembly of material having a point source of DD or DT neutrons at its center is discussed. The neutron flux and reaction rates in the cylin-
Critical Experiments 205-208


Experiments were conducted to determine the limiting concentration (the concentration for which \( k_e = 1 \)) of aqueous plutonium nitrate and uranyl fluoride solutions. For the Pu experiment, the solutions were contained in stainless steel tanks. Because of the difficulty in evaluating the effect of the stainless steel on the measurements, the uncertainty in the measured limiting concentration is approximately \( \pm 1 \text{ g/l} \). The experimental data were corrected for the effects of the Pu\(^{239} \), Pu\(^{241} \), and nitrate present to give a value of 8.4 \( \pm 1 \text{ g/l} \) as the limiting concentration for Pu\(^{239} \) in an aqueous solution. Foil irradiations were made using Au, Cu, and Pu\(^{239} \) foils. Cd ratios of 4.83 for the Au and 14.89 for the Cu were obtained. The results of the Pu-U foil irradiations indicate an effective neutron temperature of about 390\(^\circ\)K, whereas the physical temperature of the solution was 299\(^\circ\)K. The 9% enriched UO\(_2\)F\(_2\) measurement was made using containers of Al, which had no measurable effect on the results. The experimental data gave a value of 12.94 \( \pm 0.03 \text{ g} / \text{U} (12.05 \pm 0.03 \text{ g} / \text{U}^{239} / \text{l}) \) as the limiting concentration of the solution used in the measurement. Au foil irradiations gave a Cd ratio of 8.6 in the center of the system. (auth)


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

208.


The infinite medium neutron multiplication factor, \( k_w \), of a mixture of 92.1 wt % UP\(_3\) and 7.9 wt % paraffin was measured both in the Physical Constants Testing Reactor at the Hanford Atomic Products Operation and in critical experiments at the Oak Ridge National Laboratory. The density of the mixture is 4.5 g/cc and the U\(^{235} \) enrichment of the uranium is 2.0 wt %, resulting in an \( \text{H}:\text{U}^{235} \) atomic ratio of 185. The values of \( k_w \) from the two experiments are 1.216 \( \pm 0.013 \) and 1.197 \( \pm 0.015 \), respectively. In the analysis of the critical experiments a two group model was assumed for the noneleapage probability. The neutron age to thermal was determined from buckling perturbation measurements as 43.1 \( \pm 3.4 \text{ cm}^2 \). The critical buckling was measured to be \( 41.7 \pm 1.0 \) x \( 10^7 \text{ cm}^2 \), the bare extrapolation distance 2.7 \( \pm 0.3 \text{ cm} \), and the fast fission factor 1.019 \( \pm 0.004 \). Within the experimental error, the values of \( k_w \) from critical experiments at ORNL and from the PCTR at HAPO agree. (auth)
Critical Experiments 209-215


210.


Critical masses of homogeneous water-moderated assemblies containing low enrichment uranium are given. The calculations were made using the multiplegroup NDF code with eighteen energy groups. Effective absorption cross sections for \(^{235}\text{U}\) were computed with the infinite mass and narrow resonance approximations. The calculations were compared with various experiments and rather good agreement was found. The results are presented as a parametric survey for \(^{235}\text{U}/^{233}\text{U}\) atom ratios from 0.014 to 0.300 and for all \(^{235}\text{U}/^{233}\text{U}\) ratios for which criticality is possible. The decrease in critical radius with an infinite water reflector is also shown. A bare homogeneous system with \(^{235}\text{U}/^{233}\text{U} < 0.010\) cannot be made critical at any \(\text{H}^{238}/^{233}\text{U}\) ratio. (auth)

211.


The complete spatial separation of moderator and uranium fuel bearing regions are shown by experiment to result in critical reactors with low critical mass and relatively uniform fissioning density. Studies of several of these experiments to establish the accuracy of a numerical method of calculation (SNG) for this class of problems show good correspondence between theory and experiment. This method is then used for a useful survey of critical mass and \(^{235}\text{U}^{233}\text{U}\) atomic density as a function of geometry for the best moderators, \(\text{D}_2\text{O}\) and Be. (auth)

212.


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

213.


The present phase of the advanced-epithermal-thorium reactor program consists of integral-reactor-physics experiments designed to provide neutron-cross-section information at 10 Mev to 1 kev range. Nine multi-region, slow-fast, pseudospherical critical assemblies of the honeycomb, split-table type are studied. Three assemblies were run. The outer driver-decoupler region drives an interior \(\text{U}^{233}\text{-Th}^{237}\) fueled spherical test region whose neutron-flux spectrum is successively degraded by increasing the graphite moderator to fuel ratio. A square-wave oscillator experiment defines the central reactivity worths of 40 small samples of different materials to \(10^{-8}\) \(\text{A}\) for each assembly. Intercalibrated artificial neutron sources are oscillated to determine central neutron importance functions. The spectra are obtained by fission-counter measurements using calibrated foils of different thresholds and by a \(\text{Li}^{7}\text{-solid-state-counter sandwich spectrometer. A digital computer routine is used to compile all measurements into a self-consistent library of spectrum-averaged cross sections.}"

(auth)

214.


Measurements were made of the reactivity changes caused by reflectors of various materials on BeO-mod erated, \(^{235}\text{U}^{233}\text{U}\)-fueled subcritical systems with a moderator-to-fuel ratio of 247. The reflector materials included aluminum, steel, copper, brass, type 304 stainless steel, nickel, René 41, Hastelloy R-235, and BeO. The pulsed neutron technique was used. The data are tabulated in terms of the subcritical time constant as a function of reflector thickness and surface density. (auth)

215.

10607 (HW-66829) \(\text{K}_0\) OF THREE WEIGHT PER CENT \(\text{U}^{235}\) ENRICHED \(\text{UO}_2\) AND \(\text{UO}_2\)(NO\(_2\))\(_2\) HYDROGENOUS SYSTEMS. V. I. Neeley, J. A. Berberet, and R. H. Masterson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Sept. 1961. Contract AT(45-1)-1350. 32p.

The value of the infinite multiplication constant \(\text{K}_0\) was determined as a function of the hydrogen-to-uranium \((\text{H}/\text{U})\) atomic ratio for 3.04 weight per cent \(^{235}\text{U}\) enriched \(\text{UO}_2\) and \(\text{UO}_2\)(NO\(_2\))\(_2\) hydrogen moderated homogeneous systems. The work was done to evaluate nuclear safety of and establish operational limits for slightly enriched homogeneous sys-
tems and was performed in the Hanford Physical Constants Testing Reactor. The amount of thermal neutron absorber, commonly referred to as poison, was determined by a least-squares fit to the experimental data. The curve of $k_{\text{eff}}$ versus $H:U$ was determined from measurements of the amount of thermal poison necessary to reduce $k_{\text{eff}}$ to unity as a function of the $H:U$ atomic ratio for various enrichments. This derived basic data for spherical masses and volumes was extended by a combination of experimental methods and the one-group buckling equations to the estimation of cylindrical geometry. Some suggestions for an experimental program to help fill in the uncertain parts of the curves were made.

Critical Experiments 216-221


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

12699 THE CRITICALITY OF HOMOGENEOUS SOLUTIONS AND MIXTURES. I. THE CRITICALITY OF URANIUM-WATER SYSTEMS. M. A. Perks (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). Progr. in Nuclear Energy, Ser. IV, 7: 244-52(1960). A comparison of two-group diffusion results with experimental data on spherical homogeneous uranium-water systems is made. Using extrapolation lengths derived from this comparison, predictions are made of the critical dimensions of infinite cylinders and slabs for all $H/U$ ratios using the equal buckling conversion.

12700 THE CRITICALITY OF HOMOGENEOUS SOLUTIONS AND MIXTURES. II. THE CRITICALITY OF PLUTONIUM-WATER SYSTEMS. M. A. Perks, F. R. Charlesworth, and D. E. J. Thornton (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). Progr. in Nuclear Energy, Ser. IV, 7: 244-52(1960). A correlation of existing experimental data on the criticality of plutonium metal–water mixtures was made. The gap in data between effective plutonium densities of 0.1 g/cm$^3$ and 19.6 g/cm$^3$ was filled in on the basis of reasonable assumptions as to the shape of the curve. This derived basic data for spherical masses and volumes was extended by a combination of empirical methods and one-group buckling equations to the estimation of cylindrical and slab critical dimensions. Some suggestions for an experimental program to help fill in the uncertain parts of the curves are made.
Critical Experiments 222-226

222.


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

223.


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

224.


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

225.


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

226.


The criticality of arrangements of fissionable material with various moderators cellulose is considered. The proposals of the International Parkinson Committee (ICP) essentially amount to the achievement of a self-limitation of the document situation in nuclear organizations of every kind by prescribing the exclusive use of writing paper containing a homogenous distribution of U$^{235}$. In the piling-up of U$^{235}$ impregnated cellulose sheets in the form of NSP (Nuclear Standard Paper) it is necessary to distinguish between the criticality of the U$^{235}$ embedded in the sheets acting as a moderator (Nuclear Criticality) and the conventional or bureaucratic criticality. Various safety considerations are to be observed in the case of the building up of critical masses of NSP. The use of blotting pads made of cadmium and gadolinium paper weights permits especially high piles. The NSP fuel element arises from the concentration of 300 NSP in up to 2 mm thick cladding, for which cardboard is used in place of zirconium. A subcritical assembly results from the piling up of such fuel elements in filing cabinets. This situation must be avoided because of the danger of a critical state through changes in the geometry, e.g., through tearing of the files. The same safety regulations apply to the store of useless fuel material (Archives). (auth)
227.

228.
28548 (TID-7623p.41-4) FAST POWER REACTOR EXPERIMENTS IN ZPR-3 WITH METALLIC AND CERAMIC FUELS. F. W. Thalgott (Argonne National Lab., III.).

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

229.

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

230.

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

231.

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

232.

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

233.

A critical studies program for the Enrico Fermi Atomic Power Plant was run with the ZPR-III fast critical facility. The objectives of this program included determination of the U enrichment required for criticality, the effect of minor variations in core and blanket composition, reactivity coefficients, control and safety rod characteristics, power distribution, spectral indices, and the reactivity worth and wave shape of the oscillator rod. The experimental program was separated into two phases. The first phase in-
Critical Experiments 234-238

volved investigations of a clean assembly, which was a simplifed and homogenized core and blanket geometry constructed for ease of experimental manipulation and analysis. The second phase involved experiments on the engineered, or as-designed, core. This assembly included such engineering details as control and safety rod channels, core end gaps, and a precise reconstruction of the core outline. This provided detailed information on worths of control rods and fuel subassemblies, power distribution, and the effect of variations in core and end-gap dimensions. The application of critical experiment data to the determination of the Enrico Fermi reactor characteristics has established the U235 enrichment for the fuel alloy, worths of fuel subassemblies, and the B10 enrichment for the control and safety rods. In addition, material-substitution experiments and fuel-worth measurements have provided the parametric data for the determination of the net temperature coefficient of the Enrico Fermi reactor. (auth)

234.


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

235.


The physics of U233-Th epithermal systems was studied. The Epithermal Critical Assembly is described, and the composition and sizes of 9 cores are listed. The calculations of both predicted and experimental data were made using a multiregion, multigroup diffusion code. Reactivity measurements, neutron importance, spectral measurements, and criticality analysis are discussed. (M.C.G.)

236.


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

237.


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

238.


To accommodate large metal fractions, but still retain the advantages of homogeneity, the Solid Homogeneous Assembly (SHA) was designed with the metallic elements and fuel as very fine powders dispersed in paraffin. A series...
of reproducibility checks performed with the assembly demonstrated that rearranging or replacing major segments of the core leads to reactivity changes of less than 1%. In addition to critical size determinations preliminary flux distribution and reactivity coefficient measurements were made. The calculated criticalities using the KARE and KLAG programs were compared with the experimental ones. It was found that calculations run approximately 2% high fairly consistently. (M.C.G.)


239.


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

241.


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

242.


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

243.


The design of the High-Flux Isotope Reactor (HFIR) was supported by a series of preliminary experiments performed at the Oak Ridge Critical Experiments Facility in 1960. The experiments yielded results describing directly some of the expected performance characteristics of the reactor and strengthened the calculational methods used in its design. The critical assembly, like the reactor, was of a flux-trap type in which a central 6-in.-dia column of H\(_2\)O was surrounded by an annulus of fissile material and, in turn, by an annular neutron reflector. The fuel region contained a solution of enriched uranyl nitrate in a mixture of H\(_2\)O and D\(_2\)O and the reflector was a composite of two annuli, the inner one of D\(_2\)O surrounded by one of H\(_2\)O. In most experiments the ends of the assembly were reflected by H\(_2\)O. Important results evaluate the absolute thermal-neutron flux to be expected in the design reactor and describe the flux distributions within
Critical Experiments 244-249

this type of assembly. It was also observed that the cadmium ratio along the axis of the assembly was about 100, showing that a highly thermal-neutron flux was truly developed in the trap. It was shown that reduction of the hydrogen density in the central water column to about 60% of its normal value increased the reactivity about 6% and that further hydrogen density reduction decreased the reactivity as the effect of the less neutron moderation dominated the effect of the increased coupling across the central column. These considerations are of importance to the safety of the reactor. Additional experiments gave values of the usual critical dimensions and explored the effects on both the dimensions and the flux distributions of changing the concentration of the uranyl nitrate solution, of changing the composition of the solvent, and of adding neutron-absorbing materials to the D₂O reflector. These changes were made to alter the neutron properties of the fuel solution over a range including those expected in the reactor itself. (auth)

244.

12748 (ORNL-3360p.35-6) CRITICAL EXPERIMENTS WITH MIXTURES OF AQUEOUS U⁹²⁸-ENRICHED URANYL NITRATE SOLUTIONS AND BOROSILICATE GLASS RINGS. J. K. Fox and J. T. Thomas (Oak Ridge National Lab., Tenn.).

Several critical experiments were performed with mixtures of aqueous 92.8% U⁹²⁸-enriched uranyl nitrate and borosilicate glass Rauchig rings to provide information for nuclear safety applications. The mixtures were contained in 20-, 30-, and 48-in. ID Al or stainless steel cylinders, and two concentrations of uranyl nitrate solutions were used, one containing 418 g of U⁹²⁸ and the other 279 g of U⁹²⁸. The glass rings varied in size, the largest having a 1.52-in. ID, a 1.85-in. OD, and a 1.89-in. length. Their B content was 0.5 to 5.7 wt%. It was found that for solutions having a U concentration ≤415 g/liter, kᵣ will be less than 1 if the glass rings contain 5.7 wt% natural B and occupy 24.1 vol % of the mixture. For solutions having a U concentration of ≤279 g/liter, the corresponding values are 4.0 wt % natural B and 24.1 vol % glass. (auth)

245.


246.


247.


A series of critical experiments was conducted on a system in which heavy water solutions of uranyl sulfate with a U⁹²⁸ enrichment of about 20% are surrounded by a heavy water reflector. The deuterium to U⁹²⁸ atomic ratios in the solutions ranged from 7,200 to 1,600. In theoretical analysis of the system, leakage of fast neutrons from the core and competition of the leakage with resonance absorption in the core are rather important. Therefore, the resonance escape probability was defined rigorously and a multi-group model was applied to fast and epithermal neutrons. The group constants were determined by averaging over a spectrum which was calculated by the Greuling-Goertzel approximation. The agreement between the theoretical and experimental results are satisfactory except those for perturbation terms. The discrepancies in the effective multiplication factors do not exceed 1%. The spatial distributions of thermal and in-resonance neutron fluxes are well reproduced by the present theory. Agreement of the theoretical cadmium ratios with experimental values indicates that the effects due to the leakage of fast neutrons from the core are treated adequately. (auth)

248.


Critical mock-up operations of the VVR-M reactor with a beryllium reflector and a thermal neutron flux 3 x 10¹⁴ n/cm² sec at 10 Mw(t) is described. A theoretical evaluation is given of the core configuration and its measurements. Calculations were made for various core configurations, and the results were compared with experimental data for minimum critical masses. Data are also given on safety techniques employed during critical operations. (tr-auth)

249.


Maximum safe batch sizes were determined for all process steps involved in the melting and casting operations, as well as in the subsequent fuel handling procedures, of 16 wt % U–85 wt % Zr fuel rods. Criteria for hydrided as well as unhydrided rods are presented. A method is presented for the determination of an equivalent fuel rod di-
ameter so that criticality data, developed for long rods can be used in establishing nuclear safety criteria for fuel rods of any length. Packaging criteria are presented for the safe storage and transport of the fuel materials in 30 x 30 x 30 in. birdcages as well as in 55-gallon drums. The criteria developed can be used to determine the maximum safe number of storage units in an array when each unit contains no more than 45% of the critical quantity of fuel in spherical geometry under optimum conditions of water moderation and reflection. This is independent of fuel composition. Thirty-three figures are given that have been used in the development of this criteria. (auth)

250.


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

251.


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

252.


Critical experiments were conducted with plutonium nitrate solutions in an 11.5-inch diameter stainless steel sphere. The measured volume of the 11.5-inch sphere was 12.95 liters; the vessel wall thickness was 0.049 inch. Criticality in the water reflected sphere was studied as a function of nitrate concentration. Plutonium concentrations were in the range of 5 to 431 g Pu/liter, with low acid molarity (less than 2 in most cases). The data obtained yield a curve of critical concentration versus nitrate, which may then be used to estimate the critical concentration for a homogenous plutonium water mixture in the absence of nitrate, or to evaluate the minimum mass of plutonium in the sphere. Experiments were also conducted to determine the effect of the stainless steel vessel wall on the criticality of the water reflected unit. Pulsed neutron source experiments were conducted concurrently with critical mass measurements. Based on the measured neutron lifetimes and the prompt decay rates observed at various solution levels in the 11.5-inch sphere, effective multiplication constants were determined as a function of the plutonium concentration and solution volume in the sphere. One of the purposes of the pulsed neutron source experiments is to examine the feasibility of using this technique for k_eff measurements on in-plant equipment. Multigroup diffusion theory calculations were used to compute criticality for some of the solutions as used in the experiments, with a comparison between theory and experiment being given in terms of k_eff. (auth)

253.


A small, lightweight, uranium carbide-fueled reactor with a beryllium reflector surrounding the core was mocked up as Assembly 40. It was determined that the presence of beryllium in the axial and radial reflectors did not endanger the safe loading and operation of the critical assembly. The actual experiment consisted of determination of the critical mass, measurement of the reactivity coefficients for a large number of fissile and non-fissile materials, the performance of radial and axial fission traverses, and measurement of central fission ratios. The effectiveness of the radial beryllium reflector as a control mechanism was determined and the k_eff-alpha was measured. (auth)
Critical Experiments 254-260

254.


Measurements were made with plutonium nitrate solutions and uranyl fluoride solutions (93.13% enriched UO₂F₂) to determine the concentration at which kₑq equals unity (the limiting critical concentration) for each of these solutions. The limiting critical concentration for Pu²³⁹ in an aqueous solution occurred at an H: Pu ratio of 3095 ± 100; this limiting ratio corresponded to a concentration of 8.6 ± 0.3 g Pu²³⁹/liter for the solutions used in the experiments. In conjunction with this plutonium measurement, the limiting critical concentration for U²³⁵ in an aqueous solution was measured; the result (12.05 ± 0.03 g U²³⁵/liter) was compared to a similar result reported by the Oak Ridge National Laboratory. (auth)

255.


The neutron spectrum for the ORALLOY fuel system with a critical mass of 2.5 kg plutonium was measured; the result (12.05 ± 0.03 g U²³⁵/liter) was compared to a similar result reported by the Oak Ridge National Laboratory. (auth)

256.


Two series of measurements were made to examine the coupling effects of rod control rods. From two to five rods were used. The measurements were performed on a BeO-moderated, U²³⁵-fueled system with a BeO/U²³⁵ molar ratio of 247. The pulsed neutron technique was used. (auth)

257.


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

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

258.


Criticality measurements on concentrated Pu solutions were performed using PuO₂-polystyrene cubes with a H-to-Pu ratio of 15. Neutron multiplication measurements were performed by arranging contrast of the cubes on a split table that were then moved together gave the critical mass of the nonreflected and Lucite-reflected assemblies. An evaluation of control and safety rods, both poison insertion and fuel removal, showed the desirability of fuel removal type rods. The neutron spectrum for the PuO₂-polystyrene compacts was strongly epithermal. (D.C.W.)

259.


A series of neutron multiplication measurements were made on assemblies of Pu metal (density 15.8 g/cm3) in order that a safe melting crucible could be designed for charges of 10 kg or more. Suitable ingot shapes and masses that could be rolled into sheets were determined. Final charge limits must be determined by making in situ measurements with the actual production furnace when the furnace and crucible are completed. (H.G.G.)

260.


Nuclear safety in hood 9-A depends on controlling the mass and geometry of the plutonium compound not confined by the processing equipment. It is specified that plutonium concentrations in plutonium oxalate slurries and filter cakes allow to accumulate in volumes in excess of 4 liters should be limited to a maximum of 1.0g Pu/cc in the process equipment. It was concluded that the critical mass of nonconfined plutonium may exceed 7 kg for dry, unmoderated plutonium compounds. Moderation of a reflected 2.5-kg plutonium mass could make it critical in an active volume of about 8 liters. (M.C.G.)
Critical Experiments 261-266


Fast and thermal neutron spectra in the core of a semi-homogeneous reactor were calculated at atom ratios 2,000 to 10,000 for carbon/U\textsuperscript{235} and moderator temperature of 1,200 to 1,800 K. With the obtained results, spectra and \( \eta \)-\( \beta \) values measured directly in experiments, effective \( \eta \)-\( \beta \) values and breeding ratios, which are weighted averages of entire core spectra, were obtained. The results are very similar with those of Chernick, et al., of BNL and the breeding ratios were over 1.15 even including \( \text{Xe}^{135} \) and \( \text{Sm}^{149} \) poisoning at neutron flux \( 10^{14} \text{n/cm}^2/\text{sec} \). It is concluded that solid fuel thermal breeders have possibilities to be realized, depending on the designs which should be very elaborated. (auth)


The effect of commercially available borosilicate glass Raschig rings on the criticality of aqueous uranyl nitrate solutions enriched in \( \text{U}^{235} \) was investigated. The natural-B content of the glass varied from 0.6 to 5.7 wt %, and the volume of the vessel occupied by the glass ranged from 20.9 to 30%. Results from exponential experiments, using a critical layer of solution above the column of solution-ring mixture as a neutron source, provided estimates of the material buckling of the mixture as a function of solution concentration, B content of the glass, and the glass volume present. It was shown, for example, that the buckling is negative (i.e., \( k_a < 1 \)) if glass containing 4 wt % B occupies more than 22% of the mixture volume, whereas the same concentration of glass containing only 0.5 wt % B results in positive values of the buckling except for solutions more dilute than about 72 g of U per liter (H : \( \text{U}^{235} \) w 360). (auth)


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

265. ANL-6923(p.14-45) LIQUID-METAL-COOLED REACTORS. (Argonne National Lab., Ill.). Studies of Assembly 45 in ZPR-III are described. Data on the critical material and physical parameters of ZPR-IX Assembly 4 are tabulated along with data on the central worth coefficients on several metallic and ceramic materials and separated W isotope samples. Investigations are reported on the properties of U-Pu-fissium alloys, Ti-V and Al alloys, radiation effects on Nb-Zr-clad U-Pu-fissium alloys, fabrication and properties of fuel elements for zero-power reactors, fabrication of Doppler test elements, development of Pu-U-C fuels, corrosion of Croloy in high-temperature air, reactivity of Cd-Mg-Zn alloys with U\textsubscript{235} at 600°C, distribution of Pu between Mg-Zn alloys and molten chloride salts, head-end treatment of refractory fuels, and sodium coolant chemistry. Other work is reported on development and operation of FBR-II and FARET. (J.R.D.)


The critical \( \text{U}^{235} \) mass for light-water-moderated, fully-reflected, slightly-enriched, oxide-fueled cores with \( \text{H}^{2} / \text{H}^{3} \) atom ratios of 1 to 5 were determined. Bucklings were obtained and compared with theoretical values. Calculations indicate that the buckling is lowered by about 3% for Al-clad vs unclad fuel. The \( \text{U}^{235}/\text{U}^{238} \) fission ratio was also measured and compared with theory. The disad-
Critical Experiments 267-272

...vantage factor and the epicadmium/subcadmium U235 fission ratio were measured by the integral technique. The disadvantage factors disagree with calculations by Pennington. (D.C.W.)

267.


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

268.


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

269.


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

270.


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

271.


The PROSERPINE homogeneous reactor consists of a tank, 25 cm dia and 30 cm high, surrounded by a composite reflector made of beryllium oxide and graphite. In this tank plutonium or 99% enriched uranium solutions, the fissionable substances being in the form of a dissolved salt, can be made critical. In varying the concentration of the solution, critical masses were studied as a function of the level of the liquid in the tank. The minimum critical mass is 256 ± 2 gm for plutonium and 409 ± 3 gm for uranium-235. In the range of the critical concentrations studied, the neutronic properties of fissionable solutions of plutonium and enriched uranium were compared for identical geometries. (auth)

272.


The limiting enrichment of uranium for criticality and the limiting atomic ratios of Pu-to-natural uranium con-
taining 0.5, 10, and 15% Pu²⁴⁰ were determined by Monte Carlo techniques for UO₂-water mixtures. The Monte Carlo results predict an infinite multiplication factor of unity for a 1.001 ± 0.026 wt % enriched UO₂-H₂O mixture. This compares favorably to the experimentally determined limiting critical enrichment of 1.034 ± 0.01 wt % U²³⁵ in the uranium. The Monte Carlo and multigroup calculations indicate that natural uranium UO₂-H₂O mixtures with a plutonium-to-uranium atomic ratio of less than 0.00116 ± 0.00012 should have a kₑ of less than unity when no Pu²⁴⁰ is present. The relative worth of plutonium atoms to U²³⁵ from a reactivity standpoint with no Pu²⁴⁰ present is 1.90 ± 0.14. This ratio changes rapidly with the percent of Pu²⁴⁰ that is present in the Pu. (P.C.H.)


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


Over fifty fast critical assemblies were studied in the Zero Power Reactor-III (ZPR-III). All of these assemblies were fueled with Pu²³⁹ and deuterium; the core volumes ranged from 2 to 660 liters and the critical mass from 30 to 890 kg of Pu²³⁹. The experimental characteristics of a representative group of 23 of these assemblies in which oxide, carbide, and metallic fuels were simulated were compared with calculated values. The parameters studied were critical size, central fission rates, prompt-neutron lifetimes and the reactivity effects of substitution of various materials at the reactor center. (auth)


Experimental and theoretical investigations on critical parameters of uranium salt aqueous solutions with 90, 10, or 5 per cent enrichment are described. These investigations were carried out for determination of nuclear safety conditions and for studies of homogeneous reactor physics. Critical volumes of the solutions were determined at various uranium concentrations in the range of 30–450 g/l. Extrapolation lengths and effective additions of various reflectors were determined during experiments. The experimental results obtained for cylinder and rectangular cores by transformation of geometrical parameters using the extrapolation length and effective addition of light water reflector, were recalculated for cores having shapes of a sphere, infinite cylinder, and infinite flat layers. (auth)
Critical Experiments 279-285

278.

24845  (EURAEC-992) PLUTONIUM RECYCLING.

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

279.


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

Tu thickness

(in): 0 0.695 1.76 3.525 3.925 9.0

Oy critical mass (kg): 51.9 36.2 26.5 20.5 19.75 17.35

In addition, the $M_c$ of Pu in a Tu tamper thickness of 4.6093 in. was measured and found to be ~6.28 kg. (auth)

280.


Separate abstracts were prepared for 10 of the 25 papers included; 9 were previously abstracted for ESA (17: 42006, 42219, 42223, 42229, 42230, 42233, 42236, 42238, and 42387). (R.E.U.)

281.


282.


Separate abstracts were prepared for 16 of the 25 papers included; 9 were previously abstracted for ESA (17: 42006, 42219, 42223, 42229, 42230, 42233, and 42236, 18: 1236 and 7889). (R.E.U.)

283.


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

284.

36701  (HW-81659(p.26-32)) AN ANALYSIS OF URANIUM-FUELED LIGHT WATER MODERATED CRITICAL EXPERIMENTS. R. D. Liikala, J. R. Worden, and W. A. Reardon (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A detailed analysis of five uranyl nitrate critical solution experiments and six 2.70-8 enriched-uranium-dioxide critical experiments has been made. The critical solution experiments were analyzed to determine the reliability of the uranium cross section data utilized, thereby eliminating one of the potential uncertainties in the analysis of the heterogeneous experiments. The description of the methods utilized in these analyses together with the results are described. (auth)

285.

HYDROGEN-MODERATED decay constants were determined by the pulsed-neutron technique, a function of buckling for unreflected parallelepipeds or a homoiocne extrapaultion distance that was extrapolation distance for one and the buckling values were computed using the theory calculations. (auth)

Cadmium screens, and plutonium concentrations of 42.3 to 104.1 g/l. of annular cylinders containing plutonium solutions. SUR DES CYLINDRES ANNULAIRES SOLUTIONS results are plotted for cylinders with light water reflectors, Centre d'Etudes Nucleares, Saclay). June 1964. 59p.

Michel Molbert, with Annular Cylinders Containing Pu,

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

Critical dimensions of systems, (TTT).

Be-containing systems must be further investigated because the currently available data do not agree with the theoretical calculations. The relationship between the neutrons originating from the Be(n,2n) reaction and from fission was examined by means of U-Be critical assemblies, built from 100 x 50 x 10 mm thick Be plates with a density of 1.8 g/cm³ and 100 x 100 x 0.5 mm fuel plates, consisting of U₂O₃ with a ²³⁵U content of 1.34 g per element, and fluoroethene. The slowing-down length was determined up to energy levels at which the transition to the thermal neutron spectrum and the fast neutron multiplication factor must be taken into account because of the fission and the Be(n,2n) reaction. The multigroup system of the neutron-physical constants used in the calculations yielded satisfactory results. The experimentally obtained value of τ = 90.1 cm², including the thermalization range from 1.44 to 0.2 eV, allowed the slowing-down process of neutrons in the Be to be defined more exactly. A value of γ = 1.12 was obtained, assuming the threshold value of the Be(n,2n) reaction is 1.65 MeV. The effect of this reaction must be taken into consideration when calculating for a Be-containing system. (TTT)
Critical Experiments 293-300

293.  
Critical masses, as they apply to the establishment of safe geometries, are altered by the involvement of hydrogen as a moderator. Theoretically, uranium compounds and hydrogen compounds which exhibit mutual insolubility make proportional contributions to the composite uranium density of any system wherein they are combined. The uranium densities of four such systems were determined experimentally and the results obtained support the validity of the theory. Certain fundamental considerations were examined, which are potential sources of error in the application of these physical principles. (auth)

294.  

295.  
The critical dimensions and Pu masses for arrays of PuO₂–polystyrene cubes with a Pu concentration of 1.14 g/cm³ and a H/Pu atomic ratio of 15 were determined using an approach to criticality technique with a split table. Lucite reflectors were used in most measurements. Temperatures in the core regions were also measured, and neutron lifetimes in the arrays were obtained by noise analysis and pulsed neutron experiments. Additionally, temperature coefficients were measured, and flux traverses were carried out. (D.C.W.)

296.  
22850 (HW-80029(p.31-7)) CRITICAL EXPERIMENTS WITH PuO₂–POLYSTYRENE COMPACTS. C. R. Richey, J. D. White, R. C. Lloyd, and E. D. Clayton (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).  
The compacts used in the experiments have a Pu concentration of 1.12 g/cm³ and a H/Pu atom ratio of 15. The density of the polystyrene is 0.931 g/cm³. The effects on the criticality of arrays of the rubberized plastic coating on each compact and of the fuel-removal safety and control rods were determined. Comparative reflector savings for water, Lucite, and a 3% enriched uranyl nitrate solution were measured. The effect of core density change on the criticality of a Lucite reflected prism was also investigated. (D.C.W.)

297.  
36783 (HW-81659(p.45-8)) CRITICAL EXPERIMENTS WITH PuO₂–POLYSTYRENE COMPACTS. C. R. Richey, J. D. White, R. C. Lloyd, and E. D. Clayton (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).  
Critical experiments on PuO₂–polystyrene compacts with a H/Pu atomic ratio of 15 are described. The Pu is 2.2% ²³⁵Pu. Effects of a lucite reflector are taken into account. (T.F.H.)

298.  
A survey of critical sizes of simple geometric shapes is presented from experiment and calculation. An extension to computational surveys is given from the DSN computing scheme with a standard set of multigroup input parameters, 47 references. (R.E.U.)

299.  
A parametric study of the critical dimensions of U(93.5%)–graphite–water spheres, slabs, and cylinders is presented. The data were obtained by application of the LASL DSN code and include five different loadings of uranium in graphite; the range of moderation caused by water is complete. Both unreflected and water-reflected systems are considered. (auth)

300.  
A summary is presented of the results of measurements made on the ZR-1 critical assembly. The purpose of the measurements was to provide a basis for research on organic moderators. Critical masses for various H/²³⁵U ratios were determined directly. Flux distributions were measured in various lattices, and the temperature dependence of reactivity in various lattices was investigated. (T.R.H.)
301.  


302.

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

303.

1965

304.

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

305.

FRO is a fast zero power reactor built for experiments in reactor physics. It is a split table machine containing vertical fuel elements, 150 kg of ²³⁵U are available as fuel, which is fabricated into metallic plates of 2% enrichment. The control system comprises 5 spring-loaded safety elements and 3 + 1 elements for startup operations and power control. The reactor went critical in February 1964. The first assemblies studied were made up of undiluted fuel into a cylindrical and a spherical core, respectively, surrounded by a reflector made of copper. Some experiments made on these systems are described. Primarily, critical mass determinations, flux distribution measurements and studies of the conversion ratio are dealt with. The measured quantities were compared with theoretical predictions using various transport theory programs (DSN, TDC) and cross section sets. The experimental results show that the neutron spectrum in the copper reflector is softer than predicted, but apart from this discrepancy agreement with theory was generally obtained. (auth)

306.
35604 (ANL-7045(p.1-23)) LIQUID-METAL COOLED REACTORS. (Argonne National Lab., III.)

Operation and maintenance of EBR-II are described, and design and development of FARET control drives, instrumentation, and static vessels are discussed. Investigations of Na-air reactions are described. FARET reactivity using carbide-fueled cores was investigated. Doppler experiments with ZPR-3 Assembly 45 are reported. Assembly #6 was reloaded in the ZPR-9 facility. Data on critical properties of assembly 6 are included. Design work on ZPPR is summarized. Development of refractory alloys for use in O₂-contaminated Na is discussed along with progress in fast reactor fuel processing. (J.R.D.)

307.
47989 (ANL-7089, pp 1-29) LIQUID-METAL-COOLED REACTORS. (Argonne National Lab., III.)

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

308.
25346 (HW-84366(p.53-4)) UNREFLECTED CRITICAL MASS OF Pu₂₃⁹ METAL. L. L. Carter (General Electric Co., Richland, Wash., Hanford Atomic Products Operation).

A series of Monte Carlo calculations was carried out to determine the critical masses of ²³⁹Pu and ²⁴⁰Pu in metallic form. A summary of the Monte Carlo results is included. (J.R.D.)
309.


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

310.


Nov. 8, 1963. Contract AT(45-1)-1350. 39p. Dep. (mn); $2.00 (cy), 1 (mn) CFSTI.

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

311.


Technical accomplishments and relevant activities performed in the Hanford Plutonium Critical Mass Laboratory since 1961 are reviewed. Several hundred critical mass determinations have been made with plutonium in a Critical Mass Laboratory. This facility was specially designed and equipped to handle plutonium. Nearly three-fourths of the experiments were performed with plutonium nitrate solutions; the others were performed with PuO2-plastic mixtures (solids) using a remote split-table machine. There have been no unusual incidents involving any of the critical assemblies during the conduct of these experiments. (auth)

312.


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

313.


Atomics International has constructed a series of multi-region fast-thermal critical assemblies as part of the Advanced Epithermal Thorium Reactor (AETR) program to obtain data on the physics of 235U-Th-fueled reactors. In each case, the central test region was of a composition appropriate to a full-scale core. The physics data obtained in six cores, where the test regions simulated the cores of fairly large fast reactors, were analyzed using a cross-section set previously used in an analysis of ZPR-III fast critical assemblies. These data are central fission ratios, relative reactivities of certain materials, and the relative importance of neutron sources of different energies. Trends are identified in the differences between calculated and measured data, and these are compared with similar trends found in the ZPR-III assemblies. (auth)

314.


The effect of the thermal neutron spectrum on the multiplication constant in calculations for graphite-moderated lattices is examined. The point in question is not the theory of calculating the spectrum but rather how to employ spectrum results once they are obtained. The procedure for utilizing experimental flux measurements is also examined, and formulas are developed consistent with those of a commonly employed flux-averaging procedure for lattices using theoretical flux results. The average neutron velocity for each region of the lattice cell appears explicitly in these formulas. Results of calculations are given for 6 lattice spacings for a typical uranium-graphite lattice. Flux measurements are available for 3 of these spacings. Two essentially different calculational methods are employed. One of these uses the same neutron spectrum in all regions of the lattice cell while the other uses a different spectrum in each region. The latter method gives a harder spectrum in the fuel than in the moderator. Comparison of reactivity differences between these methods shows that taking into account the spectral hardening in the fuel reduces the calculated reac-
This difference would be greater for blacker fuel elements such as those employed in typical reactor cases. The need for an accurate treatment of the thermal neutron spectrum is demonstrated. In particular, it is important to take into account the additional spectral hardening in the fuel relative to that in the moderator. The free-gas graphite scattering kernel with an effective carbon mass that is dependent on the graphite temperature, and perhaps on the lattice spacing, is adequate for this spectral hardening. (auth)

315.


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

316.


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

317.

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

Measured thermal-neutron spectra for several homogeneous, light water and polyethylene moderated systems, were compared to spectra calculated by theoretical methods presently in use. Average thermal group cross sections (chosen as cross sections for neutrons between 0 and 0.683 ev) were obtained by integrating energy-dependent cross sections over the measured spectra. These were then compared with the corresponding parameters obtained using theoretical spectra. [M.O.W.]

318.


A series of critical experiments and theoretical analyses were made on two-region systems in which a heavy water solution of uranyl sulfate with $^{35}S$ enrichment of about 20% is surrounded by heavy water, poisoned with B or mixed with $ThO_2$. $D$-to-$H$ atomic ratios in the core solutions ranged from 1000 to 4500, depending on the core diameter and the blanket concentration. The theoretical effective
Critical Experiments 319-325

Multiplication factor is decreased by treating the leakage of fast neutrons from the core rigorously and is increased by using spatial-dependent effective cross sections. These treatments are justified by the agreement found between the calculated and measured values for the Cd ratio and the thermal flux distribution. The disagreement seen in the effective multiplication factor between the theoretical and experimental values may be attributed to uncertainty in the resonance integral of 235U. The discrepancy in the effective multiplication factors does not exceed 1%, when the accepted value of $\lambda_1 - \lambda_2$ is reduced to a smaller value. (auth)

319.


The gaseous core cavity reactor is receiving continuing study as a possible basis for advanced performance nuclear propulsion systems. For this reason the Los Alamos Scientific Laboratory is operating a small cavity reactor, on a low priority basis, to accumulate information on some of the more important parameters for this type of reactor system. The LASL cavity reactor consists of a cylindrical cavity 40 inches in diameter and 40 inches in length, surrounded by a 20 inch thick D2O reflector. For the initial fuel loading, 93.1% 235U metal foils lined the surface of the cavity. The critical mass for this configuration was 5.7 kilograms of enriched uranium. Heating of the reflector resulted in increased critical mass. The reflector D2O was heated through a range of about 40 degrees centigrade above room temperature. Critical mass as a function of reflector temperature is given. Measurements to examine consequences of a gaseous fuel shift were made. The fuel penalty as a function of orifice diameter is also given. (J.F.P.)

320.


321.

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

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

Criticality parameters, developed by the use of conservative calculational methods, are presented for uranium carbide fuels. The basis for advanced performance nuclear propulsion conditions. A method is also presented for applying uranium metal-water data to other uranium fuels when no other criticality data are available. (D.G.W.)

322.


323.


A critical assembly scheme for a four-zone zero-power cylindrical reactor fueled with $\text{UO}_2\cdot\text{NO}_3$, acidified with nitric acid is described. Various concentration fuel delivered through polyethylene tubes maintained the same solution level in each zone. The radial neutron flux distribution in the active zone center was measured using indium tracers. The reactor reached criticality at the active zone level $39.6 \pm 0.1 \text{ cm}$ which corresponds to the critical load of $3250 \times 30 \text{ g} \text{U}$. The diagram of radial neutron flux measurements shows the presence of bursts in the thermal neutron distributions which is characteristic of the multipole systems. The coefficient of non-uniform fission distribution along the radius is equal to 1.19. The profiled fuel loading law leading to a constant mean energy release along the reactor radius was estimated on the basis of the obtained data and on the assumption of weak thermal neutron field variations. The diagram of thermal neutron flux distribution and energy release for the given conditions is included. (R.V.J.)

324.


325.


Critical experiments were performed with graphite moderated two-region cores consisting of a central test region fueled with 24 uranium carbide elements and a driver region fueled with uranium metal elements. The uranium carbide elements consisted of eighteen fuel pins within a stainless steel process tube with sodium filling the interstices to mock-up the coolant. Three triangular lattice pitches were studied. Within each lattice pitch, variations in proposed control channel materials and locations were studied. Results are given for two-region critical masses, values of $k_{eff}$ for subcritical loadings of the 24 uranium carbide fuel elements, and first ring and peripheral driver element worths. Methods and results are presented in detail in an effort to implement the analytical techniques used in design calculations for advanced sodium graphite reactor cores. (auth)
326.  


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

327.  

Critical experiments were continued with plutonium nitrate solutions (4.6% 235Pu) in spherical geometry. The experimental vessel in current use is a 15.2-in. dia stainless steel sphere (30.2 liter volume) having a wall thickness of 0.048 in. Experiments were conducted with the vessel reflected by water and concrete; experiments were also made with the vessel unreflected. The concrete reflector was separated from the vessel by a 3/4-in. air gap. During the course of the experiments, the effect on criticality of the vessel wall was evaluated. Gold foil traverses were made of a number of critical solutions to determine their respective critical bucklings. Also, pulsed neutron source measurements were made on a number of the assemblies. Neutron lifetime values are reported for the full bare critical spheres. (auth)

328.  

Studies intended for the determination of the neutron-physics characteristics of a critical assembly with BeO moderator are discussed. The geometry of the assembly, using flat 235U fuel elements, is described. Results of measurements are given for the dependence of the length of the active zone with three layers of fuel elements on the geometry, the variation of the geometrical parameter z on the number of elements in a layer at constant fuel concentration, and the distribution of thermal neutron density in a layer with three vertical fuel elements. The coefficient of fuel self-screening was calculated by various methods, including the Monte-Carlo method and the P$_3$ approximation, and compared with measurements. (M. J. T.)

329.  

A number of critical assemblies were studied for purposes of reactor safety and criticality evaluation. These were H$_2$O-moderated 235U, 239U, and 232Pu critical experiments, and D$_2$O, Be, BeO, and C moderated, enriched 235U critical experiments that provided simple parametrics and extremes in type. The atomic densities and dimensions directly useful for computational purposes are listed for fast-to-thermal-flux spectrum assemblies. (auth)

330.  
*31504* (LA-3221) REACTOR MINIMUM CRITICAL DIMENSIONS. Carroll B. Mills (Los Alamos Scientific Lab., Univ. of California, N. Mex.), Oct. 1959. Contract W-7405-eng-36. 52p. Dep.; $3.00(cy), 2(mm) CFSTI.

The parametric study of minimum critical reactor dimension as a function of moderator, fissile isotope, and size has been made, based on a consistent variety of critical experiments. Minimum critical size and mass have been computed for a range of concentration of 235U, 239U, and 232Pu for H$_2$O-moderated bare and reflected slab, cylinder, and sphere geometries, as well as corresponding results for 235U and heavier atom moderators D$_2$O, Be, BeO, and C. Some results are presented of the same sort for D$_2$O, Be, and C reflector-moderated reactors. (auth)

331.  
*40155* (LA-3229, pp 562-76) CRITICALITY. Mills, C. B. (Los Alamos Scientific Lab., Univ. of California, N. Mex.).

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

332.  

Measurements were made of the prompt neutron decay constant, alpha, for several subcritical assemblies. These measurements included the determination of both the de-
Critical Experiments 333-340

layed critical alpha and the variation of alpha with buckling. These near-homogeneous assemblies were BeO- or graphite-moderated and were fueled with enriched uranium (93.2% 235U). The following moderator-fuel molar ratios were studied: BeO/235U = 123, 247, 370, 493, 5850, and C/235U = 1145. A method is described by which the data can be used for a sensitive test of quality of the nuclear cross sections such as are found in multigroup diffusion calculations. (auth)

333.


Multiregion critical cores, built on a split-table assembly machine, were used to investigate reactor compositions typical of epithermal and fast power reactors. The design and operating features of the facility, constructed to study the Th-232U fuel cycle, are described. Criticality calculations were in good agreement with experimental values. (auth)

334.


From a review of recent experimental data for a homogeneous, hydrogen moderated and reflected U(4.9)O2F4 system, and correlation with 16-group transport theory calculation, it appears that the minimum critical, infinite length cylinder diameter is 12.1 inches for the hydrogen moderated and reflected U2F4 system. The maximum uranium density of the U2F4 is taken as 3.2 g. U/cc. Calculations indicate that the use of a 0.20 inch wall thickness stainless steel or nickel container may increase the assigned minimum critical cylinder diameter about 0.9 inch. (auth)

335.

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

Research summaries are presented on critical experiments using enriched U cylinders, critical dimensions of enriched aqueous UO2F4, or geometrically complicated enriched-U critical assemblies, critical experiments with enriched U, prompt-neutron lifetime in critical U cylinders HFIR critical experiments, reactivity effects of perturbations in critical experiments, and neutron multiplication in fissile material. References to publications in which the research is described are included. (J.R.D.)

336.


337.


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

338.


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

339.

42038  (BNWL-95, pp 27-35) CRITICAL MASS PHYSICS. CRITICAL EXPERIMENTS WITH PuO2-POLYSTYRENE COMPACTS 239Pu = 8.0 PERCENT, H/Pu = 15. Richey, C. R.; White, J. D. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

A series of critical experiments were conducted with homogeneous PuO2-polystyrene compacts hydrogen-to-plutonium atomic ratio = 15 containing both 2.2 and 3.0% 240Pu. Criticality was determined for a series of Plexiglas reflected rectangular prisms ranging from near cubes, to long columns, and to slabs; bare arrays of near-cubic geometry were also studied. Critical thicknesses were 16.09 ± 0.41 and 5.99 ± 0.10 cm, respectively, for the bare and reflected infinite slabs of PuO2-polystyrene containing 2.2% 240Pu. Corresponding values for the 8.0% 239Pu mixtures were 18.48 ± 0.41 and 7.38 ± 0.09 cm. The infinite slab thicknesses for an equivalent 239Pu-water mixture (H: Pu = 15, p = 1.62 g Pu/cm^3) were 11.66 ± 0.30 and 4.38 ± 0.08 cm, respectively, for the bare and water-reflected slab. Corresponding critical radii for infinitely long cylinders were 10.52 ± 0.16 and 6.54 ± 0.14 cm; radii for critical spheres were 13.81 ± 0.16 and 10.40 ± 0.17 cm. (auth)
341. 29524 CRITICAL DIMENSIONS OF WATER-REFLECTED SYSTEMS CONTAINING U-235-H2O-Zr.


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

The results of a calculational survey cover the critical sizes of homogeneous water-moderated systems fueled with 239Pu, 232U, and 235U (at various enrichments). Also included are 233U enrichment values of water-moderated, low-enrichment uranium mixtures for which \( k_{\infty} = 1 \). (auth)


Critical experiments are carried out with PuO2-polystyrene compacts at a H/Pu atomic ratio of 15 using a split-table device. The core material is composed of alternate 2-in. fuel cubes and 2-in. Plexiglas cubes. The core itself has a H/Pu atomic ratio of 35. Criticality characteristics are tabulated for the bare core and for various reflector and interface configurations and materials. (T. F. H.)

Critical Experiments 341-350

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

1966

The third core of the zero energy fast reactor FRO consisted of 20% enriched uranium diluted to 29 vol % with graphite and had a volume of 30 liters. Like previous cores it was surrounded by a thick copper reflector. Measurements of critical mass, control rod reactivities, fine structure flux variations, and conversion ratios are summarized. In particular, effects associated with the heterogeneous arrangement of the uranium and graphite plates are examined. (auth)

349. 6579 (BNWL-149, pp 48-56) CRITICAL MASS PHYSICS. (Batelle-Northwest, Richland, Wash. Pacific Northwest Lab.).
Three separate papers are presented along with their results. Foil activation studies in two plutonium critical systems. Monte Carlo calculation values for the critical mass of 239Pu as a function of hydrogen to fuel atomic ratios and 239Pu enrichment, and resonance absorber admixed in the moderator of heterogeneous arrays are the presented papers. (M. O. W.)

Critical experiments were made in 11 assemblies in ZPR-3, simulating metallic, oxide, carbide, and ceramic cores of fast reactors of various volumes with uranium fuel. The following characteristics of spherical assemblies were measured and calculated: critical mass (effective
Critical Experiments 351-355

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

351.

0779 THE ZERO-ENERGY EXPERIMENT FOR THE MULTI-PURPOSE REACTOR (MZR), Behrens, Ernst; Giesen, Arnold; Ritz, Hilmär (Siemens-Schuckert Research Center, Erlangen, Ger.). Siemens Rev., 33: 179-186 (Sep. 1965).

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

352.


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

353.

3369 (CEA-R-2814) ALECTO—RESULTATS DES EXPERIENCES CRITIQUES HOMOGENES REALISEES SUR LE REACTEUR A I.C.E.P., BRUNA, J. S.; BRUNET; JEAN-GEORGES; BRUNET, JEAN-PAUL; CAIZERGUES, ROBERT; CLOUET D'ORVAL, CHRISTIAN; KREMSER, JACQUES; TELLIER, HENRY; VERRIER, PHILIPPE (Commissariat a l'Energie Atomique, Saclay (France), Centre d'Etudes Nucleaires). 1965. 130p. Dep. mn.

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

354.


The pulsed neutron technique was applied to the study of light-water-modulated homogeneous core assemblies. Using fissile materials in solution form it is easy to achieve large variations in geometrical buckling and moderation ratio. In the initial series of experiments, the fuel consisted of \( ^{235}\text{U} \) or \( ^{239}\text{Pu} \) in the form of uranyl nitrate. The solution concentrations used varied from 44 to 326 g of uranium per liter. Adoption of a coherent series of cross sections made it possible to deduce, from the variations in the prompt neutron decay constant as a function of geometrical buckling, data on the non-leakage probability and the slowing-down area. Effective cross sections calculated on the basis of assimilating light water to a secondary differential thermalaver were adopted. Interesting comparisons are made possible by the use of two fissile materials with markedly different \( \eta \) and resonance capture values. In a second series of experiments, devoted mainly to safety measures, the maximum permissible concentrations in various containers at processing plants were deduced by measuring the prompt neutron decay constants in weak plutonium nitrate solutions. (auth)

355.

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

The properties of \( ^{239}\text{Pu} \), \( ^{235}\text{U} \), and \( ^{233}\text{U} \) were compared on the basis of critical experiments carried out on ALECTO assemblies with essentially the same geometry. On the basis of the results obtained, an attempt was made to determine empirically the slowing-down area as a function of concentration; the multiplication factor was calculated by using, for the cross sections, a secondary differential thermalaverization model developed for light water. Attempts were also made to determine prompt-neutron lifetimes using a pulsed-source method and to obtain information on absorption and fission cross sections in a hydrogenated medium. After carrying out various geometrical variations which made it possible to go from critical dimensions in cylindrical geometry to critical dimensions in spherical geometry, the critical masses obtained in ALECTO were compared with others and with the French criticality guide curves. (D.C.W.)
356


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

357


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

358


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

Included are significant results obtained from criticality experiments with plutonium solutions and compounds and the analyses of the obtained data. Previously estimated curves for critical dimensions of homogeneous water moderated $^{239}\text{Pu}$ in the undermoderated range were revised. Computed curves are also given which show the effect of $^{240}\text{Pu}$ on criticality, together with experimental data points at 2.2 and 8% $^{239}\text{Pu}$. Measured decay constants from experiments on critical and subcritical systems are compared with computed values for $\text{H/Pu}$ atomic ratios of 15-106. An analysis of critical data on Pu solutions was made showing the effect of nitrate on criticality for reflected spheres with nitrate and Pu concentrations extending to several hundred grams per liter. The results from a subcritical neutron interaction experiment performed with cans of dry PuO$_2$ powder in a low density reflected cubic array are also presented. Aspects of the current criticality research on plutonium and of planned experiments are discussed. (M.O.W.)

359


The results of experiments and calculations on the critical parameters of aqueous solutions of the salt $\text{UO}_2(\text{NO}_3)_2$ at enrichments of 5%, 10%, 36%, and 90% are presented. Measurements were made over a wide range of uranium concentrations in the solution in cylindrical vessels, flat slabs, parallelepipeds, and spheres, with water reflector and without reflector. Extrapolation lengths and geometrical parameters $\gamma$ were determined. The relation between the critical parameters and the nitrogen content of the solution is discussed. The minimum critical dimensions of a flat infinite layer and an infinite cylinder were determined. Measurements were made with the addition of different reflectors—water, graphite, steel and beryllium. The experimental data are compared with calculations from the equations obtained using an empirical slow-down kernel. Multigroup calculations were carried out for the critical parameters of aqueous solutions in $\text{P}_0$ and $\text{P}_4$ approximations of the spherical harmonics method. Good agreement was obtained between the experimental and theoretical results. The results of experiments for determining the interaction between two, three and five subcritical assemblies, in the form of cylinders and parallelepipeds, in air and water are described. Critical parameters are also given for a system of many homogeneous subcritical assemblies with and without reflector. Critical parameters for interacting systems of complex form such as intersecting cylinders and flat slabs at different angles are obtained. The spatial distribution of neutron fluxes is studied for interacting sub-critical assemblies, with and without an intervening absorber, for different $\text{k}_\text{eff}$ ratios of the interacting assemblies. (G.O.Y.)

360

6587 (NASA-TN-D-3097) CRITICAL MASS STUDIES WITH NASA ZERO POWER REACTOR II. I. CLEAN HOMOGENEOUS CONFIGURATIONS. Fix, Thomsus A.; Hudler, Robert A; Ford, C. Hubbard; Alger, Donald L. (National Aeronautics and Space Administration, Cleveland, Ohio, Lewis Research Center). Nov. 1965. 20p. CFSTI S1 cy; $0.50.

The NASA Zero Power Reactor II (ZPR-II) was used to determine experimentally the critical masses for more than a tenfold range of highly enriched (93.2 percent $^{235}\text{U}$) aqueous uranyl fluoride fuel concentrations in clean cylindrical geometries. The ZPR-II reactor tank permits the assembly of cylindrical cores 76.2 centimeters (30 in.) in diameter and with lengths (heights) up to 90 centimeters either bare or radially reflected by about 15 centimeters (6 in.) of water. The specific range of fuel concentrations was from hydrogen to uranium-235 atom ratios of about 150 (167 g $^{235}\text{U}$/liter of fuel solution) to 1650 (15 g $^{235}\text{U}$/liter). In addition to the critical masses, data are presented on the temperature coefficient of reactivity and the incremental reactivity worth at criticality for a similar range of concentrations. Some of the physical properties for the fuel concentrations are presented. A brief description of the ZPR-II and the experimental procedures used are also included, (with)
Critical Experiments 361-368

361.
34749. CRITICAL PARAMETERS OF URANIUM (1.95)
METAL CYLINDRICAL ANNULLI. Johnson, E. B. (Oak
9: 185-6 (June 1966).

362.
2995. THE CRITICALITY OF 24Cm. Keshishian,
Vahe; Otteswite, Eric H.; Dunford, Charles L. (Atomica

363.
35165. THE SORA CRITICAL EXPERIMENTS,
Kistner, Gustav; Mihalczo, John T. (Oak Ridge
1966).

364.
18053. ON THE COMPARISON OF THE THEORETICAL
AND EXPERIMENTAL PARAMETERS FOR HOMOGENEOUS
URANIUM-WATER CRITICAL ASSEMBLY. Kochenov, A. S.;
(In Russian).

The P1 approximation of the neutron transport equation
and the single-velocity thermal-neutron diffusion equation
were used for the calculations; the energy range was di­
divided into 12 groups, including the thermal range for the
slowing-down equation. The critical assembly used for the
experimental determinations consisted of 70 x 35 x 250
mm holders, with 250 x 70 x 2.7 mm fuel sheets pressed
from polyethylene and $\text{U}_3\text{O}_8$ with a $\text{U}^{235}$ content of 90%.
Foils of Al, Cu, and stainless steel were used as cover­
ing. It was found that the assembly was quasi-homoge­
neous when the hydrogen and $\text{H}^{235}$ concentration ratio was
$\text{H}/\text{H}^{235} \leq 50$. For a value of this ratio of about 50, the
water gap between the foils amounted to about 5 mm, or
equal to the mean free path of the thermal neutrons in
water; therefore at >50, the effect of the heterogeneity
must be taken into account, and the method may be used
only in the first case. For the calculation it was assumed
that the assembly was spherical, with a 50-mm thick water
reflector. Comparison of the results revealed agreement
between theory and experiment. Thus, this method of cal­
culation may be used for determining the critical dimen­
sions of homogeneous, epithermal reactors using a hydrog­
enous moderator. (TTT)

365.
28721. MINIMUM CRITICAL MASS FOR LIMITED
URANIUM CONCENTRATIONS. Kochurov, B. P. At.

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

366.
16721. CRITICALITY STUDIES WITH PLUTONIUM
SOLUTIONS. Lloyd, R. C.; Richey, C. R.; Clayton,
(BNL-SA-232).

A series of criticality experiments were performed with
plutonium (4.6% $\text{Pu}^{239}$) nitrate solution in stainless steel
spheres of 11.5-, 14-, and 15.2-in. diam. Reflectors of
water, concrete, paraffin, and stainless steel were used;
experiments were also performed on the 15.2-in. sphere
unreflected. The spheres were made critical with pluto­
nium concentrations varying from 24 to 435 g Pu/liter and
molality varying from 0.2 to 7.7. The minimum critical
volumes for $\text{Pu(NO}_3\text{)}_4$ in water containing 4.6% $^{240}\text{Pu}$ were determined to be about 22 and 11 liters, respectively, for bare and reflected spheres at a concentration of 175 g Pu/liter. The effect of a 0.030-in. cadmium shell or a 4-in. air gap between the reflector and the vessel reduced the reflector worth to that of a nominal reflector (1-in. of water or less) for the concentrations of plutonium measured. Comparisons were made between experimental and theoretical results using multigroup diffusion theory. (auth)


Exponential experiments were carried out to give the materials buckling of a number of near-homogeneous $^{235}\text{U}$/aluminum alloy fueled systems having fertile oxides intimately mixed with the $\text{BeO}$ moderator. Relative fission rates of $^{235}\text{U}$, $^{238}\text{U}$, and $^{238}\text{Pu}$ were also measured in the equilibrium spectrum region of each assembly. Five assemblies having 5 wt % natural uranium oxide in $\text{BeO}$ were investigated for a range of $\text{BeO}/^{235}\text{U}$ atomic ratios from 1500:1 to 5700:1. A similar range covering four assemblies was examined for 5 wt % thorium oxide in $\text{BeO}$. A comparison of the experimental results with diffusion theory calculations is included. (auth)


These experiments form part of a series designed to obtain information against which the nuclear data for $^{235}\text{U}$ and $^{239}\text{Pu}$ can be checked over the energy range 10 to 500 keV. Seven fast reactor assemblies are described. All the cores contained $^{235}\text{U}$ and graphite, and some also contained $^{237}\text{Pu}$ and $\text{H}_2$. Detailed measurements were made on four of the assemblies but for the other three assemblies the critical masses only were measured. (auth)


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


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


376. 6503 (ORNL-3858(Vol.1), pp 13-19) CRITICAL EXPERIMENTS. (Oak Ridge National Lab., Tenn.)

Summaries of 14 various critical experiments are presented. Experiments were carried out with uranium dioxide, uranium-metal, uranium-molybdenum, and uranium nitride solutions. In addition, criticality studies of various reactor lattices are reported. (M.O.W.)
Critical Experiments 377-385

377.
From American Nuclear Society Meeting, Denver.
The research program for producing 244Cm in the Savannah River Production reactors is discussed. The feed material for the program is primarily 235Pu. After the concentrated fuel was fabricated into fuel assemblies, it was used for critical experiments. A brief discussion of these experiments is presented. (M.O.W.)


378.
A discussion of criticality control is presented to encourage a working knowledge of the part of those who design and perform operations with fissile material. Responsibilities of the AEC and of operating organizations and an outline of nuclear-safety experience lead to reasonable principles of nuclear safety. Next, empirical criticality information is presented to help develop a feel for conditions to be avoided during operations. The final portion covers criticality-control methods that are consistent with the stated principles and available criticality data. (auth)

379.
30858 CRITICAL AND EXPONENTIAL EXPERIMENTS WITH PLUTONIUM. Pluemee, Karl E. Power Reactor Technol., 8: 211-14 (Fall 1965).
A brief survey of experiments on plutonium-bearing assemblies is presented. (D.C.W.)

380.

381.

382.
The data and results of the single region UO2 and PuO2-UO2 critical experiments performed under the Xanadu Plutonium Program are given. These results include experiments made to determine critical configurations, reactivities, power distributions, flux shapes, reactivity coefficients of various perturbations and temperature coefficients. (auth)

383.
Data are tabulated for homogeneous and quasi-homogeneous molten-salt-fueled reactors. The development and state-of-the-art of this type of reactor is discussed. Data are given for 48 one-region and 7 two-region reactors. (ref references. (T.F.H.)

384.
From IAEA Symposium on Criticality Control of Fissile Materials, Stockholm.
Criticality parameter measurements on various combinations of fissile, moderating, and scattering materials are reported. The systems studied have simple geometrical configurations. The performance of various computational methods for predicting the criticality of these systems is evaluated. (T.F.H.)

385.
As a part of the continuing criticality program with uranium enriched to 5% in 235U, a comparison between theory and experiment was made with regard to the criticality of a volume of aqueous U(4.98)O2F2 solution (H235U = 356) contained in a thin-walled stainless steel cylinder. The solution had a height-to-diameter ratio of about 2.5, thus making the end leakage reasonably small compared to the radial leakage. Computations indicated that the end leakage from the finite system corresponded to only 1.5% reactivity. One-dimensional calculations, with an approximate correction for end leakage, are therefore valid. The Carlson S4 (DTF) code with the Hansen-Roach 16-group
cross sections was used for the calculation. The computed
multiplication factor of 1.002 for the critical system lends
further confidence to the method and to the cross section
set. (auth)

1967

386

557 (ANL-7044) CRITICAL STUDIES FOR THE FRENCH
FAST REACTOR "RAPSODIE" (ZPR-3 ASSEMBLY 44). Hess,
A. L.; Keeney, W. P.; Caumette, P.; Boyer, J. P. (Argonne Na-
38. 73p. Dep. mn. CFSTI $3.00 cy, $0.65 mn.

Criticality studies were carried out on ZPR-3 with a mockup of
the French fast reactor, RAPSODIE. Critical data pertinent to the
fuel specification for RAPSODIE, the evaluation of the RAPSODIE
design control systems, and the collection of other data useful in
verifying calculational techniques were acquired. The RAPSODIE
core simulated in the mockup was approximately a 40-liter cy-
linder. The experiments included a physics-core and an engineer-
ing-core study. For the physics-core study, a simple core-blanket
arrangement was constructed, and spectral indices and local re-
activity coefficients were measured. In the fuel for the physics-
core mockup, the ratio of U to Pu was about 5:1. In the fuel for
the engineering core, the U-to-Pu ratio was reduced to 3.2:1. Mockups of the RAPSODIE axial regions, steel-Na end gaps, and
different axial blankets were constructed with the engineering
core, along with mockups of the RAPSODIE control and safety
rods. Experiments in the full engineering mockup included rod-
worth studies, material-reactivity-coefficient measurements, and
traverses of neutron detectors and fuel-reactivity samples. This
was the first ZPR-3 critical assembly to contain a significant load-
ing of Pu. Heat generation from alpha decay of the Pu affected the
accuracy of reactivity measurements, and the neutron source from
spontaneous fission required different techniques for measuring
reactor periods. (auth)
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2. LATTICES

1956

1. 8947 AEC-4229
Brookhaven National Lab., Upton, N. Y.

R E F. IN URANIUM—WATER LATTICES. T. Auerbach.

Theoretical values of k_e for slightly enriched U—H_2O matrices are determined and compared with available experimental values. (D.E.B.)

2. 4912 AERE-N/R-134
Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England.


The critical size of a pile built up of a particular arrangement of fissile material and moderator can be related to the Laplacian of the arrangement. In these experiments the Laplacian has been measured for three arrangements in order to check the effect on critical size of a thin annulus of water around the U rods, and a 2 cm. wide annulus of air around the U rods. The arrangement studied consisted of U rods of 1.36-in. diameter spaced at 9-in. intervals in a block of graphite 10 ft., 8 in. by 10 ft., 8 in. by 1 ft high, the U rods being enclosed in concentric Al tubes so that water could be placed in the annular space between the tubes. The method of measurement was to determine thermal-neutron densities throughout the structure when a point source of neutrons was placed at the base. Indium foils were used to detect the neutrons and the experimental density distributions were fitted to the expected theoretical form. The constants necessary to give a least square fit were used to reduce X^2. The results obtained were: no water present, with air-gap X^2 = 0.83 ± 0.03 m^4; no water present, without air-gap X^2 = 0.99 ± 0.02 m^4; water present, without air-gap X^2 = 0.58 ± 0.02 m^4. (auth)

3. 6372 BNL-1578
Brookhaven National Lab., Upton, N. Y.

ANALYSIS OF THE CLEAN BUCKLINGS OF 1.3 PER CENT ENRICHED URANIUM-WATER LATTICES. J. Chernick.

4. 1922 AEC-3877
Hanford Atomic Products Operation, Richland, Wash.

EXPERIMENTAL PILE MEASUREMENTS IN GRAPHITE-URANIUM LATTICES. E. D. Clayton. June 1, 1954.

The results from a series of exponential experiments involving 30 graphite—uranium lattices are presented. The diameter of the slug was varied from 0.325 in. to 1.65 ft, and measurements were also taken with one hollow slug (1.66 in., OD, 0.81 in. ID). The lattice spacing was varied from 1/4 in. to 15 in. Both wet and dry lattice measurements were taken; that is, the lattices were measured with water and without water in the cooling annulus to determine the effect of the cooling water on the buckling or reactivity. The buckling values are given for various slug sizes and lattice spacings. The lattice diffusion lengths, utilization in the moderator, utilization in uranium, and multiplication constant are also listed for those lattices in which these quantities were determined. Calculated conversion ratios are given for all except the hollow slug size. (auth)

5. 5974


Exponential pile experiments designed to check theoretical calculations an U-graphite lattice buckling are described. (D.E.B.)
LATTICES 6-12

6. 3391 BNL-1785
Brookhaven National Lab., Upton, N. Y.

Exponential measurements on H2O-actually enriched U lattices are reported. A volume ratio of 3:1 and U enrichment of 1.15% is used. (D.E.B.)

7. 3392 BNL-1916
Brookhaven National Lab., Upton, N. Y.

Pile parameters for a natural U-H2O lattice having a 1.5 H2O-to-U ratio were measured. The best M2 value for this pile was found to be 33 cm2, giving a k of 0.989. (D.E.B.)

8. 8662 NAA-SR-1535
Canoga Park, Calif.

Buckling and intracell flux measurements have been made on three graphite moderated exponential assemblies each having seven rod enriched U fuel clusters. Theoretical values of the buckling are calculated and compared to the experimental values. In evaluating the resonance escape probability, two values are used for the lethargy spread of resonance neutrons. The larger value of 5.6, which appears in most of the literature, compares favorably with the experimental values than does the smaller value 2.6. The theoretical buckling compares favorably with experimental values for the two larger lattice spacings and is about 10% per cent low for the smallest lattice spacing. (auth)

9. 3038 BNL-1627
Brookhaven National Lab., Upton, N. Y.

Three methods are described for measuring the buckling of light-water lattices. In method 1, a sub-critical, cylindrical reactor lattice is placed in the neutron field of the Brookhaven Reactor thermal column. The vertical relaxation length (L) of the neutron-excited thermal neutron flux is then measured as a function of the loaded radius (R), and best values of B2 and λ (reflector savings) are found which fit the calculated functional dependence of L and R to the measured data. In the second procedure, that of the standard exponential experiment, B2 and λ are determined from plots of the measured radial and axial variation of the thermal neutron flux, determined by foil exposure. Method 3 is essentially that of the critical assembly. U is loaded in the presence of a Po-Be source until a k of 0.99 is attained. The source is then removed, and the lattice is partly unloaded by steps. Foil counting methods and experimental procedures for obtaining relaxation lengths and radial traverses are explained. Tabulated buckling measurements are given for 1.15 and 1.3% enrichment factors at 4:1, 3:1, 2:1, 1.5:1 volume ratios. Graphs for B2 and λ vs. volume ratio and spontaneous fission fluxes vs. R are also included. (K.S.)

10. 3229 BNL-1812
Brookhaven National Lab., Upton, N. Y.

A study was made of pile core parameters for light water moderated, slightly enriched uranium rod assemblies. This information is provided by measurements in a series of exponential assemblies which differ in uranium enrichment, moderator-to-fuel volume ratio, and rod diameter. The enrichment range explored varies from 1.3% to 1%, the rod diameters vary from 0.600 to 0.250 in., and the volume ratios lie in the range from 4:1 to 1:1 (and in some cases are even smaller). The quantities measured were f, p, B2, M2, and reflector savings (since the assemblies are reflected). Similar measurements reported at the last Reactor Information Meeting were done with 0.750” diameter rods with 1% nominal enrichment. (auth)

11. 3398 BNL-2184
Brookhaven National Lab., Upton, N. Y.

12. 4936
REACTOR PARAMETERS OF A LIGHT WATER—NORMAL URANIUM LATTICE. H. Kouts. G. Price, K. Downes, R. Sher, and V. Walsh (Brookhaven National Lab.,
15. 13172 NAA-SR-140
North American Aviation, Inc., Downey, Calif.

An experimental determination of the mean neutron temperature in the fuel rods of a natural U-D₂O lattice using rods 1-in. in diameter on two different cell spacings is reported. The temperature of the neutrons was found to be 35° ± 25°C and 33° ± 16°C above ambient room temperature for the two lattices. The experimental technique used was to compare the flux distributions in a unit cell in (1) a normal hot multiplying lattice and (2) a single U slug in a cold, Pb-Cd alloy, non-multiplying lattice. This allowed a comparison to be made between diffusion length and absorption cross section of the U in the two cases, and accordingly, a comparison between the mean temperatures of the neutrons in the two lattices. (auth)

16. 9983 BNL-418
Brookhaven National Lab., Upton, N. Y.

Values of buckling and reflector savings are given for boron-poisoned lattices of 0.387-in. diameter, 1.15% enriched U rods. Fast effects values for the same diameter rods in H₂O are given. The distribution of La between the fused-salt eutectic MgCl₂-KCl-NaCl and a U-Bi-La alloy was measured. Further work with EMF concentration cells on the Na–Bi system has yielded activity coefficients for Na in the concentration range of 0 to 0.20 mole/fraction. These coefficients increase slightly with concentration and are of the order of 10⁻⁵. On the basis of infrared spectra obtained on pure liquid samples of BrF₃, Br₂F, and UF₆, it appears that the Perkin-Elmer infrared analyzer can be used to provide continuous analysis of the circulating dissolver stream of the pile plant for the volatile fluoride process. The solubility of Ti in Bi is given by the equation log₂⁺₁₅ ppm Ti = 2250/T + 0.08. It is shown that the product of the solubilities of Cr and Fe is constant at a fixed temperature. (For preceding period see BNL-380.) (M. H. R.)
Lattices 19-22

in light water are presented. The buckling was determined for two rod sizes. The water-to-U-volume ratios were found. The information obtained with the preceding rod sizes and H2O/U-volume ratios was sufficient (when coupled with existing data for other rod sizes) to estimate the maximum obtainable buckling for 1.007% U in light water. A curve is given relating maximum buckling with H2O/U-volume ratio and rod diameter from which estimates may be made regarding criticality hazards with the 1.007% enriched U. The effect of replacing a water reflector with one of uranyl nitrate solution of two different concentrations. Some information was also obtained on the extrapolation length to be used in unreflected cases.

19. 4013 HW-40245(Del.)
Hanford Atomic Products Operation, Richland, Wash.

A new method of calculating resonance escape probabilities, results for the effective mass of a proton bound in the water molecule, and a comparison of similar techniques for calculating lattice constants developed independently by Hanford physicists and by a school of Russian physicists (as reported at the Geneva Conference) are reported. An estimate of maximum errors in \( f \) and \( \eta \) for thermal systems is described. Experiments and analysis thereof on the effects of neutron streaming in air channels through a moderator are reported. Buckling calculations and experimental results are given for graphite lattices employing 1.17-inch diameter natural U slugs and U235-Al alloy slugs. An experimental measurement of the critical mass of an annular cylindrical array is described. Some measurements on, and the status of, construction of the Lattice Testing Reactor are reported. Development of a BF3 counter suitable for operation at elevated temperatures is described. A method of determining the screening parameter in the Thomas-Fermi model of the atom is given for use in calculating atomic displacements produced by irradiations in crystals. Calculated results are reported for chain-reacting concentration limits of U-Pu-H2O systems.

19. 2116 HW-44525
Hanford Atomic Products Operation, Richland, Wash.

The Lattice Testing Reactor has been used to measure \( k_e \) for a number of lattices. The temperature coefficient of this reactor has been measured. End effects on the measurement of \( k_e \) have been investigated. A series of measurements of the bucklings of enriched (1% U235) U-graphite lattices are reported. Calculations are reported on the equilibrium parameters of Pu in a steady state U-Pu cycle reactor, on the reactivity effects of various coolants in a U-graphite lattice, on multiplication factors of homogeneous enriched U-water systems, and on neutron interaction between separated fissile systems. Measurements are reported on a precise determination of the U235 fission cross section with respect to Au197, on the slow neutron fission cross section of Pu239, and on the variation of \( \eta (Pu239) \) with neutron energy. An analysis of the Pu239 fission cross section is given. A series of measurements on the critical masses of enriched U slugs in water is reported and analyzed to determine safe masses for handling. The energy losses of fission fragments in the U crystal are calculated and results are given. (For preceding period see HW-42182.)

20. 1275 HW-43441
Hanford Atomic Products Operation, Richland, Wash.

The measurement of the multiplication factor of a lattice in the Lattice Testing Reactor is described and results given. Measurements are reported of the "favorableness" factors for this reactor. An experiment is described to measure that enrichment of uranium which will just permit unit multiplication in a water-U system. Buckling measurements are reported for graphite-enriched U lattices and for natural U lattices employing cluster-type rods. An exponential experiment to measure temperature coefficients of reactivity is described. Calculational results are given on the variation of lattice buckling with enrichment, on the thermal neutron flux in a cell and on the effect of graphite density on buckling. In theoretical reactor physics, a variational principle for multiple scattering is formulated, a new method for estimating fast leakage from bare reactors is outlined, and the spherical harmonic components of the flux in a void are computed. Experimental results showing that \( v \) for Pu239 does not vary in the thermal region are quoted. A nuclear safety problem involving the interaction of an array of vessels is described. Finally, two instruments, a new type of pulse height analyzer and an automatic scanner and sample changer are described. (For preceding period see HW-43441.)

22. 5493 HW-47012

The Thermal Test Reactor was put in operation and the initial calibrations are described. Measurements of bucklings of enriched uranium–graphite lattices are described. Results are also given for light water–enriched uranium lattices. A high temperature exponential experiment is described. Calculated curves for the bucklings of some special graphite lattices are given. A method for com-
Lattices 23-28

109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

It is concluded that for natural U-graphite lattices with $K_e = 1.04$, if the Xe effect is $\sim 3\%$, then $K_{c\mu}$ must be greater than 1.03 and it would be better to know $K_e$ for the design of such a reactor. However, for enriched U reactors or D$_2$O lattices where $K_e = 1.1$, and the Xe effect is $\sim 3\%$, it would be better to know $B_{\mu}$ for the design of the reactor. (T.R.H.)

27.

7843 NAA-SR-259
North American Aviation, Inc., Downey, Calif.

Exponential Experiment. For measurements of the thermal neutrons, bucklings of depleted U-D$_2$O lattices has been completed with the investigation of three additional lattices. Three lattices of slightly enriched U (0.9% $^{235U}$) in D$_2$O have also been studied. Besides the buckling measurements, intra-cell neutron density distributions and effective neutron temperature variations have also been measured. A nomograph for making resolving-time corrections has simplified the reduction of raw counting data to saturated full activities. Data obtained from foils exposed to a neutron current were found to exhibit a linear correlation between the magnitude of the current and the ratio of activities counted on the two sides of the foil. A measurement of the thermal neutron diffusion length in D$_2$O can be used to determine the macroscopic absorption cross section of the water, and hence the light-water Impurity. A discussion of the accuracy which can be obtained with this method of impurity determination is given. Water Boiling Neutron Source. Use of the water boiling neutron source for the exponential experiments and for miscellaneous irradiations has continued to increase. Preparations have been made for doing accurate danger coefficient measurements with the water boiler. These include the installation of an accurate indicator of control rod position and a thermocouple to indicate reactor temperature. Standard samples of B have been prepared. General Reactor Program. An investigation has been made into the possibility of securing intrinsic safety for heterogeneous power reactors. A possible method is to include in the reactor a relatively small homogeneous region containing a moderator, such as graphite, impregnated with $^{235U}$ and a substance whose thermal neutron cross section increases with increasing neutron temperature. Er is a suitable material for this purpose. Formulas and curves relating to the heat production by capture $\gamma$ radiation in slabs have been worked out as an aid to reactor design work. (auth)
An Al-U, heterogeneous, water-moderated critical assembly having a metal-to-water ratio (MWR) of 1.39 was constructed to measure criticality and reflector savings and the age of the system. The age is inferred from the nonleakage probability. The multiplication curve is somewhat irregular and tends to change its direction of curvature as it approaches criticality. The age calculated from the measured buckling is in fair agreement with previous experiments. (F.S.)

The material buckling of various hexagonal lattices of natural uranium rods (diameter 2.00, 2.53, and 3.05 cm) in heavy water has been determined in an exponential assembly. The diameter of the tank was only 1.0 metre. The detectors used were long BF3 proportional counters, which give a high counting rate and by a special arrangement eliminate higher radial harmonics. For comparison between different lattices the measurements yield the material buckling to an accuracy better than ±0.05 × 10⁻⁴ cm⁻². The absolute value is less accurate, owing to the uncertainty of the extrapolation radius (±0.2 cm). The multiplicative factor is calculated using published values of the disadvantage factor, and the constant and the resonance integral are evaluated. The possibility of correlating intensity and diffusion length is pointed out. (auth)

A series of exponential experiments was carried out on lattices of cylindrical uranium metal rods in a moderator of diphenyl. Three different concentrations of U⁹² in uranium were available for study, 0.50, 0.72, and 0.91 at. %. The fuel rods were arranged in square lattice arrays at moderator to fuel volume ratios 1.5, 2, 3, and 4. A cadmium clad 2 1/2 ft diameter aluminum tank was used to hold the lattices. External heaters maintained the contents at 185°F, just above the melting point of diphenyl. The lattices were driven by neutrons from the thermal column of a water boiler reactor. Foil activation techniques were employed to make the buckling and intracell flux distribution measurements. Measurements in pure and borated diphenyl were made in order to obtain the thermal neutron diffusion length and transport mean free path in diphenyl. Buckling and intracell measurements were made for some lattices in the borated diphenyl in order to determine the neutron age in these lattices. (auth)
Lattices 34-38

The critical parameters of aqueous UO2F2 solutions enriched to 90% U235 in various arrays of interacting slab geometries have been determined for a single concentration. Critical solution heights for various configurations of three nominally 3-in. thick Al slab vessels and one 6-in. Al slab vessel were determined. (T.R.H.)
Lattices 39-44

slab seed with U metal reflector were made with a UO₂ reflector. Differential control rod worth and differential moderator worth curves were obtained and were compared with similar curves obtained from the metal reflected slab. From these curves, the excess reactivity of a slab was determined from a measurement of the slab critical height. Measurements of flux shapes in three directions were made in order to determine critical bucklings in the vertical and one horizontal direction and to investigate water channel flux peaking. (auth)

39.
16717 WAPD-BT-8(8p.84-9)
CRITICALITY CALCULATIONS FOR THE TRX. R. B.
Horst. p.84-9 [of] BETTIS TECHNICAL REVIEW.
REACTOR PHYSICS AND MATHEMATICS. 6p.
The high U²³⁵ content of slightly enriched, water-
moderated reactors makes the theoretical prediction of
criticality difficult. Seven slightly enriched clean cri-
tical assemblies were constructed and their nuclear char-
acteristics were measured and compared with those
calculated for a theoretical, high U²³⁵ content core.
While the design of a power reactor would introduce
more complex problems, the agreement between the re-
activity characteristics of the model and the seven criti-
cal assemblies obtained in these experiments is good and
the model described herein should be useful in establish­
ing reactivity characteristics of future cores of this type.
(auth)

40.
6813 BNL-3145(Suppl.)
[Brookhaven National Lab., Upton, N. Y.]
CRITICAL ASSEMBLIES OF LIGHT WATER MODERATED,
SLIGHTLY ENRICHED URANIUM ROD LATTICES AT
BROOKHAVEN. Supplement to Hazards Summary Report,

41.
15076 A/CONF.15/P/1841
Brookhaven National Lab., Upton, N. Y.; Westinghouse
Electric Corp. Bettis Atomic Power Div.,
Pittsburgh; and Combustion Engineering Corp.,
New York.
PHYSICS OF SLIGHTLY ENRICHED, NORMAL WATER
LATTICES (THEORY AND EXPERIMENT). H. Kouts,
R. Sher, J. R. Brown, D. Klein, S. Stein, R. L. Hellena,
and H. Arnold. 70p. $0.50(OTS).
Prepared for the Second U. N. International Confer­
Measurements made with water moderated lattices of
slightly enriched uranium rods since the last Geneva
conference are presented and discussed. Additional
measurements on assemblies having uranium oxide as
the fuel are also presented. Improvements in technique
are described and analyzed. The evolution of theoretical
methods of interpreting and predicting the neutron be-
behavior of reactor cores of this kind is recounted. The
best present ways of analyzing the neutron economy and
criticality are described and discussed, and compari-
sions with the available data are given. The methods
range from multigroup treatments based on homogeniza-
tion to a few group treatments which may be carried out
by desk calculator methods. The derivation of the few
group parameters from multigroup results is described.
The theoretical treatments are supplemented by com­
parisons with the older four-factor methods. Finally,
some measurements obtained with plutonium enriched
rods are described, and tentative results are given.
(auth)

42.
15019 A/CONF.15/P/594
Atoms International Div., North American Aviation,
Inc., Canoga Park, Calif.
EXPERIMENTAL EXPERIMENTS ON GRAPHITE
LATTICES WHICH CONTAIN MULTI-ROD FUEL EL-
ELEMENTS. R. A. Laubenstein. 16p. $0.50(OTS).
Prepared for the Second U. N. International Confer­
A series of exponential experiments were carried out
on graphite-moderated lattices containing multi-rod
fuel clusters. In order to obtain information for design
calculations on sodium-graphite power reactors,
measurements of the lattice buckling and intracell flux
distribution were included. Experimental measure­
ments of the critical mass with and without sodium in
the core of the Sodium Reactor Experiment were in­
tended as a check on the application of exponential ex­
periment results to a critical assembly. (M.H.R.)

43.
1643 LA-749
Los Alamos Scientific Lab., N. Mex.
POLYTHENE-25 CRITICAL ASSEMBLY AND NEUTRON
DISTRIBUTION STUDIES. Hugh C. Priest and G. A.
Contract No.7405-eng-36, $9.30(ph OTS); $3.60(mf OTS).
Delayed critical behavior was investigated for an array
of 7/3 in. polythene cubes and U²³⁵ cubes of average com­
position UF₄₄C₁₄, in an 8 in. thick natural U tamper. The
pseudosphere of critical size (2130 cm²) contained 12.0 kg
of 94.5% U. Rossi time-scale measurements gave a =
-0.56 x 10⁹ at delayed critical. Activity distributions for
S, Au, U²³⁵, and U²³⁸ samples were obtained as a function
of radius of the assembly. Appreciable local variations in
neutron spectrum resulted from the inhomogeneity of the
active array. (auth)

44.
16730 AEC-tv-3362
EXPONENTIAL PILE MEASUREMENTS ON R3a FUEL
ELEMENTS. Rolf Persson. Translated by W. K.
Ergea (Oak Ridge National Lab.) from report AEF-65,
Clusters of seven uranium rods were investigated in
an exponential assembly. The material buckling was
measured by an extrapolation procedure and the thermal diffusion length was established from the neutron intensity. (auth)

45. 15025 A/CONF.15/P/600
Argonne National Lab., Lemont, Ill.

PROPERTIES OF EXPONENTIAL AND CRITICAL SYSTEMS OF THIORIA-URANIA AND HEAVY WATER, AND THEIR APPLICATION TO REACTOR DESIGN. W. C. Redman and J. A. Thie. 18p. $0.50(OTS).


Physics statics information was obtained through a series of heavy water exponential and critical experiments employing quarter-inch diameter fuel rods with thorium-U atom ratios of 25 and 50. Bucklings were determined for exponential triangular lattices of 1 1/2 to 1 1/2 in. pitch. Void and temperature coefficient of buckling also were measured. Systems of concentrical zones having different bucklings were investigated.

Critical experiments with a 1/8 in. triangular lattice spacing and height to diameter ratios from 0.8 to 2.8 gave buckling and reflector savings information.

Migration area was determined both by differential water worth and by variation of enrichment. Reactivity worth of voids and control rods was investigated. Thermal utilization and reactivity escape were also measured. Physics considerations in boiling reactor design are assisted by these statics experiments and also the observed operating characteristics of BORAX-IV. In seeking compatibility among reactivity, void coefficient, conversion ratio, safety and stability, yet heeding boundary conditions imposed by mechanical, hydraulic and heat transfer considerations, judicious compromises are necessary. Illustrative of these design principles, a small 40 MW ThO-U-D2O prototype boiling reactor having 400 kw/liter of coolant is presented. Although quite small, its attractiveness is further enhanced when extrapolated in size and power to those of practical interest. (auth)

1959

46. 6589 A/CONF.15/P/675


Measurements of the buckling of natural uranium oxide clusters in heavy water are presented. The clusters contain 6 and 7 oxide rods canned in aluminum. (A.C.)

Lattices 45-49

47. 6590 A/CONF.15/P/1192


The material buckling of natural uranium-beryllium oxide lattices was measured. The square cell or element had a pitch of 16 cm and uranium rods of 2.60, 2.92, 3.56, and 4.40 cm. A critical experiment was conducted with hollow uranium rods enriched to 1.35%. (W.D.M.)

48. 7623 NAA-SR-1026


Studies of 13 exponential experiments with graphite moderated lattices containing multirod fuel clusters are presented. The bucklings and detailed intracell flux distributions were measured for each lattice. Average flux values for each material of the unit cell are given. The theoretical analysis yields a value of the effective resonance integral and of the resonance neutron inverse diffusion length in the moderator, which can be used in 2-group sodium graphite reactor calculations. There is evidence that neutron spectral hardening corrections are important, but a crude treatment of this effect did not improve the fit to the experimental measurements. The calculations are presented in detail, and various lattice parameters are tabulated. (auth)

49. 22211 NAA-SR-Memo-3980


CRITICAL MASS OF AN OMR CORE USING PLATE TYPE FUEL ELEMENTS. H. C. Field. June 8, 1959. 4p. $1.80(pb), $1.80(mf) OTS.

The clean critical mass of an OMR core was determined experimentally using low enrichment U plate-type fuel elements in a six-inch lattice spacing and Santowax R at 340°F as the moderator. Criticality was attained with 39 fuel elements (2015 kg) present in the core, three shim rods withdrawn, and the fourth shim rod about 1/4 withdrawn. (W.D.M.)
Lattices 50-55

50.

2857 CF-58-8-3
Oak Ridge National Lab., Tenn.
CRITICAL EXPERIMENTS WITH 2.09% U\textsuperscript{235}-ENRICHED URANIUM METAL PLATES IN WATER. J. K. Fox, J. T. Mihalecko, and L. W. Gilley. Aug. 3, 1958. 9p. Contract \{W-7405-eng-26\}. $1.80 (ph OTS); $1.80 (mf OTS).

Experiments were performed with 2.09\% U\textsuperscript{235}-enriched uranium metal plates in a light-water-moderated and -reflected assembly. Each plate was 30 in. long, 3/4 in. wide, and 1/4 in. thick and contained 7.09 kg of uranium. In the first assemblies the plates were arranged in rows with edges adjacent, and the spacing between rows was varied from 1/4 to 1 1/2 in. The optimum spacing was about 1/4 in. With the spacing between rows maintained at 1/4 in., the spacing between the edges of the plates was varied from 0 to 1/2 in. The optimum spacing between edges was 1/8 in. The minimum critical mass in these assemblies was 6.74 kg of U\textsuperscript{235}. (auth)

51.

2859 CF-58-8-40
Oak Ridge National Lab., Tenn.

Critical experiments were performed with ORR and BSR fuel elements to determine safe arrays in which the elements could be handled and stored. The data indicate that the optimum spacing for criticality of 168-g ORR elements in water is 0.2 in. between locating bosses and that an infinite array of vertically placed elements one element high would probably be subcritical with a 1 1/16-in. spacing between locating bosses. For uniform arrays of adjacent elements in water, variation in the fuel loading per element between 140 and 200 g made very little difference in the critical mass. When 132 elements with an average loading of 160 g per element were closely packed in water in an 11 by 12 element array in which the rows were separated with 20-mil-thick cadmium sheets, no appreciable source neutron multiplication was observed; nor was there any appreciable multiplication when both the cadmium sheets and the water moderator were removed and the array was surrounded with 12-in.-thick paraffin reflector. A two-row slab-shaped array with 24 200-g center elements and 14 168-g elements on each end, all spaced 0.2 in. between locating bosses, was subcritical, and it appears that two infinitely long rows of 168-g elements would be subcritical. (auth)

52.

2859 CF-58-8-28
Oak Ridge National Lab., Tenn.

Experiments were performed with 2.09\% U\textsuperscript{235}-enriched uranium metal plates in a light-water-moderated and -reflected assembly. Each plate was 30 in. long, 3/4 in. wide, and 1/4 in. thick and contained 7.09 kg of uranium. In the first assemblies the plates were arranged in rows with edges adjacent, and the spacing between rows was varied from 1/4 to 1 1/2 in. The optimum spacing was about 1/4 in. With the spacing between rows maintained at 1/4 in., the spacing between the edges of the plates was varied from 0 to 1/2 in. The optimum spacing between edges was 1/8 in. The minimum critical mass in these assemblies was 6.74 kg of U\textsuperscript{235}. (auth)
tion for the thermal neutron flux in a nonabsorbing heavy gas with a temperature discontinuity, and an alternative method of deriving the group constants for Selengut's two-thermal-group approximations is given. Calcula-
tions are shown for thermal flux in a cell with tempera-
ture discontinuities by a method employing the formal-
ism of few-group theory but retaining the qualitative features of neutron distribution. Results of buckling measurements for fuel elements in random vs. uniform array are summarized. Measurements of the number of fuel rods required for criticality for several lattices with 1.6% enriched U and light water moderator and reflector were completed using the "Approach-to-
Criticality" method. In connection with processing of fuels enriched to 5%, $K_\infty$ values were measured in PCTR for UO$_2$ hydrogen-moderated systems. A re-
evaluation of $\eta$ which results in more realistic critical parameters for natural U-light water lattices is re-
ported. A summary is given of the empirical method used to estimate critical masses of oralloy vs. core density when surrounded by a full density natural U reflector. The techniques for measurement of $K_\infty$ in PCTR are extended to the case where there is a tem-
perature change in the fuel. Measurement of the metal temperature coefficient of $K_\infty$ for 19-rod clusters of UO$_2$ in the PCTR is described. Expressions are given for the harmonic and end corrections for a three-region exponential pile. Measurement of material bucklings of seven-rod clusters of 0.925-in.-diam. natural U fuel ele-
ments in a graphite moderator was completed. Fission cross sections for Am$^{244}$ and Np$^{237}$ at 0.1 to 5 ev were measured using the crystal spectrometer. The weights of the U$^{235}$ fission foils used at Hanford and Harwell for absolute fission cross section measurements are com-
pared. Results of measurements of the age to in reso-
nance of monoenergetic neutrons in kerosene and water are given. (For preceding period see HW-58789.)

Lattices 56-59

enriched U rod-water lattices are given. The develop-
ment of a semi-empirical formulation for simplifying calculations of the fast effect for U metal and UO$_2$ fuel elements in light water is presented. Fuel cloutment en-
richment, rod size, and lattice spacing are the vari-
bles considered. (For preceding period see HW-56919.)

57.

1969

NP-7762

iowa State Coll., Ames.

OPERATING CHARACTERISTICS OF A URANIUM

GRAPHITE SUBCRITICAL ASSEMBLY WITH COOLANT


88p.

Experimental and theoretical investigations with ura-
nium-graphite subcritical assemblies were carried out
to determine the various nuclear constants of a lattice.
The lattice constants were the material buckling, multi-
plication constant, lattice diffusion length, and thermal
utilization of the unit cell including the process tube
assembly. In this investigation six lattice configurations
were considered. (W.D.M.)

58.

8237

HW-58879

General Electric Co. Hanford Atomic Products

Operation, Richland, Wash.

CORRELATION OF EXPONENTIAL AND PCTR MEAS-

UREMENTS ON CLUSTER FUEL ELEMENTS WITH


$4.80(ph).$2.70(mf) OTS.

Recent improvements in methods of lattice parameter
calculation are reported. By refinement of conventional
calculation methods, successful applications to cases of
cluster fuel elements were found. The model used is
described, and the resonance escape probability integ-
ral, fast effect, fission neutron release to neutron fuel
capture ratio, and thermal utilization are calculated.
In addition, the diffusion length and Fermi age were calcu-
lated as well as neutron streaming and density correc-
tions. The experimental work is described, and a
comparison of theoretical and experimental results is
made. (J.R.D.)

59.

22209

HW-61547

General Electric Co. Hanford Atomic Products Opera-
tion, Richland, Wash.

CRITICALITY OF FUELS OF LOW ENRICHMENT IN


$3.30(ph).$2.40(mf) OTS.

Curves are presented of critical masses as a function
of cylinder diameter and fuel rod size for 1.6, 3.0, and
5.0% U$^{235}$ enriched metal-water systems. (W.D.M.)
LATTICES 60-66

60. BNL-486
Brookhaven National Lab., Upton, N. Y.
EXPERIMENTAL STUDIES OF SLIGHTLY ENRICHED URANIUM, WATER MODERATED LATTICES. PART I. 0.600-IN.-DIAMETER RODS, Herbert Kniss and Rudolph Sher. Sept. 1957. 41 p. $1.25(OTS).

Studies on 15 uranium-water lattices of 0.600-in.-diameter uranium rods are presented. Three fuel enrichments were used; these had nominal U\(^{235}\) contents of 1.0%, 1.15%, and 1.3%. Actual enrichments and densities for the fuel is given. Five ratios of water-to-uranium volumes were studied with each fuel enrichment. The pertinent geometrical data on these lattices are given. (W.L.H.)

61. 7104 A/CONF.15/P/245

Lattices of natural uranium and heavy water were investigated for two types of fuel elements—the rod cluster and tube cluster types. The geometric dimensions of uranium are determined by thermal considerations assuming a surface heat flux of 100 \(\text{w/cm}^2\) and fixing the number of subelements per cluster. The specific power \(w/g\) of fuel is a parameter varied within reasonable limits. New methods for the calculation of the lattice parameters \(c\), \(p\), and \(f\) were employed, including recently published American, Canadian, and French methods. Some further refinements for the resonance capture of interior surfaces of the composite elements were developed which are applicable to systems with a coolant material different from the moderator. Finally, the material buckling is obtained for the cold-clean as well as the hot-poisoned case. The methods of calculation are then applied for other subdivided elements very similar in geometric shape for which the material buckling is known from exponential or critical experiments. (auth)

62. 18252 WAPD-TM-130

A series of critical experiments was performed on slab geometry critical assemblies to study the basic behavior of seed-blanket type cores. These experiments included water-worth measurements, inference of excess and shutdown reactivities, temperature coeffi-

63. 16919 IRW-57553

The buckling was measured for three different fuel element types in random arrays. The average value of the buckling was less than the buckling of uniform arrays in all three cases, but in view of the 95% confidence limit, it is not considered advisable to increase the safe critical limits of random arrays above those of uniform arrays. (D.E.B.)

64. 17365

The design of efficient systems for processing spent power reactor fuel elements requires information on the critical masses involved. To obtain some of the needed information, approach-to-critical experiments were carried out with light-water-moderated and reflected hexagonal-lattice assemblies using enriched solid U fuel elements. Experiments were made for fuel elements of 0.925, 0.600, and 0.300-in. O.D. with the \(\text{H}_2\text{O}/\text{U}\) ratio for lattice spacings to span maximum buckling and minimum critical mass. (W.D.M.)

65. 17049 BNL-536

The study of effective resonance integrals in a heterogeneous system was continued. Measurement of the buckling and reflector savings of the 0.250-in.-diam. rod, water-moderated lattices was completed for both 1.027 and 1.5% enriched U metal.

66. 19651 CF-59-7-87
Oak Ridge National Lab.
MULTIPLICATION MEASUREMENTS WITH HIGHLY ENRICHED URANIUM METAL SLABS. J. T. Mihalesco

A series of neutron multiplication measurements with arrays of 8 by 8 by 11 in. slabs of 93.4% U\textsuperscript{235}-enriched uranium metal was made to provide data from which safety criteria for the storage of these fissile units can be established. Each slab contained 22.9 kg of U\textsuperscript{235}. A maximum of 125 units was assembled. The arrays studied were cubic lattices of the units and were usually parallelepipedal in shape. Arrays were both unmoderated and Plexiglas-moderated and were surrounded in most cases by a 1-in.-thick Plexiglas reflector. The lattice densities (ratio of fissile unit volume to lattice cell volume) were between 0.023 and 0.06. Unmoderated lattices with a density of 0.06 would require 145 ± 5 units for criticality, while those with a density of 0.023 would require 350 ± 30 units. In lattices in which the fissile units are separated by 1 in. of Plexiglas, approximately 27 units would be required for a critical array with a lattice density of 0.06 and about 75 units for a density of 0.023. Distributing Foamiglas (containing 2% boron) throughout a moderated array increased the critical number of fissile units by a factor of 5, while Styrofoam had a small effect. (auth)

67.

10629  CRRP-648

Atomic Energy of Canada Ltd. Chalk River Project, Chalk River, Ont.


The natural uranium metal fuel was in the form of cylindrical slugs 1.31 ± 0.01 cm diam, by 16 cm long fitted loosely into aluminum tubes 210 cm long. The clusters were suspended by a gimbal arrangement on beams which were placed across the top of the pile. The buckling of the lattice under investigation was obtained from measurements of the macroscopic flux distribution through the reactor. The thermal flux distribution through the central cell was measured to obtain the thermal utilization and thermal diffusion area. The slowing down area for the lattice was calculated. In addition to the fine structure measurements made with manganese wires, the distribution of neutron capture by U\textsuperscript{238} through the central cluster was observed. (W.D.M.)

68.

7071  A/CONF.15/P/151


Methods have been developed for predicting the buckling of cold clean lattices of uranium metal or uranium oxide (U\textsubscript{3}O\textsubscript{8}) rod clusters in heavy water. The methods involve refinements and modifications of the standard two-group recipe and have been correlated with exponential and critical experiments. In principle, the lattice parameters $\kappa$, $p$, $f$, $L^2$, and $\Psi$ are calculated, and $\Psi$ is determined from measurements of the buckling. (auth)

69.

6587  A/CONF.15/P/160


A description is given of some unique features of the experimental facility. The theory of mixed lattices is discussed. Measurements are reported on the buckling for lattices consisting of clusters of uranium metal and oxide rods in D\textsubscript{2}O at room temperature, the temperature coefficient between 10 and 80°C for D\textsubscript{2}O and for different lattices, and control rod effects. (W.D.M.)

70.

9432  BNL-483(p.61-72)

Argonne National Lab., Lemont, Ill.

AGE AND RESONANCE ESCAPE PROBABILITY FOR THE THUD CRITICAL. W. C. Hedman. p.61-72 [of]

THORIUM—U\textsuperscript{233} SYMPOSIUM, SPONSORED BY THE UNITED STATES ATOMIC ENERGY COMMISSION AT BROOKHAVEN NATIONAL LABORATORY, JANUARY 9-10, 1958. 12p.

The term THUD designates a system of Th\textsubscript{2}O—U\textsubscript{3}O\textsubscript{8} fuel elements and heavy water currently under investigation in both exponential and critical experiments. The specific object of the THUD program at its outset was to develop the information required for the design of a heavy water—oxide fuel loading for the EBWR. The geometrical buckling for a 1/4-inch triangular lattice of 25:1 Th to U\textsuperscript{233} fuel rods was determined from a sequence of seven clean criticals. Two independent methods have been employed to establish experimentally the neutron age for the THUD system. Values of the resonance escape probability for the 1/4 inch triangular lattice have been obtained by fuel and analysis of the fuel substitution experiments. Several experimental determinations have been made of the effective resonance integral for Th and Th\textsubscript{2}O. (W.L.H.)

71.

1802  AERE-T/M-166


The discrepancy between theoretical and experimental values of the reactivity of enriched uranium—heavy water lattices was investigated. A description of the
experimental apparatus, known as DIMPLE, is given, along with a discussion of the procedure. It was concluded that all of the discrepancy could not be accounted for; however, some reasons are given, such as incomplete allowance for the inhomogeneity of the core resulting in an underestimate of the leakage. (J.R.D.)
77. Distributions were made of the control system for the LMFR. It is described which will maintain continuous control of the fuel concentration. In a test of the "torch" process for removing Pa and U from ThF₄ blankets by the fluorination of contaminants in a H₂-F₂ flame, 77% of the original feed lost 72% of its original activity. A method using fused salts was less successful. It was found that 92 to 95% of mixed fission product activity in molten NaNO₃ scavenged out by a suspension of metallic oxides such as Mn and Fe. Isothermal studies of Th₂Bi₂ and a liquid Pb-Bi eutectic showed that after 2000 hr, the compound particles grow larger in Bi, but thermal cycling tests with the same dispersions indicated that the particles grow larger in Pb-Bi. Static corrosion tests of B and W alloy 701 and chromized Croloy 2½ showed no corrosion in Bi or U-Bi. Wetting experiments revealed that pure Fe can be wetted by 1600°F under high vacuum and immersing in Bi at 725°C for 5 hr. Additional data are presented on the erosion of graphite by Th-Bi, the solubility of Fe and Zr in Bi, the adsorption of Zr on Fe from liquid Bi, the radiation damage of SiO₂, and the thermal cycling of U. The transmission of neutrons through an air slot was investigated as a function of slot dimensions. Parameters of several neutron resonance levels in U²³⁸ were obtained. (For preceding period see BNL-285.) (C.J.G.)

78.

19689 HW-65328


A series of investigations involving critical approach and exponential measurements made with 3.06% enriched U is reported. Data from these experiments were used to calculate critical parameters and safe values which are shown graphically. The data pertain to U rods of diameters which in one case result in the minimum critical mass, and in the second case the maximum buckling. An illustrative problem in nuclear safety is included in which mass and volume are discussed for a hypothetical dissolve used to process U fuel elements of 3.1% enrichment. (J.R.D.)

80.

22619 WCAP-1413


Critical experiments at high water-to-metal ratios using 2.1% enriched UO₂ stainless steel clad fuel rods and the associated core components from the Yankee critical experiment were performed to obtain critical mass data in the range of moderating ratios close to the optimum. Experimental work for the loose-lattice experiments was completed. Calculations of the critical mass at moderating ratios close to the optimum were made. (See also WCAP-1414.) (C.J.G.)
Lattices 81-87

81.

13048
May
PHYSICS UNIT - APPLIED

13048


The spectra of Tl and Ra were determined. The results of neutron streaming calculations for Hanford graphite are reported. Buckling measurements on a 6 in. small slug lattice were completed. Critical lattice experiments and neutron distribution studies are also reported. (D.E.B.)

82.

2878

NAA-SR-Memo-3980


CRITICAL MASS OF AN OMN CORE USING PLATE TYPE FUEL ELEMENTS. H. C. Field. June 8, 1959. 4p. OTS.

The clean critical mass of a core using low enrichment U plate-type fuel elements in a six-inch lattice spacing, and using Santowax R at 340°F as the moderator was experimentally determined. Criticality was attained with thirty-nine fuel elements, containing 3915 kg of U, present in the core, three shim rods withdrawn, and the fourth shim rod about 1/4 withdrawn. Interpolated clean critical mass was determined to be 3910 kg of U. (W.L.H.)

83.

9183


The effects of H2O contamination on lattices of natural uranium metal in D2O were measured. The buckling changes associated with H2O contamination were determined for two lattices with moderator-to-fuel volume ratios of 12.3 and 14.6 over a range of H2O concentrations from 0.2 to 8.2 mole %. The agreement between calculated and experimental changes in buckling for these lattices was within ± 25 x 10^-4 cm^-2. Similar measurements on seven other lattices with moderator-to-fuel ratios in the range from 31 to 212 were made for a change in the H2O concentration from 0.18 to 3.92 mole %. For these measurements the experimental change in buckling was about 15% greater than the calculated change. (auth)

84.

20783

RFP-182

Dow Chemical Co., Rocky Flats Plant, Denver.


1)-1106. OTS

Neutron multiplication measurements were made on 6.5-in. diam cylindrical assemblies of enriched U discs immersed in aqueous solutions of enriched UO2(NO3)2. Diffusion calculations were made on homogeneous mixtures of the enriched U with varying H:U atomic ratios. (auth)

85.

18134

LA-2026

Los Alamos Scientific Lab., N. Mex.


A solid cube of oralloy becomes critical at 24 kg when immersed in an infinite water reflector. Various critical lattices were obtained by dividing this solid shape into small units and uniformly dispersing them at various mean densities. For a given size of oralloy unit, there is a mean density at which the critical mass is a minimum. The H to U atomic ratio of the cores with minimum critical masses was determined. Measurements with uniformly dispersed oralloy do not indicate a critical mass below the minimum observed with a uniform lattice. Multiplication measurements with Au, Ag, and Cd rods inserted in the oralloy matrix yielded the effective cross section ratios: \( \sigma_v(Au)/\sigma_v(Ag) = 0.86 \), and \( \sigma_v(Cd)/\sigma_v(Au) = 1.58 \). These values are independent of position and lattice spacing for ranges examined. (auth)

86.

16424

International Atomic Energy Agency, Vienna.


Reactor physicists from Chalk River, Saclay, Kjeller, Harwell, Savannah River, A.B. Atomenergi, and Argonne attended the panel meeting on the Physics of Heavy Water Lattices. The research programs of the various laboratories were discussed and the existing data were compared and evaluated. A series of 14 technical papers is presented in the appendix; they cover practical aspects of heavy water reactor lattices such as operating experience with the NRU, definitions of lattice parameters, lattice buckling, resonance escape probability, and reactor spectra. (D.L.C.)

87.

2877

NAA-SR-Memo-3892


CALCULATIONS ON THE TRX CRITICAL ASSEMBLY. L. Maki and W. Allen. [1959?]. 5p. OTS.

The following items were calculated for the different lattice spacing and were compared with the experimental results: resonance escape probability of U, relative thermal flux in fuel and moderator, and thermal utilization and effective multiplication. (W.L.H.)
SHAPE PERTURBATIONS IN CRITICAL EXPERIMENTS.

W. A. Reardon and R. C. Lloyd. p.70-6 of NUCLEAR PHYSICS RESEARCH QUARTERLY REPORT (FOR) OCTOBER, NOVEMBER, DECEMBER 1959.

When making heterogeneous critical mass measurements, several perturbations of unknown magnitude are usually present, such as the effect of an irregular outer boundary. Critical approach type measurements were made with circular, elliptic, and rectangular cylinders to try to evaluate some of the effects. The uranium rods used were 23.5 inches long, 0.175 inches in diameter, and 3.0573%, U²³⁵. These were encased in 0.025-inch wall Lucite tubes and were arranged in a 0.5-inch triangular lattice; the resulting H₂O/U (total) volume ratio was 8.0. The assemblies were both fully reflected and moderated with H₂O. (auth)

4726 RFP-169
Dow Chemical Co. Rocky Flats Plant, Denver.


Previously published data were used for an empirical study of critical arrays of slab and cylindrically shaped vessels interacting through water. These vessels contained an aqueous solution of UO₂F₂ and the U²³³ enrichment was greater than 90%. Approximate isolation thicknesses of water for each of the above two cases are given. (auth)

90.
9910 RFP-174
Dow Chemical Co. Rocky Flats Plant, Denver.


Neutron multiplication measurements were performed on two identical finite Pu-metal slab assemblies separated and reflected by Plexiglas. (auth)

91.
9837 NAA-SR-Memo-3872

FLUX DISTRIBUTION MEASUREMENTS IN NAA OMR CORES. V. A. Swanson. May 14, 1959. 14p. OTS.

The neutron flux distributions in a flexible mockup of an OMR core portion were measured. Two lattice configurations were studied. The diffusion length of thermal neutrons in the moderator and the critical size of the lattices were also measured. Results, together with some comparisons with theoretical calculations, are tabulated. (J.R.D.)

94.

Studies were made to determine the enrichment required to provide excess reactivity of 4.5% above hot and poisoned for various numbers of clusters of UC fuel in the SRE. The fuel element is in the form of seven-rod clusters 6-ft. long. A plot is given of effective multiplication constants K_eff vs. enrichment for cores consisting of 31, 37, and 43 clusters. The effects of additional clusters on reactivity with constant enrichment are shown. Cell thermal flux and core axial thermal flux plots are given for 31 clusters with 5.0 at.% enriched fuel and are also plotted for 43 clusters with 4.0 at.% enriched fuel. (W.L.H.)

83
Lattices 95-100

95.


A summary of the work accomplished during the third contract quarter (January 1 to March 31, 1961) on the Spectral Shift Control Reactor (SSCR) Basic Physics Program is presented. The major objective of the program is to determine basic physics parameters of tight-packed lattices of slightly enriched fuel in moderators consisting of D_2O-H_2O mixtures. The concentration of the D_2O-H_2O mixtures are varied so as to apply to the spectral shift concept. The required license amendment was issued, and the first critical experiment containing heavy water in the moderator (76.7 mole % D_2O) was performed with 4%-enriched UO_2 fuel rods. Reported are measurements of the critical mass, critical buckling, thermal disadvantage factor, cadmium ratio of U^{235} and U^{238}, epithermal neutron spectrum, and the effect of moderator channels. The operating license for the room temperature exponential experiments was issued, and preliminary buckling measurements with (U^{235}+Th)O_2 fuel in a moderator containing 80 mole % D_2O were made. The results of hot exponential experiments at 70 to 400°F with 4%-enriched UO_2 fuel in light water moderator are also reported. Neutron age measurements in a lattice of ThO_2 rods in light water and 90 mole % D_2O, parallel and perpendicular to the rod axis, were completed and a preliminary analysis of the results is presented. Theoretical studies continued. The BPG computer code, which will be used to analyze experiments with D_2O in the moderator, was refined by improving the methods of computing resonance absorption and the transport cross sections for fast neutrons. Alternate methods of computing Dancoff shielding in the lattice and resonance absorption in fertile material were considered, and BPG and PLMG calculations were compared to assess the importance of the choice of neutron spectrum in the reflector. Additional calculations in support of the experimental work also continued. (auth)

96.


97.


The values of k_{ex} and f were measured for a cluster fuel element in a 7-in. graphite lattice. The value of k_{ex} was negative for natural uranium fuel; so two methods of measurement were compared: (1) the normal method of poisoning the lattice with thin strips of copper with an extrapolation of the result back to a k_{ex} of one and (2) producing the condition of unit k_{ex} by adding to the cell U^{235} as thin strips of U^{238}-aluminum alloy. A measurement was made for a slightly enriched fuel in the same geometry, with k_{ex} positive, to provide a check on the other methods used. The measured values of k_{ex} agreed within the quoted error. The values of k_{ex} and f are, respectively, 0.944 and 0.907 as measured by adding U^{235} to the cell containing natural uranium fuel. (auth)

98.

24540 EXPONENTIAL EXPERIMENTS ON NATURAL URANIUM-IMPREGNATED GRAPHITE LATTICES. R. Bonalumi, C. Bruschi, G. B. Zorzoli (Laboratori CISE, Segrate, Italy). Energia nucleare (Milan), 8: 321-5 (May 1961). (In English)

Exponential experiments on natural uranium-impregnated graphite lattices are described. Buckling was measured on three different lattice pitches, corresponding to spacings of 10, 14.1, and 20 cm; the experimental results are discussed. (auth)

99.


The calculated bucklings are compared with experimental bucklings measured on uranium rods in water at enrichments of 1, 2, and 3 per cent U^{235}, for rod diameters up to three inches. Twenty two such measurements are included. Bucklings are calculated using the one-group criticality equation. Lattice parameters are calculated. The maximum material bucklings obtained are biased high (conservative from the standpoint of nuclear safety). The bias, however, is not unreasonably large: for 1.0-inch diameter rods at 1.0 per cent U^{235}, the bias is 130 ± 320 microbucks; for 0.5-inch diameter rods at 3.0 per cent U^{235}, the bias is 190 ± 330 microbucks; and for 0.5-inch diameter rods at 5.0 per cent U^{235}, the bias is 320 ± 420 microbucks. Maximum bucklings and minimum critical masses are presented for water-reflected lattices of 1.63, 2.0, 3.063, and 5.0 per cent U^{235} enriched uranium rods in light water. Extrapolation lengths are estimated from experimental measurements. (auth)

100.


A series of experiments were carried out during the past several years to study the basic reactor physics of several nuclear reactors and their constituents. Three moderators, heavy water, graphite, and diphenyl, were used, in turn, in these lattices. Three metal fuels (but not mixtures thereof) were used. The fuel enrichments (at %) were 0.4962 (depleted), 0.7205 (natural) and 0.9124 (enriched). The fuel elements made from 4-in. long slugs, were in the form of a cylinder which had a diameter of one inch. These elements were 5 ft long. These elements are summarized and simplified two group theoretical comparison of the results presented. It is demonstrated that this model is not accurate for the description of these lattices. (auth)


Material buckling and intracell flux distribution measurements are made for a series of diphenyl-moderaated, uranium-metal exponential assemblies. The lattices consist of square cells with a 1-in.-diameter fuel element located at the center and spaced to give moderator-to-fuel (M-F) ratios of approximately 1.5, 2, 3, 3.5, and 4. Fuel enrichments of 0.4962, 0.7205, and 0.9124 at % are used. The lattice is maintained at a temperature of 180°F. Measurements are conducted for natural fuel with a 3.15 M-F ratio, and the buckling is found to be \(-4.1 \pm 0.6 \, \text{m}^{-1}\). Intracell flux distributions are measured for eight lattices and compared with calculations using cross-sections averaged over a Wigner-Wilkins spectrum. Agreement is good for small spacings but grows worse as the cell size increases, because the cell structure departs from the homogenized cell used to calculate the neutron spectrum. Thermal utilizations are calculated from both the measured and calculated flux distributions and are found to disagree by less than 2%. (auth)


Critical Experiments. After reports of fuel-element and control-rod fabrication progress, the critical experiments performed are outlined. These experiments included measurements on a 3.7% core, criticality measurements on a three-region core, peripheral fuel rod worth for a three-region core, fuel-rod and foil scans in a three-region core, multiple foil measuring techniques, analysis of errors in \(\text{U}^{235}\) Cd ratio measurements, and comparison of 3.7 and 2.7% cores. Nuclear Analysis of Multi-region Reactor Cores. Work continued on calculations of lattice parameters, comparison of results with experiment, and evaluation of methods of analysis. (For preceding period see WCAP-1419.) (T.H.H.)


Results are reported for an extension of critical experiments and analyses which was made in the program for evaluation of water-moderated multiregion cores to study the discrepancy between calculated and experimental values of \(\rho^{98}\) (resonance capture/thermal capture ratio in \(\text{U}^{239}\)) and \(\delta^{98}\) (resonance fission/thermal fission ratio in \(\text{U}^{239}\)). (auth)


Progress in analytical studies and critical experiments for the multi-region reactor lattice studies of the fuel cycle development program is described. The experiments utilized stainless steel clad uranium dioxide fuel rods of three different enrichments and two moderating ratios. Performance of the scheduled experiments with the 4.5:1 W/U lattices was continued without interruption. The experimental results are reported under the headings of criticality measurements, flux distributions, and microscopic parameter measurements. Criticality measurements were made on various two- and three-region cores. Results for loading, critical water height, banked rod position, and peripheral fuel rod worth are tabulated. The cross section schematic diagrams of the cores utilized are shown. The cadmium ratios for gold and \(\text{U}^{235}\) in the moderators were determined. Scans are shown of fuel rods, gold foils, and \(\text{U}^{238}\) foils for water slots and for various slab materials inserted in the slot. The parameters of the lattices studied experimentally were calculated and the results were compared with those obtained experimentally. (M.C.G.)
Lattices 107-115


Light water critical experiments are performed in order to measure microscopic parameters and conversion ratios in single region and multi-region cores containing slightly enriched, stainless steel-clad, UO₂ fuel rods. The experimental results are analyzed with multi-group codes, which are supplemented for the resonance energy region by a Monte Carlo code. Experimental and analytical results are presented, along with a description of investigations of discrepancies between theory and experiment. (auth)


109. 6999 (HW-64866(p.65-87)) MATERIAL BUCKLINGS OF GRAPHITE URANIUM LATTICES. G. W. R. Endres and D. E. Wood (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Material buckling measurements were completed for a series of lattices with a tube-in-tube natural uranium fuel element for most lattice spacings with tube-and-rod and I and E fuel elements and for one lattice, using a second tube-and-rod fuel element. Bucklings were measured in exponential piles for both air and water coolant. Material buckling values and side-to-side extrapolation lengths for the tube-in-tube, tube-and-rod, and I and E fuel elements are shown. Dimensions, cladding, densities, volume ratios, and volumes per centimeter of length for the fuel elements are given. (W.D.M.)


A series of exponential measurements performed on uranium-heavy water lattices is summarized. Three enrichments of fuel materials were used: 0.49, 0.71 (natural), and 0.90 wt. % U³⁹. The uranium fuel slugs ranged from 0.75 to 2 in. in diameter and were encased in tubes of 0.40-in. aluminum. The lattice cell spacings varied from 3 to 12 in. The analysis of the data is divided into two parts. The first part deals directly with the neutron flux measurements and reports for each lattice, the material buckling (B²), a pair of disadvantage factors (F and F₀), and a flux ratio (R). The second part of the analysis is concerned with the long-range problem of improving reactor calculations. Attempts were made to correlate the measurements with themselves and with other nuclear measurements so as to obtain a consistent formalism for computing uranium-heavy water reactor parameters. (W.L.H.)


112. 6594 (ORNL-3016(p.72-3)) CRITICAL DIMENSIONS OF RANDOM ARRAYS OF ThO₂-UO₂ PELLETS. J. K. Fox and L. W. Gilley (Oak Ridge National Lab., Tenn.).

Several criticality experiments were performed with assemblies of unclad ThO₂-UO₂ pellets, randomly stacked in aluminum or steel cylinders. Water was added to the cylinders to serve as neutron moderator, and all assemblies were completely water-reflected. The results of these measurements and of preliminary experiments in which the pellets were contained in plastic bottles indicated that the critical dimensions of assemblies of such pellets are extremely sensitive to the degree of neutron moderation. (auth)


The loose lattice experimental work performed under the Multi-Region Lattice Program conducted at the Westinghouse Reactor Evaluation Center is described. The fuel used in these experiments was the 2.73% en-
riched Yankee CRX fuel, described on page 14 of
YAEc 84. The purpose of the experiments was to obtain
critical mass information in the range of moderating
ratios close to the optimum and to verify the physics
computational methods now used for fuel storage calcu-
lations. The experimental program consisted of critical
size, buckling, reflector savings, and microscopic param-
eter measurements. (auth)

116.
21786 HETEROGENEOUS URANIUM METAL-
GRAPHITE CRITICAL EXPERIMENTS. C. A. Guderjahn
Nuclear Soc., 4: No. 1, 104(June 1961).

117.
13459 (HW-67219) NUCLEAR PHYSICS RESEARCH
QUARTERLY REPORT, JULY, AUGUST, SEPTEMBER 1960.
(General Electric Co. Hanford Atomic Products Operations,
114p.

Results are reported of the neutron-induced fission in
low-energy resonances of Pu239 and of resonances in the
target nuclides Pu238, U236, U238, and Pa231. A three-axis
crystal spectrometer was constructed for the study of dif-
ferent inelastic scattering cross sections of neutrons in
the energy range from near thermal to a few tenths of an
ev. Results are reported of the scattering of 0.147-ev neu-
trons from water. Hanford's 2-Mev positive ion Van de
Graaff accelerator was equipped with motor-driven sup-
ports, which permit mechanical positioning of the acceler-
ator from a remote station in the control room. The elec-
trode system installed to determine the location of the ion
beam within the vacuum system is described. The system is
designed to furnish information necessary for the posi-
tioning of the accelerator and further alignment of the
beam. A study is reported of the position and energy de-
pendence of the neutron flux in a heterogeneous system.
The effect of a 1/4 absorbing cylinder on the thermal flux in
a surrounding nonabsorbing moderator is treated in an
improved approximation. The basic improvement embo-
dled in the solution is the use of a more accurate description
of the energy dependence of the blackness of the absorber.
A study was made on the interpretation of PCTR measure-
ments of Kp for the case in which the addition of poison perturbs the neutron flux distribution in any or all of the
various components of the test cell. Measurements were
also made of Kp for lattices in which only one fuel element
or one central cell is available. Lattice-cell flux traverses
from experiments were examined in order to develop a
method for calculating reaction rates in uranium-graphite
heterogeneous systems. Variations in the neutron energy
spectrum were examined to improve standard methods, es-
pecially for systems at elevated physical temperatures.

Lattices 116-120

of 14/8 in. A tube-in-tube natural-uranium fuel element
with water coolant was used in all three piles. Proposed
studies for the critical facility of the Plutonium Recycle
Program are reported. A series of exponential pile mea-
surements were made to determine the criticality of 2.8%
Enriched uranium dioxide rods in light-water lattices.
These measurements were made primarily to obtain data
for establishing nuclear safety criteria relative to propos-
als for processing power reactor fuels. (For preceding
period see HW-64865.) (W.L.H.)

118.
5572 WATER-MODERATED CORES WITH BORON
STEEL SEPTA AT ELEVATED TEMPERATURES. G. D.
Hickman, J. A. Biatline, and L. A. MacNaughton (Knolls
Atomic Power Lab., Schenectady, N. Y.). Nuclear Sci. and

A series of 15 experiments was carried out on a 8 x 30 x
32 in. core in the Pressurized Critical Assembly at KAPL.
In 12 of these experiments, 0.030-in. boron stainless steel
septa bisected the 8-in. dimension. These septa contained
vastar weight per cent B10. In the remaining three exper-
mements, there were no boron-stainless steel septa in the
core. The eigenvalues and neutron density distributions
were compared with values calculated from Deutsch cross
sections and "Thin Region Theory." The calculated eigen-
values were within 1% of the experimental values, with a
spread of approximately 1%. For all the cores, the calcu-
lated eigenvalues were lower than the experimental values.
Analyses of the neutron density distributions showed the
calculated results in fairly good agreement with the ex-
perimental results. In all cases, this agreement was as
good for the cores which contained the boron septa as for
the other which did not. It therefore appears that the boron
was well represented by "Thin Region Theory," and that
the main discrepancies between calculated and experimen-
tal values are due to the inadequacies of adapting the
Deutsch scheme to these cases. (auth)

119.
21785 NATURAL URANIUM-D2O BUCKLINGS OVER
AN EXTENDED RANGE OF PITCH AND FUEL ASSEMBLY
SIZE. T. J. Hurley, Jr., H. R. Fike, and G. F. O'Neill

120.
28336 (CRRP-894) SOME CLOSE-PACKED LATT-
ICES IN LIGHT WATER AND HEAVY WATER. PART I.
BUCKLING MEASUREMENTS. R. G. Jarvis, G. J. Phillips,
and W. H. Walker (Atomic Energy of Canada Ltd, Chalk
(AECL-1254)

Measurements of buckling were made in an exponential
system for a set of twelve lattices. The fuel was in rods
1.28 in. in diameter, in aluminum cans, and was arranged
in square lattices at spacings of 1.60, 2.11, and 3.20 in. At
each spacing the buckling was measured for natural and
depleted uranium contained 0.26% of depleted uranium in light water and in heavy water. The depleted uranium contained 0.26% of depleted uranium in light water and in heavy water. The

121.
7019 (ORNL-3016(p.76-8)) CRITICAL EXPERIMENTS WITH PRNC RESEARCH REACTOR FUEL ELEMENTS. E. B. Johnson and K. M. Henry, Jr. (Oak Ridge National Lab., Tenn.)

A series of critical experiments with the Puerto Rico Nuclear Center research reactor fuel elements was completed in the Pool Critical Assembly of the Bulk Shielding Facility. The experiments validated the computations on which the design fuel-plate loading was based, and they determined the amount of excess reactivity to be expected from three clean, cold loadings proposed by the reactor's designer as operational configurations. (auth)

122.

Criteria were established for the safe fabrication and storage of 9.87% U235 enriched UO2 fuel rods. Due to the limited amount of fuel to be handled in this project, safe criteria were based on fuel rods at optimum rod size and lattice spacing and water flooding. When the fuel is handled in pellet form, 530 g contained U235 is the maximum safe batch size independent of container size. The maximum safe volume, independent of mass, is 6.5 liters. When handled as 0.30-inch diameter finished fuel rods, it is safe to handle 26 rods (920 g contained U235). A two-dimensional infinite array of fuel elements (24-rod bundles) is safe with a minimum edge-to-edge separation of 24 inches. This same spacing between safe batches of pellets is also safe. Fifty columns (2.75 inches by 2.75 inches cross section) of fuel elements are safe with a minimum edge-to-edge spacing of 12 inches. (auth)

123.

Criteria were established for the storage and transportation of a maximum of one kg contained U235 scrap in each 55-gal drum. These drums are stored four drums per pallet in a square lattice array. An array of such pallets, two pallets wide by two pallets high by six pallets long containing a total of 96 drums, is safe independent of the degree of water flooding. Two kg contained U235 is allowable per drum in a 180-drum, single-plane array. These arrays may be repeated with an edge-to-edge separation of at least 5 ft. (auth)

124.
6591 (HW-68656(p.143-7)) CRITICAL APPROACH AND EXPONENTIAL MEASUREMENTS WITH 3.1 PER CENT ENRICHED URANIUM RODS IN LIGHT WATER. R. C. Lloyd, E. D. Clayton, and R. B. Smith (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A series of exponential and critical approach measurements were previously made with 3.063% enriched uranium rods in water. The data are summarized, and plots are given of maximum buckling and minimum critical mass in spherical geometry vs. rod diameter and the extrapolation length vs. H2O/U volume ratio. (W.D.M.)

125.
7004 (HW-68666(p.149-8)) EXPONENTIAL EXPERIMENTS WITH POISONED MODERATOR. R. C. Lloyd (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A series of measurements was completed for determining the amount of neutron absorber required to make a heterogeneous system of 1.007% enriched uranium rods safe. The fuel rods (aluminum clad) were 0.925-in. in diameter and 44-in. long. Eighty-five of the fuel rods were loaded into a stainless-steel tank, and increments of boric acid (H3BO3) were added to the moderator until the buckling became negative. (W.D.M.)

126.
6592 (HW-68666(p.150-1)) EXPONENTIAL MEASUREMENTS OF NATURAL URANIUM-WATER SYSTEMS. R. C. Lloyd (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A series of exponential measurements was started with 0.925-in. diameter natural uranium fuel elements in water. The purpose was to obtain further data on the critical mass of natural uranium in light water and to obtain buckling values for the interpretation of planned experiments involving a 5% enriched uranium lattice reflected by a natural uranium tamper. (W.D.M.)

127.
2064 HW-65552.


Contract W-31-109-eng-52. OTS.

A listing of all measurements on water-uranium heterogeneous lattices made at Hanford is presented. All lattices were water-moderated hexagonal arrays loaded with uranium enrichments up to 3.1%. (J.R.D.)
and
6997 (HW-66215(p.30-3)) CRITICAL APPROACH AND EXPONENTIAL MEASUREMENTS WITH 2.00 PER CENT ENRICHED URANIUM RODS IN LIGHT WATER. R. C. Lloyd, E. D. Clayton, and B. L. Jones (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Critical approach measurements were completed with 2.00% enriched, 0.925 in. diam rods 16 and 32 in. long; exponential measurements were made with the 32-in. rod length. The rods were incased in lucite tubes for insertion in the water reflected lattice assemblies. The critical number of rods in cylindrical geometry and critical mass values for spherical geometry were determined. (W.D.M.)

6998 (HW-66219(p.34-8)) EXPONENTIAL MEASUREMENTS OF NATURAL URANIUM–WATER SYSTEMS. R. C. Lloyd (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A series of exponential measurements was completed with 0.925-in. diam. natural uranium fuel elements in light water. A plot of buckling vs. \( \frac{H_2O}{U} \) volume ratio is given. (W.D.M.)


A series of approach-to-critical type experiments were conducted to obtain information needed for the design of efficient systems for processing spent power reactor fuel elements. The series included experiments such that the lattice spacings spanned maximum buckling and minimum critical mass. Maximum buckling and minimum critical mass data are tabulated for the three fuel rod sizes tested. (B.O.G.)

131. 7003 (HW-64866(p.137-42)) NEUTRON MULTIPLICATION MEASUREMENTS WITH Pu–Al ALLOY RODS IN LIGHT WATER. V. I. Neeley, R. C. Lloyd and E. D. Clayton (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Neutron multiplication and exponential experiments were conducted with Pu–Al alloy rods having 5 wt. % plutonium enrichment. The rods were 24-in. long and 0.506-in. in diameter and were clad with 0.030-in. ZircaloY-2 with 0.020- and 0.125-in.-thick end caps. The average rod contained 11.01 g of plutonium. This gave a \( Zr/Pu \) of 31.92 and Al/Pu of 168.20. Experiments were conducted in the TTR reactor room. Experiments were conducted using critical approach, “back-off,” and exponential measurement techniques. (W.D.M.)


Natural uranium fuel rods of equal strength made of uranium metal and \( U_3O_8 \) are investigated as to their maximum excess reactivity. The result is that with high moderation, a maximum reactivity gain of 0.020 is obtained with uranium metal as compared to \( U_3O_8 \). As regards the maximum excess reactivity, a metallic natural uranium fuel rod is equivalent to a \( U_3O_8 \) fuel rod of equal strength enriched to 0.748% uranium–235. (auth)


Neutron multiplication measurements were made on cylindrical assemblies containing layers of plutonium metal disks, Flexiglas disks, and boron carbide impregnated Epolene-n disks. In addition to the above nuclear safety measurements, curves were drawn for a 42-in.-diameter stainless-steel tank containing an aqueous solution of \( U_3O_8(NO_3) \) and poisoned with Pyrex Raschig rings. Attempts were made to calculate sphere, infinite cylinder and slab shapes from the experimental finite cylindrical assemblies. (auth)


Measurements are made of the buckling of diphenyl-cooled, heavy-water–moderated, natural U fuel clusters. Seven- and nineteen-rod clusters are used. The buckling is measured as a function of the number of rods in the cluster, the rod diameter, the cell radius, and the minimum distance between the rods. The measurements are carried out in the ZERLINA assembly. (T.F.H.)


The macroscopic reactor core parameters determined by exponential experiments (buckling, reflector saving, relaxation length, and migration area) are reviewed. The most significant experimental results, some still unpublished, are compared with the correspondent theoretical values. (auth)
lattices 136-142


Some lattice parameters of a natural U, light water subcritical assembly were measured. Results obtained with and without radial reflector are compared. Values of $B^2$ for the bare assembly and reflector savings for the reflected assembly are given. Fine distribution of thermal flux throughout a cell was measured and, the thermal utilization factor, determined. (auth)


Lattice parameters for a concentric tube fuel element were measured in the Physical Constants Test Reactor (PCTR). The measurements reported include $k_w$, $f$, $p$, and $x$ for a 10$^{3}$-inch graphite lattice with both water and air in the coolant channels, and $k_w$ and $f$ for an $8^{3}5$-inch lattice, water cooled only. The value of $\eta$ derived from the $10^{3}$-inch lattice measurements is 1.30. Measured fluxes are compared to $P_2$ calculations using an adjusted neutron temperature. Some of the correction factors and sources of error in the measurements are discussed. (auth)


139. 7085 (HW-66215p.14-17) LATTICE PARAMETERS FOR THE EXPERIMENTAL GAS COOLED REACTOR. J. R. Worden and P. F. Nicholas (General Electric Co., Hanford Atomic Products Operation, Richland, Wash.).

The final experiment of a series in the PCTR, designed to support the EGCR program, has been completed. Values of $k_w$ and $f$ were measured using standard PCTR techniques. The fuel consisted of 2.6 wt. % enriched UO$_2$. Results for a 21.875-in. cell are given. (W.D.M.)

140. 18853 THEORETICAL ANALYSIS AND EXPERIMENTAL RESULTS FOR C-H MODERATED ASSEMBLIES. G. B. Zorzoli (CISE, Milan). Energia nuclear (Milan), 8: 255-60(April 1961). (In English)

A survey is made of the exponential experiments on natural uranium lattices having a hydrogenous moderator, with particular reference to organic moderators. The measured bucklings for diphenyl moderated lattices appear to be high; a comparison is made between previous measurements and the experimental values obtained at CISE on impregnated graphite moderated lattices. A theoretical analysis confirms the presence of a systematic error in the measurements on diphenyl moderated lattices. This error can be shown to result from the contributions of spurious epithermal neutrons. (auth)

1962


For a cylindrical natural uranium and light water subcritical system, an optimum cell and uranium rod radius was found among all the cells and rod radii studied: $k_w = 0.97$, $r^3 = 1$ cm, $r^1 = 1.8$ cm. The feasibility of interchanging natural uranium rods from the U-H$_2$O subcritical system to a U-D$_2$O critical zero power system was shown possible utilizing only optimum diameter rods. The effect of an air gap around the uranium rod, for the light water system, was studied and it was shown that for a given cell and rod radius, there was an air gap thickness that optimized the infinite multiplication factor. Some estimates on the masses of both the subcritical natural U-H$_2$O and the critical U-D$_2$O systems are given. Monoenergetic diffusion theory was used and all computations were done on the IBM-650 computer of the University of Mexico. (auth)


Work performed on the Spectral Shift Control Reactor Basic Physics Program is summarized. The major objective of this program is to study the nuclear properties of slightly enriched lattices moderated by D$_2$O-H$_2$O mixtures. Critical experiments were performed with 4$\frac{3}{8}$-enriched UO$_2$ fuel rods in a lattice having a non-moderator-to-moderator volume ratio of 1.2, and moderated by a 49.7% D$_2$O-H$_2$O mixture. Measured were critical mass, $dp/dh$, buckling and reflector savings, and cadmium ratios of U$^{235}$ and U$^{238}$. Small systematic errors in the cadmium ratios were evaluated by correlating the measurements over a range of
moderator compositions from zero to 70.1% D_2O. The critical parameters of the first assembly with 2.46%-enriched UO_2 fuel rods are reported. The lattice had a non-moderator-to-moderator volume ratio of 1.0, and was moderated by light water. Physical and chemical properties of the 2.46%-enriched fuel are summarized. The objectives, conceptual design, and final design details of the small lattice experiments are given. These experiments will test the applicability of the PCTR technique to lattice measurements in epithelial systems. Also presented are improvements to the BPG code: a neutron source is now permitted in each group and a standard set of thermal cross sectional data is incorporated. Previous analytical methods were used to check the most recent critical experiment data. The results of a systematic analysis of other light water moderated systems, using methods developed for D_2O-H_2O moderated lattices, are summarized. Design calculations for the small lattice experiments and the reactivity-period relation for critical assemblies using 2.46%-enriched UO_2 fuel are given. (auth)


The results of exponential experiments performed under the Spectral Shift Control Reactor (SSCR) Basic Physics Program are summarized. The material buckling and cadmium ratio of U^{235} were measured at room temperature in rod lattices moderated by D_2O-H_2O mixtures in which the non-moderator-to-moderator volume ratio was approximately 1.0. The two types of fuel studied were 2.46%-enriched UO_2-ThO_2 pellets (N_{Th}/N_{U} = 13) clad in 0.308-inch OD aluminum tubes and 4% enriched UO_2 swaged in 0.476-inch OD stainless-steel tubes. The results are shown in the following tabulation, where the measured cadmium ratio of thorium in the first lattice is 1.14 ± 0.02. Also given in design information on the Lynchburg Source Reactor (LSCR), which was assembled and operated at 1000 watts to provide neutrons for these experiments. (auth)


Buckling measurements were made in the Process Development Pile (PDP) in D_2O-moderated lattices of tubular natural uranium fuel assemblies, placed at a triangular pitch of 7 inches. The assembly types studied included four sets of single fuel tubes and six sets of double tubes. Bucklings were determined by an analysis of the critical water heights obtained when seven test assemblies were inserted into the central test region of the PDP. The effects of removing the central fuel assemblies and of replacing them by varying numbers of poison rods were also studied. Fuel activation measurements of the lattice param-


Buckling measurements were made in ZED-2 in a series of hexagonal lattices of 55 7-element UO_2 clusters with spacings from 18 cm to 36 cm. Three materials in the cluster coolant region were investigated: D_2O of moderator purity; helium, to simulate a voided condition; and an organic liquid, HB-40. The experimental results were compared with values calculated using the latest Chalk River lattice recipes. Buckling measurements made previously with these 7-element UO_2 clusters in ZEEP are in agreement with the measurements made in ZED-2. A short description of the ZED-2 reactor is included. (auth)


The results of several critical experiments performed on the Ispra I reactor are discussed and compared with calculations. Two-group theory is shown to yield accurate results in most cases. The relevance of the various effects, in experiment and calculation, is analyzed. (auth)


A semi-empirical method for calculating material bucklings for slightly enriched uranium tubes in light water is presented. Lattice parameters were calculated by the IDOT code. Buckling was calculated by the one group critical equation, using an adjusted regeneration factor (eta) to obtain a correlation to measured values. Measured bucklings on tube lattices at 0.95, 1.0, 1.25, 1.44, 1.47, and 1.63% U^{235} were used in the correlation. The calculated and measured maximum bucklings at these enrichments agreed to within ±350 10^-8 cm². A curve of adjusted eta versus U^{235} enrichment is given. The reliability of the method is limited to the range of tube sizes and U^{235} enrichments used in the correlation. Beyond this range, the method is less reliable, but useful for lattice survey studies. (auth)
LATTICES 149-153

152. 8301  (B&W-1231) SPECTRAL SHIFT CONTROL REACTOR BASIC PHYSICS PROGRAM. CRITICAL EXPERIMENTS ON LATTICES MODERATED BY D2O—H2O MIXTURES. L. L. Businaro (FIAT, Sezione Energia Nucleare, Turin), Energia nucleare (Milan), 9 : 460-6 (July 1961). (In English)

The results of critical experiments performed under the Spectral Shift Control Reactor Basic Physics Program are summarized. Nine major critical assemblies of rod lattices were studied in moderator mixtures of light and heavy water ranging from zero to 81.2 mole % D2O. In some assemblies, the moderator was poisoned with boron acid. The non-moderator-to-moderator volume ratio in all lattices was approximately 1.0. The fuel in most lattices was 4%-enriched UO2 swaged in stainless steel, although two experiments were performed with 93%-enriched UO2—ThO2 pellets in aluminum tubes. One assembly was zone-loaded radially and contained both types of fuel. The critical mass, D2O concentration, boron concentration, buckling, thermal disadvantage factor, and cadmium ratio of U235 were measured in each assembly. In most assemblies, the cadmium ratio of U235 or Th233 was measured, and in five assemblies, the epithermal neutron spectrum was derived from the measurements taken of the resonance activity of detector foils. In special experiments at high D2O concentrations, the perturbations by moderator gaps and control blades were studied, and the reflector savings versus reflector thickness was measured. The flux distribution in the zone-loaded assembly was also mapped. The measured lattice parameters are summarized. (auth)


In order to obtain information on the isotopic composition of irradiated fuel elements and on certain other characteristics of the Pu oxide-fueled BR-5 fast reactor, a portion of the U fuel was investigated after a 100-day long exposure to an integrated flux of about 5 × 1021 n/s/cm2. The center of the material examined was located 12.6 cm from the center of the reactor. The number of fissions was calculated from the absolute Cs137 activity while the number of captures in U238 was derived from the specific activity of samples taken from various locations. The neutron leakage in the reactor was high as a result of the many Na-filled channels. The concentration of Pu239 which was found to be about 0.13% was determined by comparing the number of spontaneous fissions in test samples and in reference samples using an ionization chamber. The experimentally obtained ratio of the average capture and fission and capture cross sections agrees well with the value of 1.93 calculated from 18 neutron spectrum group constants and with data obtained with natural U. The average capture cross section of Pu239 was found to be equal to 0.415 ± 0.035 barn while the capture and fission cross section ratio was 0.19 ± 0.02. The value of this ratio

154. 1205 MATERIAL BUCKLING OF HEAVY WATER LATTICES. STATISTICAL FIT TO EXPERIMENTAL DATA. U. L. Businaro (FIAT, Sezione Energia Nucleare, Turin), Energia nucleare (Milan), 8 : 149-53 (July 1961). (In English)

Available experimental data on material buckling for heterogeneous, heavy water lattices were statistically analyzed with quadratic polynomial correlations. The data were divided into four groups: metal, single bars; metal, bundles of plates; oxide, bundles of bars. For each group correlations were obtained and compared. In the case of bundles-of-plates lattices, three different definitions for the effective bundle surface were tested. For the case of bundles-of-bars lattices, the difference between experimentally and theoretically determined values were statistically analyzed and a statistically significant deviation between experimental and theoretical values was indicated. (auth)

150. 31002 CRITICAL EXPERIMENTS ON NATURAL URANIUM OXIDE, ORGANIC COOLED, HEAVY WATER MODERATED LATTICES. G. Casini, C. Foggi, and F. Toeselli (Euratom CCR, Ispra, Italy). Energia nuclear, (Milan), 9 : 455-61 (Aug. 1962). (In English)

A series of critical experiments was carried out at Saclay, France, by the Reactor Physics Department of the C.C.R., Euratom. The purpose of the measurements was to determine the material buckling of nine different ORNEL lattices using natural uranium oxide fuel elements, moderated by heavy water and cooled by an organic liquid. The experimental technique, the results obtained, and their comparison with calculated values are described. The latter turn out to be systematically lower than the former. Some criteria are given to improve the agreement between the two sets of values. (auth)


A series of buckling measurements was undertaken in wet and dry graphite lattices at spacings of 4.75 to 10.75 in. for U235—aluminum fuel elements for U235 compositions of 4% and 14.45 g/cm3 by means of critical size from multiplication measurements, exponential methods, and quasi-exponential methods. Wet lattice bucklings, at small spacings, are considerably higher than the dry lattice bucklings, because of the smaller age in the lattices which overcompensate for the increased thermal leakage and decreased k. The extrapolation length and the graphite reflector savings for a rectangular prism of 4 x 4 ft cross sectional dimensions were determined for the lattices measured. (I.O.G.)
decreased rapidly from the center of the core toward the periphery of the core because of the influx of neutrons into the core which were slowed down by elastic and inelastic collisions in the Ni screen. (TTT)

154.

1992


This paper was previously abstracted from the original language and appears in NSA, Vol. 16, abstract no. 11164.

155.

1224


The minimum critical loading of the SRE with Th-U fuel and Na in the core at 340°F was found to be 30.6 ± 0.2 fuel clusters. The operational loading, defined as criticality with two of the four shim rods in the core, was found to be 40.6 ± 0.2 fuel clusters. An evaluation of various plotting techniques for critical experiment data is given. It was found that plotting the inverse multiplication as a function of radial buckling, using the physical radius plus reflector savings, gave the most linear approach to criticality. (auth)

156.

8310


Experiments were performed in the Process Development Pile (PDP) with a critical mockup of the Heavy Water Components Test Reactor (HWCTR) lattice. Reactivity measurements served to determine control rod worths, the temperature coefficient of the reactor at low temperatures, the dependence of k_eff on concentrations of U^{235} and poisons in the “driver” region of the reactor, and the changes in k_eff that occur in certain conceivable accidents in the HWCTR. Flux distribution measurements were made to determine over-all flux shapes, local flux variations around individual components, asymmetrical flux variations due to asymmetrical combinations of control rods, and vertical flux variations obtainable through the use of control rods of partial length. Sufficiently detailed flux measurements were made to permit the calculation of neutron economy tables and to determine the neutron leakage from the reactor. Suitable adjustments in the input parameters made it possible to fit the experimental results with a two-group calculation. (auth)

157.

33580


Performance tests were carried out on a circuit designed to distinguish between recoil electrons and recoil protons, due to incident photons and neutrons, respectively. In a hydrogenous scintillator. The activities produced in lutetium foils by irradiation were found to be a function of thickness. Data are presented for correction of lutetium activity measurements from thick foils to that for thin foils. Subcritical experiments with 1.82 wt % Pu–Al rods in light water were conducted. A series of criticality experiments were begun with plutonium–nitrate solutions in a 14-in. diameter stainless steel sphere fully reflected with water. Monte Carlo calculations were made of the limiting critical concentration of Pu^{239} and U^{235} in aqueous solutions. (M.C.G.)

158.

757


159.

12683

EXPERIMENTAL BUCKLINGS OF HEAVY WATER MODERATED LATTICES OF NATURAL URANIUM METAL ROD CLUSTERS. T. J. Hurley, Jr., H. R. Fike, and G. F. O’Neill (E. I. du Pont de Nemours & Co., Aiken, S. C.). Nuclear Sci. and Eng., 12: 341-7(Mar. 1962). Studies performed in the Process Development Pile of the Savannah River Laboratory provide precise measurements of the material bucklings of a number of D_2O-modulated lattices of natural uranium metal rods over an extended range of fuel assembly sizes and lattice pitches. The 1-in. uranium rods are clad with 0.032 in. of aluminum. Fuel assembly sizes vary from single rods to clusters of 3, 7, and 19 rods (0.09 to 1.81 kg U/cm) and lattice spacings from 7.00 to 21.00 in. covering a range of moderator-to-fuel volume ratios from 10.23 to 161.53. A few lattices are studied at different D_2O purities, in loadings of different sizes, and in reflected loadings. (auth)

160.

4858


Nuclear safety criteria were established for the fabrication, storage, and transportation of UO_2 fuels having an enrichment of 2.0 and 3.0% U^{235}. Twelve figures are included summarizing the results of this study. The following criteria were established for the 3.0% U^{235} enriched fuel and may be safely applied to the 2.5% U^{235} enriched
Lattices 161-166

UO₂ fuel of interest. The maximum safe batch size of UO₂ independent of container size or degree of water flooding is 1.22 kgm contained U²³⁵. Safe batch sizes as a function of container volume are also presented. The maximum safe geometries are 27 liters for the sphere volume, 9.75 inches for the infinite cylinder diameter, and 4.18 inches for the infinite slab thickness. The maximum safe numbers of rods, independent of container size or degree of water flooding, are 145, 110, and 90 for 0.30-inch, 0.34-inch, and 0.38 inch diameter rods, respectively. Safe criteria are also given for the spacing between batches of pellets or rods. Safe criteria were also established for the 2.0% U²³⁵ enriched UO₂. The maximum safe batch size, independent of container size or degree of water flooding is 1.99 kgm contained U²³⁵. Safe batch sizes as a function of container volume are also presented. The maximum safe geometries are 52 liters for the sphere volume, 12.5 inches for the infinite cylinder diameter, and 5.95 inches for the infinite slab thickness. The maximum safe numbers of rods, independent of container size or degree of water flooding, are 290, 230, and 180 for 0.30-inch, 0.34-inch, and 0.38-inch diameter rods, respectively. Safe criteria are also given for the spacing between batches of pellets or rods. (auth)

161.


Nuclear safety criteria are given for the handling of 1 1/8-in. by 1 1/4-in. compacts having a composition of 10 wt.% highly enriched U and 90 wt.% Zr. The maximum safe batch size for these compacts is 1.68 kg contained-U²³⁵, independent of spacing between compacts or the degree of water flooding. For long compacts the maximum safe mass per unit length is 0.94 kg contained-U²³⁵/ft of length. This, too, is independent of spacing between compacts or the degree of water flooding. Maximum safe batches containing no more than 1.68 kg U²³⁵ each are safe with a minimum edge-to-edge spacing of 20 in. (auth)

162.


Measurements were made of the nuclear parameters of heavy-water-moderated lattices of uranium metal rods, 3 inches in diameter and enriched to 3.0 wt.% U²³⁵, at a triangular lattice pitch of 18 inches. The value of k∞, obtained from the four-factor formula with experimental values of f, η, ε, and p, agreed to within 0.6% with the value obtained from the two-group diffusion kernel together with experimental values of L² and Bn². The value of k∞ obtained from the four-factor formula using experimental values of the parameters agreed within 0.5% with that obtained by inserting calculated values, but the measured and calculated values of the individual parameters differed by as much as 3%. (auth)

163.


164.


The present work intends to clarify the question of whether material bucklings of D₂O—natural uranium lattices can be measured with precision in subcritical two-zone systems. The modified two-group formalism developed in this report was checked in two different experimental series. The results of the analysis of the experiments are very satisfactory and encourage similar studies under different conditions. In the last section, a special set of orthogonal functions for two-zone systems is presented and correlated with some experiments. (auth)

165.


Uniform lattice critical experiments with 3 and 4% enriched uranium dioxide, clad with stainless steel, were performed in ordinary water. These experiments were done with fuel pins, 0.500 inches O.D. with an active fuel length of 66.5 inches. The fuel pins were fabricated by swaging the UO₂ powders to 87% theoretical density, in stainless steel tubing. The final wall thickness was 0.0282 inches. The pitch spacings which were used and the volume fractions of materials present in the core are listed. The minimum critical radius obtained for the fully reflected core is shown as a function of the ratio of water to equivalent uranium metal. Measurements of the differential reactivity as a function of water height for each lattice yielded the common straight lines of differential reactivity as a function of water height. Numerical integration of the differential reactivity curve yields the excess reactivity as a function of core radius. (auth)

166.


An engineering critical experiment of the Pressure Tube Reactor (PTR) core using the facilities of the Industrial Reactor Laboratories pool-type research reactor is described and some of the results are presented. The ex-
Experiment was planned and performed so that both engineering and physics objectives would be realized simultaneously and costs minimized. Successful data are obtained on the following engineering and physics objectives: control rod performance; performance of fuel handling tools; compatibility with pool reactor instrumentation; general suitability of component design to a pool type installation; mass coefficients; cadmium ratios; critical mass; control rod worths; reactivity effects of void, boron, aluminum, and stainless steel; and detailed flux distributions. (N.W.R.)

167.

2398


A description and results of a second series of critical experiments performed on the SM-2 core mockup are presented. The SM-2 core mockup contains 36.4 kg of U enriched to 92% and an estimated 67.9 g of B10. The equivalent diameter and the active height are about 22 in. The metal-to-water volume ratio is 0.344. Data are presented on activation, reactivity, and control rod measurements. All measurements were conducted on the open 7 control rod array employing 38 stationary fuel elements. Activation measurements consisted of neutron flux measurements using uranium fission foils for relative power distribution studies, the effect of flux suppressors on reducing power peaks, blocked coolant channel measurements, and gamma dose distribution. Reactivity measurements were performed to determine the effect of flow divider, flux suppressors, and simulated high temperature and pressure operation, B10 loading in the SM-2 core, and core material coefficients. For the latter, the worth in cents per gram or cents per cm3 was determined at a simulated temperature of 510°F for B10, U233, stainless steel, and void. Stuck rod measurements were made to obtain an indication of the criticality margin in the event one or more control rods should stick in the operating position. (auth)

168.

11170


In an exponential experiment the multiplication characteristic magnitudes of various heavy water moderated reactor assemblies were measured. The lattice studied consisted of a quadratic arrangement of cylindrical natural uranium rods (32 mm diameter) with 1 mm thick aluminum can (fuel rods of the FR-2 Research Reactor); the lattice distances d were varied in the range from 10.8 to 24.0 cm. By measurement of the macroscopic flux distribution within a tank of 136 cm diameter, the material formfactor "material buckling" B0 was determined with the help of a BF3 counter. These measurement values were in good agreement with those obtained by Cohen on similar assemblies. By measurement of the microdistribution of the thermal flux within a lattice cell with small Mn probes, the ratio of the mean flux value in the moderator and fuel element and therefore the thermal utilization f and the effective diffusion length L were determined. The ratio found is about 2 to 3% higher than the corresponding value found by Cohen. From the magnitudes B0, L, and the Fermi age, the multiplication factor k0 was calculated, and the value η = 1.307 ± 0.015 was found by an extrapolation method given by Mummery. (tr-auth)

169.

4679


Neutron multiplication and exponential measurements were conducted with Pu-Al alloy fuel elements in light water moderated lattices; a hexagonal pattern was used for the lattices which were fully water reflected. The critical mass was determined for 24-inch high cylinders by neutron multiplication measurements. The extrapolation length and bucklings for the lattices were determined by equating the buckling expression from the exponential measurements to the buckling expression for the critical size as determined from the neutron multiplication measurements. These equations were then solved for the extrapolation length and buckling. The critical mass for spherical geometry was calculated from the measured buckling and extrapolation length. The minimum critical mass for the Al-5 wt % Pu alloy rods in light water was 1.5 kg plutonium. The maximum buckling was 11,300 × 10-4 cm-1. The effect on the critical mass of the Pu244 (~5%) was determined from calculations; the results indicate the critical mass (including all isotopes) would be reduced by about 8.2% in the absence of Pu244, or to 1.38 kg Pu239. A curve of the critical mass and buckling obtained from these experiments is shown. The various data are tabulated. (auth)

170.

15525


The experimental and analytical bases of the determination of the material bucklings of uranium-D2O lattices are presented. Techniques which were developed, particularly with the intent of measuring material bucklings in the MIT lattice facility, are described. The design considerations and experiments dealing with the spatial distribution and magnitude of the neutron source in the lattice facility are discussed. The source distribution was analyzed as it entered the subcritical assembly tank when the tank contained only D2O and when the tank contained a lattice of uranium rods in D2O. The detailed investigation of the over-all flux distributions in lattices included a study of the non-

Lattices 167-170
Lattices 171-175

separability of the macroscopic and microscopic radial distribution. A set of computer codes was developed to reduce and analyze fully the data from flux distribution measurements. The bucklings of three lattices of 1.016-inch diameter natural uranium rods in D_2O were measured. These measurements are shown to be in good agreement with measurements made in similar lattices at other laboratories. (auth)

171.


Buckling measurements on clusters of 19 UO_2 rods in heavy water were performed in an experimental assembly and by means of substitution measurements in a critical facility. The material buckling was determined as a function of lattice pitch (range of V_m/V_n: 7 to 22), internal spacing, void, and temperature (20 °C < T < 90°C). The change of diffusion coefficients (about 6 to 8%) caused by voids was studied with single test fuel assemblies. The progressive substitution measurements were analyzed by means of a modified one-group perturbation theory in combination with an unconventional cell definition. The buckling differences between test and reference lattices were of the order of -1.0 to -3.5. The results of the exponential and the critical experiments were compared with similar measurements on the same kind of fuel. This comparison showed that the results of the various experiments agree quite well, whereas theoretical predictions fall in the extreme range. (auth)

172.


A critical experiment program was undertaken to supply the basic physics information, over a wide range of water-to-fuel volume ratios, which is necessary to predict with confidence the nuclear behavior of superheat cores utilizing slightly enriched annular oxide fuel. The experimental measurements included were: critical size, (dp/dH) versus H (p = reactivity, H = water height), void coefficient, temperature coefficient, flux distributions, thermal utilization, and conversion ratio. All the measurements were carried out utilizing uniformly spaced arrays of fuel (no controls or water gaps). The experimental results were compared to the prediction of engineering design methods as well as more detailed calculations. Based on these comparisons, a reliable engineering design model was developed for this lattice type. (auth)

173.


The nuclear characteristics of a variety of small reactors composed of thorium—uranium fuel in heavy water were determined in a program of critical experimentation. The fuel element consisted of ceramic ThO_2–UO_2 pellets stacked to a height of 1.5 m within 0.787-cm-OD aluminum tubing. The pellets used most frequently were of 0.587-cm diameter and had a Th/U_235 atom ratio of 25. Rods containing similar pellets with only half as much U_235 were used to achieve small changes in the U_235 content of the cores.

Some cores were assembled with 0.660-cm-diameter pellets having an atom ratio of 15. All cores were located in a 2-m-diameter tank containing D_2O. Three distinct core structures were used, allowing measurements with uniformly distributed fuel rods, loading patterns compatible with the EBWR core geometry, and clustered lattice arrangements with D_2O, H_2O, and air surrounding the clustered fuel. Most of the cores assembled had some amount of radial D_2O reflector. A 0.3-m-thick bottom reflector composed of D_2O and aluminum was always present. For most of the assemblies, the control rods were fully withdrawn, criticality being achieved by adjustment of the water level. Observed critical dimensions and the results of the conventional reactivity, foil activation, and fuel substitution experiments used to determine core parameters such as p, c, f, and r, are reported. Information on void temperature coefficients and control rod worth is included. No comparison of the experimental results with theoretical predictions is made nor are observations made on flux-trap systems of this composition included. (auth)

175.


Exponential experiments and theoretical calculations were performed to determine the reactivity effect of expelling D_2O coolant from D_2O-moderated lattices of natural uranium fuel assemblies composed of coaxial metal tubes. Both the exponential experiments and the calculations confirmed independent critical measurements in showing large increases (up to 4%) in kopt when water is expelled from the coolant channels. (auth)
176.


An empirical equation was derived which relates the surface-to-volume ratio of a fissile unit (enriched uranium metal ~90% U) in lattices immersed in water with the array minimum critical mass. The critical mass data used in this study were developed at the Los Alamos Scientific Laboratory. (auth)

177.


178.


A survey is given of facilities, experimental data, and theoretical interpretations of D₂O-moderated exponential experiments utilizing UO₂ or ThO₂–UO₂ mixtures as fuel. (T.F.H.)

179.


An experimental and theoretical program on the physics of heavy water moderated, partially enriched uranium metal lattices is being conducted. A subcritical assembly which uses the M. I. T. research reactor for the neutron source has been built. Theoretical and experimental research on buckling, fast fission, resonance capture, and thermal capture has been carried out for the calibration lattice of one-inch diameter natural uranium rods. Programs in pulsed neutron source research and reactor control research have been initiated. (auth)

180.


181.


Buckling measurements of 42 lattices of natural uranium 1-in. rods in heavy water were made in an exponential facility at two moderator purities. The fuel assemblies were single rods and clusters of 3, 7, and 19 rods. Lattice pitches varied from 3.0 to 21.59 in. A comparison was made between the bucklings that were measured in the exponential facility and those that were obtained from critical measurements. On the basis of a constant radial buckling for the exponential, systematic differences between the exponential and critical measurements were noted. Possible causes of these differences were discussed. Changes in buckling produced by changes in moderator purity were also given. (auth)

182.


Measurements of material bucklings for graphite–uranium systems are summarized. A comprehensive listing and guide to the original data sources is provided. Complete information on physical and nuclear properties of the lattice and the geometry of the exponential assembly is included, along with some of the auxiliary data taken. The fuel sizes vary from 0.925 to 2.5 in. diameter for five different fuel geometries. The lattice spacings vary from 4/10 to 15 in. Over 300 measurements of material buckling are included. (auth)

183.

6920 BUCKLING MEASUREMENTS IN URANIUM–DIPHENYL LATTICES. G. B. Zorzoll (CISE, Segrate, Italy). Energia nucleare (Milan), 8: 780(Dec. 1961). (In English)

An experiment is described that yields a positive buckling value for a natural U-diphenyl lattice. It is shown that systematic errors, insufficient measurements, and unfavorable vessel dimensions render the experimental results incorrect. (T.F.H.)

184.


The results of critical experiments on EK-10 fuel ele-
LATTICES 185-188

...ments with the use of the organic compound, monododecapolyclayvinyl, as a moderator are presented. The experiments were carried on a square array at spacings of 16, 19, 22, and 25 mm. The critical loadings at these spacings, and the change in critical loading on heating the assembly to 50°C were obtained. Measurements of the distribution of the thermal neutron flux were also carried out along the radius and the height of the active zone. The experimental data were compared with calculated values. Moreover, for comparison, experimental determinations of the critical loadings for the FK-10 elements were carried out at lattice spacings of 16, 19, and 22 mm with ordinary water as the moderator. In the second part of the program, experimental determinations of the effectiveness of regulating rods of different configuration were carried out as well as determinations of the effect of replacing part of the reflector with a water-beryllium reflector on the magnitude of the critical loading and on the thermal neutron distribution. (auth)

1963

185.


A simple interaction theory is presented for calculating safe arrays of fissile units of arbitrary size, shape, and distribution. At every stage of the theory an endeavor was made to err on the side of safety. Mixed arrays of fast and thermal units are considered. The reflecting material and void present between units may be well defined, as for example in certain kinds of fissile store, or it can be random as in plants where personnel and portable equipment may be in close proximity to vessels. The theory is compared, for the case of well-defined reflectors and fissile units of one type, with the results of three interaction experiments made; on a pair of 20-kg spheres of uranium immersed in water, on arrays of similar spheres in a concrete vault, and for 3- and 7-unit arrays of cylinders containing aqueous solutions of uranyl fluoride. The theoretical results are generally more conservative than the experimental ones, but not to an unduly pessimistic degree. It is not feasible to test a novel feature of the theory, in that it can deal with the case of random or unknown distributions of reflecting material. Since the experiments for well-defined reflectors indicated that the assumptions of the theory do err on the side of safety, the predictions for random reflectors are considered to be conservative. These estimates are somewhat greater than those for well-defined reflectors, but the increases are insufficient to cause real difficulties in the design and cost of plant. To those dealing with large numbers of interaction problems the theory may prove to be attractive on four accounts. The good agreement obtained so far with experiment suggests that the theory is reasonably accurate. Unlike Monte Carlo methods, which in principle are more accurate, the method requires very little computation; (as many as thirty

cases of an interaction problem can be dealt with in five minutes on a medium size computer such as Mercury). Unlike the interaction parameter method it does not require an experiment on a simple assembly of units in order to make a prediction for a complicated array of units. The theory makes predictions using the same core and reflector constants as employed in ordinary reactor calculations for single units. (auth)

186.


Critical experiments in the active zone of VVER showed that homogeneous multiplying lattices are well described by the critical equation up to the material parameter $x_2 = 60 \times 10^{-8} \text{ cm}^{-1}$. The magnitude $\eta$, describing the deviation of critical conditions from one-group, was equal to 1.04 $\pm$ 0.08. Results of experimental determinations of $K_n$ and $x_2$ for an assembly of 1.5 and 2% enriched fuel elements and for a mixed assembly were in good agreement with theoretical calculations. Variation in absorber efficiency indicates that fuel rod distribution effects reactor geometry parameters. (R.V.J.)

187.


The extent to which the reactivity of cores uniformly loaded with natural Hf may be calculated is evaluated. Two series of critical lattices are used; in one lattice Al is employed alone, while in the other, Zircaloy plates are used instead of those made of Al. (J.R.D.)

188.


Buckling measurements were made in the Savannah River Laboratory. Exponential facilities on D,O-moderated, 19-rod clusters of natural uranium oxide rods. Four triangular lattice pitches from 7.00 to 12.12 inches and a single square pitch of 8.00 inches were used. Measurements
were made at room temperature and at selected temperatures up to 218°C. Buckling changes resulting from voids inside housing tubes that surrounded the clusters were also measured. Intercomparisons of the results with critical void coefficient measurements demonstrated the validity of calculations for the anisotropic diffusion effects of the voided lattices. Foil activation measurements of the lattice parameters f, L^2, p, and c were made for representative lattice configurations. The experimental results are compared with calculations. (auth)

189.


The buckling values for 0.5-in.-dia, 1.8 wt % Pu-Al alloy rods in light water, reported in HW-74190, were used to estimate the critical parameters for water-moderated and reflected arrays of these elements. The values obtained are: minimum critical mass, 2.28 kg Pu; minimum critical volume, 445.4 liters; minimum critical mass/unit area, 41.8 lbs of alloy/ft^2.

The maximum material buckling obtained from the measurements was 0.00613 cm^-2 at a water-to-alloy volume ratio of 1.55. The minimum critical mass occurs at a water-to-alloy volume ratio of 2.0. The above values will be used as the basis for nuclear safety evaluations concerning the handling and storage of these fuel rods. (auth)


A simple method for computing the interaction in a critical grouping of units, each of which would be subcritical if isolated from the others, is reviewed. Calculations by this method are compared with experimental results for reflected and unreflected cubic arrays with various numbers of units. (D.C.W.)


A criterion is described for testing whether an array of fissile units is subcritical, critical, or supercritical. A precise meaning is given to the term "degree of criticality." Formulas are derived for the multiplication of an array immersed in a nonabsorbing medium for both external and internal neutron sources. The criterion is shown to apply no matter in what detail the neutron emission is considered. (auth)


An initial application of the general method formulated in Part I to some unreflected air-spaced arrays of similar fissile spheres is described. (auth)


The Small Lattice Experiment (SLE) consists of a 0.5-ft test region containing the experimental lattice surrounded by a 2-ft diameter buffer region having similar nuclear properties. This in turn is enclosed in an 8-ft diameter driver region which is a graphite honeycomb loaded with sufficient graphite and/or U²³⁵ to achieve criticality. By suitable adjustment, the buffer produces an asymptotic spectrum throughout the test region, permitting standard cell measurements at comparatively high power levels. Furthermore, kₘ of the test lattice can be derived from poisoned test region-void reactivity comparisons. SLE and critical experiment measurements of ρ₂₈, δ₁₉, δ₁₉/δ₁₉ and kₘ in a common epithelial lattice (4% enriched U₂₃₅ rods, 70% D₂O in H₂O moderator) agree within 1%. A generalized theoretical method for deriving kₘ from experimental data is given. Spectral mismatch errors are analyzed to show that accurate kₘ measurements can be made with less than 1% of the test fuel required for an equivalent critical experiment. The application of SLE techniques to Th recycle SSCR design problems, particularly U²³³ effects, is discussed. The application to reactor design and the relative cost of direct kₘ measurements vs k_eff and B² from critical and/or exponential experiments are evaluated. A comprehensive experimental program developed to support the design of the Spectral Shift Control Reactor is discussed. (M.P.G.)


The material bucklings of twenty-five D₂O moderated lattices of natural U₂₃₅ rod clusters were measured in the Process Development Pile (PDP). The measurements were made in one-region loadings, and should therefore be subject to little systematic error. A number of the lattices employed voided housing tubes around the fuel assemblies. Values of migration areas inferred from measurements of positive periods are also presented. An evaluation of the errors in the buckling measurements indicated that the bucklings should be accurate to about 1%. The migration areas are compared with theoretical values obtained from the Benoist theory, and the agreement is shown to be good. (auth)


Operation of critical, exponential, and pressurized-expontential heavy-water moderated reactors is described. The applicability of these facilities in the following fields of reactor experimentation is discussed: buckling measurements in uniform lattices; anisotropic and void effects; evaluation of control systems; temperature coefficients; mixed lattices; and subcriticality studies of fuel for heavy-water moderated reactors. (M.C.C.)

Results of experiments on ZED-2 (enlarged ZEEP) are reported. Analyses of these results were made by calculating lattice parameters and by comparing bucklings obtained from these parameters with experimental bucklings. Other work is reported on reaction rate measurements in a hot ZEEP reactor channel, reactor fuel changes resulting from long-time irradiation, startup and operation of NPD-2, and conversion of NRX to natural UO₂ rods on power step up to 42 Mw. (J.R.D.)


A study was made of the reactivity status in the PRCF following a cell flooding event which would add H₂O to the core. A 55-inch D₂O moderator level was assumed, and a fuel loading of 17 Pu-Al Mark I type fuel elements on an 8-inch equilateral triangular lattice was calculated to have a critical moderator level at 55 inches. The effects of H₂O dilution of D₂O on the reactivity were calculated up to 50%, dilution (110-inch moderator level) and are shown graphically. A continuous reactivity decrease was noted with increasing H₂O addition. (D.L.C.)


The influence of heterogeneity on neutron spectra in nuclear reactors and assemblies was investigated. The heterogeneous reactor model, a principal classification of neutron spectra, slowing down spectra in a plate-type reactor, equilibrium spectra in a plate-type reactor, reactor spectra in the thermal range, the representation of reactor spectra by simple sets of functions, and the iterated multigroup method are discussed. Graphs are included. (M.C.G.)

203. (TID-7658(p.96-8)) BORAX-V COLD CRITICAL EXPERIMENTS. J. Häger (Argonne National Lab., Ill.).

Rod calibrations in BORAX-V were carried out as boric acid was added in discrete steps so as to achieve criticality with all rods in a bank at several different heights. The total available excess reactivity was found to be about 12% and the shutdown margin to be about 7%. These compare with calculated values of 13.5 and -14.4% respectively. A measurement was made of the temperature coefficient with boric acid present and with the aid of electric heaters. Over the range of 79 to 101°F, a value of -0.005% Δk/k was obtained. Steam voids were simulated by the use of ½-in. OD x 0.035-in. thick walled aluminum tubing with ends sealed and inserted into water channels. Reactivity effects of removing fuel rods and entire assemblies were assessed. (M.C.G.)

204. (HW-74761(p.51355)) V. PHYSICS. (General Electric Co., Hanford Atomic Products Operation, Richland, Wash.).

The use of lutetium as a spectral index detector was studied. Data are presented for correcting lutetium activity measurements from thick foils to that with thin foils. Subcritical experiments with 1.82 wt% Pu-Al rods in light water were conducted as part of the series of experiments which utilize plutonium containing various concentrations of the isotope Pu²⁴⁰. The experiments furnish critical mass data which allow more accurate nuclear safety specifications to be set for the reprocessing of plutonium fuels and which can be used as check points in the calculation of light water moderated lattices containing plutonium. The experiments were conducted using critical approach and exponential measurement techniques. A brief study was conducted to examine the applicability of the Westcott scheme to the analysis of some plutonium-aluminum-H₂O assemblies and to suggest some possible prescriptions for survey calculations of reactor types not readily described by means of the Westcott cross section routines. Many of the simpler reactor surveys employ point reactor calculations. Calculations of this type can be meaningful if they are normalized to experimental values or are calibrated by more refined models. In addition to the point reactor systems, one-dimensional calculations were made for the same series of experiments using a three-group diffusion-theory physics model. (auth)


Migration areas are calculated from a one-group critical equation, using experimental variations of buckling coefficients with lattice size. The assembly is a fully reflected, slightly enriched-UO₂-plutonium-fueled, light-water-moderated facility. Buckling values are tabulated for cylindrical, rectangular, and parallelepipedal core geometries. (T.F.H.)

206. (CVNA-133) PHYSICS PROPERTIES OF CLUSTERED UO₂-D₂O LATTICES AT VARIOUS LATTICE PITCHES. J. Jedruch, J. D. Cleary, R. D. Leamer, and...
Lattices 207-211


An experimental and an analytic investigation of the effects of changes in the lattice of a core consisting of slightly enriched UO₃ rod clusters moderated by D₂O is described. Quantities related to thermal utilization and resonance escape probability are obtained for different water-to-U volume ratios. Thermal spectrum and conversion ratios are investigated by integral experiments. Critical masses and bucklings are given for different lattice pitches as well as spatial and total void coefficients. An optimum computational model is selected which gives good agreement with experimental values of selected parameters. (auth)

207.


Experiments were conducted to determine the source neutron multiplication of EGR fuel assemblies under conditions which might exist in transportation and storage. Twenty-eight fuel assemblies were arranged to yield maximum nuclear reactivity under conditions which might be expected for out-of-reactor environments. It was found that these 28 assemblies cannot be made critical when water moderated and reflected. (auth)

208.


Material buckling values for a series of graphite-modified uranium carbide-fueled lattices were obtained by experimental using the AE-5 thermal column as the neutron source. Three lattice pitches with center-to-center fuel spacings of 9.02, 11.015, and 12.056 in. were examined. The unit cell of the lattices was hexagonally shaped with a central control channel surrounded by 120° segments of six fuel elements located at the corners of the hexagon. The fuel consisted of 0.5-in.-diam. pins of 3 wt.% enriched uranium carbide arranged in an 18-pin cluster to mockup the element proposed for a 500 Mwe power reactor design. The fuel process tubes also contained solid sodium to mockup core coolant. The central control channel can be changed to simulate either the wet or dry control concepts. The measurements were compared with theoretical values obtained from four-group theory with fast-flux weighting and thermal pin disadvantage factors. There is considerable disagreement between the two sets of results. Critical experiments performed for the 11-in. spacing only tend to bear out the validity of the exponential measurements. (auth)

209.


Critical and exponential measurements were made in a number of light water moderated lattices, fueled with highly enriched U. The results are compared with theoretical predictions based on the MUFT-SOCOCATE few-group model and shown to be in good agreement. Further work is reported on an experimental comparison between U-water and Pu-water lattices which may point to a breakdown in the methods of calculation or a discrepancy in the nuclear data in current use for Pu. (auth)

210.


Research on material buckling for lattices with natural uranium involved developing configurations to provide an incoming flux appropriate for macroscopic and microscopic measurements. Analytical and experimental methods giving accurate results were devised; computer codes for deriving buckling from measured flux traverses were arranged.

211.


An experimental and theoretical research program on exponential lattices of normal and partially enriched U rods moderated by heavy water is discussed. The special technique developed to supply neutrons from the horizontal thermal column of M.I.T. Research Reactor to the exponential tank, which has a vertical axis, is described. Several volume ratios can be investigated. Measurements of buckling, resonance capture, conversion ratio, and fast fission are made, as well as intracell thermal flux distributions with spectrum-sensitive detectors. The experiments have shown where improvements in calculations are necessary, e.g., in treatment of the cell boundary in the thermal flux space-energy distribution. Most measurements are made by foil activation. A number of techniques have been developed, including a coincidence counting method for the measurement of Np-239 activity in resonance capture experiments, La¹⁶ fission-product counting for fast fission experiments, and thermal flux plots using foils of Au, Lu, and Eu. Results, including bucklings, are
available for 2.5 cm normal U metal and 0.65 cm 1.027% enriched metal. Preliminary and future work is outlined. (M.P.G.)

212.


The Advanced Test Reactor Critical Experiment (ATRCE) is described and data are presented from chemical analyses of components. Core design, loading, and criticality conditions are given. Preliminary control cylinder sensitivities and the prompt neutron decay constant at the critical conditions were measured. The decay constant was used to determine the worth of a variety of fully inserted rods. (H.D.R.)

213.

31692 PULSED NEUTRON EXPERIMENTS ON SUBCRITICAL HEAVY WATER NATURAL URANIUM LATTICES. H. Meister (Kernforschungszentrum Karlsruhe, Ger.). J. Nucl. Energy, Pt. A & B, 17: 97-114(June 1965), (In English)

Prompt neutron decay in several subcritical D0-modified natural uranium lattices is investigated by the pulsed technique. Employing various detector positions inside the multiplying medium, a separation of flux harmonics is achieved by means of a Fourier transform.

From the modal decay curves the prompt neutron decay constant $\alpha$ is found as a function of geometrical buckling $B^2$. In the lower buckling region, $B^2 < 25 m^{-2}$, experimental $\alpha$ vs. $B^2$ curves agree with simple two-group calculations on the basis of lattice parameters determined in experimental experiments. Deviations for higher $B^2$ are shown to be produced by spectrum shifts towards lower neutron energies. In addition, the change of radial buckling $\Delta B$ caused by cadmium rods fully inserted into the lattice is determined.

The results are in agreement with stationary experiments as well as two-group calculations. With single fuel rods withdrawn from the lattice, however, the corresponding radial buckling change $\Delta B$ turns out to be strongly dependent on the buckling of the axial flux distribution. (auth)

214.


The research and development programs at Westinghouse Atomic Power Division have emphasized the utilization of machine methods of computation and the evaluation of such methods by means of interpretation of critical experiments. The most recent program of critical experiments was the Multi-region Lattice Studies carried out under the Fuel Cycle Development Program. This program involved the measurements of criticality, neutron flux distribution, and reaction rates (microscopic parameters) in lattices of light water moderated, low enriched, $\text{U}_2\text{O}_3$ fuel clad in stainless steel. Fuel rods of $\text{U}_2\text{O}_3$ enriched to 1.6%, 2.7%, and 3.7% were utilized in lattices of 2.5:1 and 4.5:1 water/$\text{U}_2\text{O}_3$ ratio. Currently, these experiments are being supplemented by measurements in the same lattices with boric acid dissolved in the moderator.

The present status of a number of experimental and analytical problems is summarized and areas where further effort is needed are noted. An attempt is made to point out inter-relationships between different measurements or calculations where significant points of agreement or contradiction can be useful in interpretation of results or in planning future work. (auth)

215.


A series of critical experiments was conducted with water-reflected, undermoderated zirconium hydride assemblies. The purpose of these experiments was to evaluate the consequences of water immersion of SNAP 2/10A-type reactor cores. Critical loadings were measured with several combinations of lucite rods in vacant lattice locations, beryllium inserts, internal water, neutron poison annuli at the core-reflector interface, and ammonium pentaborate in the water supply. The reactivity worth of the upper tank water as a function of height and the incremental worth of substitution of fuel rods for lucite rods was measured by the pulsed neutron method as the loading was continued past the critical point. From these measurements, an extrapolation for the excess reactivities of fully loaded, fully water-reflected assemblies was obtained. The ratio of the effective delayed neutron fraction to the effective prompt neutron generation time ($T^*$) was measured for several unpinned configurations, employing both pulsed neutron and noise analysis methods. The two methods were in satisfactory agreement, giving a best value of $4.76 \times 10^2 sec^{-1}$. The corresponding value for the beryllium-reflected reactor was measured as $1.38 \times 10^2 sec^{-1}$. The much greater $T^*$ for the water-reflected assembly is attributable to reflector delayed neutrons. (auth)

216.


217.

Lattices 218-222


A series of experimental measurements was made on the Experimental Gas Cooled Reactor (EGCR) lattice in the Physical Constants Test Reactor (PCTR). The measurements provide a broad basis for normalization of reactor calculations for lattices of this type. The fuel assembly is a cluster of seven uranium oxide rods, enriched in the, U²³⁵ isotope and clad with stainless steel. The fuel is spaced on an eight-inch square pitch in a graphite moderator. Values of the lattice parameters for the plots, f, p, and f were obtained for 1.8% enrichment of the uranium oxide fuel. The values of k and f were also obtained for 2.6% enrichment fuel. The techniques of using the PCTR were extended so that supercell measurements may be made. The values of the strength of a boron carbide control rod and a stainless steel loop tube were obtained in this way. The strength of such an inhomogeneous poison in the lattice is expressed as the difference in the supercell multiplication factor at with and without the poison in the supercell. This difference is the same quantity as is obtained in the usual reactor cell calculation. The fuel temperature coefficient of k for this cluster was also measured between 50 and 500°C. The coefficient obtained is temperature dependent. The more important of the lattice parameters for the 1.8% enriched fuel are k = 1.146 ± 0.004, f = 0.899 ± 0.005, p = 0.924 ± 0.006, ε = 1.019 ± 0.002, δk (control rod -16 cell supercell) = -0.157 ± 0.012, δk (empty loop tube -3 cell supercell) = -0.117 ± 0.011, and (1/k_∞)(dk_∞/dT) = -0.68 ± 0.05 x 10⁻³. For the 2.6% enriched fuel, results are k = 1.256 ± 0.009 and f = 0.845 ± 0.006. (auth)

218.


The multiplication factor for light-water lattices was determined with a zero-reactivity measurement. By mixing D₂O with the moderator water, k_∞ of the lattice was made equal to one, which was determined with the help of a test reactor. The reactivity change of the reactor had to be zero if the exactly poisoned insertion was exchanged for air in a place in the reactor in which the fuel gradients of all examined neutron groups were zero. This condition was produced by surrounding the measurement insertion in the reactor with an adapter. For the apparatus used a measurement accuracy of ±1% for k_∞ could be calculated. (tr-auth)

219.

28452 (TID-7650[67-87]) COMPARISON OF CRITICAL EXPERIMENTS AND THEORY FOR SOME THORIUM-WATER LATTICES. D. H. Roy (Babcock and Wilcox Co., Atomic Energy Div., Lynchburg, Va.).

An extensive series of critical and exponential experiments was performed with uniform lattices of fuel rods composed of fully enriched UO₂ in ThO₂. These lattices were moderated by either H₂O or mixtures of H₂O and D₂O ranging up to 82 mole % heavy water. Lattice spacings ranged from those yielding a somewhat over-moderated core to those yielding an extremely tight lattice in which more than half of the fissions were epithermal. The experimental measurements included buckling, ratio of epithermal to thermal captures in Th₂O₃, ratio of epithermal to thermal fissions in U²³⁵, the thermal disadvantage factor, and water height reactivity worths leading to δk_∞/δh. (auth)

220.


Calculations were made of the criticality of Pu(NO₃)₃ solutions in two sub-critical arrays. An extrapolation of the inverse multiplication curve for a shielded array of S-in.-dia. tanks of stainless steel indicated that the 5 tanks could be of infinite length and remain subcritical. Results for the 30-in.-dia. Rasching-ring-filled stainless steel tank indicated that the vessel would not be critical at any height. (M.C.C.)

221.


Empirical equations were found that relate the critical heights of cylinders in arrays with the array base dimensions for various surface-to-surface separations of the individual cylinders in the array. The methods were applied to critical arrays of cylindrical vessels containing aqueous solutions of UO₂(NO₃)₂. These critical data were generated by the critical mass facility of the Oak Ridge National Laboratory. (auth)

222.


Gathering and disseminating information on D₂O lattices, integral parameters required to specify heavy water lattice performance, status of the method used for D₂O lattice calculations, data needed for the operating conditions of reactors, use of the substitution method to study lattice buckling with interpretation of the results by the one-group perturbation or the two-group two-region method, experiments to get information on burn-up in D₂O lattices, and determination of neutron spectra and cross sections relevant to burn-up studies are examined. (D.C.W.)

Results are given of exponential buckling measurements on graphite lattices containing clusters of U metal rods, clusters of uranium oxide rods, and single U metal rods with a range of near-natural U content. In the case of the metal fuel elements the results of the measurements are correlated with theory. (auth)

LATTICES 223-230

Results are given of neutron flux fine structure measurements in graphite lattices containing wide ranges of U metal and uranium oxide fuel element numbers. In the case of the metal fuel elements the observed fine structure parameters are correlated with theory. Measurements of neutron diffusion area in graphite, and of buckling and flux distribution in assemblies containing super lattices of empty channels are included. A correlation with theory of migration area asymmetry measurements in U-graphite lattices is also given. (auth)
Lattices 231-236


Natural U lattices in heavy water were investigated with the pulsed-neutron-source method. The systems were very small; buckling ranged from 300 to 900


The reactivities of a number of cores containing BeO and polyethylene, which simulated water flooding, were measured and calculated. Other determinations included core symmetric axes, reactivity worths of rods and rod voids, and perturbations on assembly reactivities. The experimental arrangements and measurements, the calculation methods used in the analysis, and the results of the analytical-experimental comparison are described. (D.C.W.)


Reactor physics measurements on UO2 lattices moderated by mixtures of light and heavy water have been performed in the NORA zero power reactor. The fuel used for these experiments has been UO2 enriched to 3% U235 clad in stainless steel. A series of measurements on different H2O/D2O ratios and different moderator-to-fuel volume ratios has been completed. Macroscopic and microscopic neutron flux distributions have been measured using different types of fuel materials resulting in sets of $\theta_i$ values and cell parameters for thermal and resonance neutrons. The ratio of epi-cadmium to sub-cadmium capture in $\text{U}^{235}$ ($\rho^{25}$), the ratio of epi-cadmium to sub-cadmium fission in $\text{U}^{235}$ ($\delta^{25}$), and initial conversion ratios have been measured. The ratio of Lu-foil activity to Cu-foil activity has been measured in different positions inside the fuel and in the moderator and was compared to calculated values of the same activation ratio and used as a spectral index for the lattices studied. Supplementary to experiments on clean critical cores some measurements on two-region cores have been performed to make it possible to extend the range of measured buckling values considerably. Analysis of the experimental results using multigroup methods both for the fast and thermal neutron region is presented. The neutron energy spectrum in the fast region is calculated by a code BIGG. Resonance absorption and fission are given a correct distribution in energy and are governed by input resonance integrals. BIGG is well adapted to handle slowing down in mixtures of light and heavy water. The thermal part of the neutron spectrum is calculated by the K-7 THERMOS code. (D. F.)
237. 238. 241. 240. 239.


**Lattices 237-243**

reactor for a series of uniform, square, D2O-moderated lattices with moderator-to-fuel ratios ranging from 99:1 to 12:1 for cylindrical fuel rods of the following specifications: Fuel: 3% enriched, unsintered UO2, density = 9.41 g/cm³, diam. = 1.128 cm. Cladding: 304 stainless steel, integral diam. = 1.128 cm, outer diam. = 1.270 cm, thickness = 0.071 cm. The buckling values obtained are compared with different theoretical estimates. (auth)

**242.**


A series of clean critical experiments was performed in the SGR critical facility utilizing 2 and % enriched, uranium metal, hollow cylinder, fuel elements in ACGT graphite moderator. Six lattice spacings were used, ranging from 6.93 to 16.9 in. on a triangular pitch. Critical loadings and fuel element worths were determined and compared to the results of 4-group diffusion theory. Calculations utilized TEMPEST, S4, FORM, and AIM-5 programs on the IBM 7090. The calculated Kine compared well with experiments over the full range of moderator-to-fuel volume ratios when using a 2200 m/sec graphite absorption cross section of 4.07 mb. The sensitivity of the calculation to variations in the graphite absorption cross section was examined and the experimental error due to inventory uncertainties was assessed. The differential worths of both the central and peripheral fuel elements were obtained and agreed in general with AIM-5 calculations. The thermal flux traverse of a unit cell was shown to agree best with a Wilkins' spectrum option of TEMPEST. Details of both the experimental and theoretical methods are given. (auth)

**243.**

**4688** (BNL-799(p.1-25)) REACTOR PHYSICS DIVISION. J. Chernick and H. Kouts (Brookhaven National Lab., Upton, N. Y.). Theoretical research was mainly in the fields of neutron normalization and reactor dynamics. The analysis of the approximated characteristics of the uranium plate lattices was completed with excellent agreement between buckling measurements and the theory. Alternate cadmium ratio techniques of determining resonance capture in lattices were compared. A new uranium-graphite assembly was brought critical in the Minimum Reflection Critical Facility. Kinetics of the High Flux Beam Reactor were investigated. Measurements in uranium slabs in water were completed. For the Settled Bed Reactor program, uranium carbide fuel was analyzed. The characteristics of the settled bed are studied. Work is continuing on the magneto-hydrodynamic systems as alternatives to the SNAP turbo-generator for generating electricity. Pulsed fast neutron plasma reactor studies have indicated that UBr2 or UBr5 would be much more suitable gaseous fuels than UF6 be-
cause of their lower disassociation temperatures. The characteristics of a thermally regenerated cell utilizing the Pb–PbBr₂–Br₂ system are being investigated. A study of the characteristics of a boiling oxygen chemonuclear reactor for ozone production was carried out. The feasibility of gaseous-moderated reactors was continued and a number of cases using HF, CF₄, DF, and HF–CF₄ moderators with UF₆-fueled cores were investigated. Studies of the stability of selected condensed aromatic compounds to heat, gamma radiation, and reactor radiation are continuing. Fouling tests are evaluated for organic coolants. Neutron and reactor core section studies are reported.

(7.8.3.)


Calculations of the critical sizes of cubic arrays of interacting fissionable units are compared with critical experiments. The units were of two types: vessels containing 5 liters of an aqueous solution of highly enriched hydrogenous reflectors. Agreement between calculation and experiment is reasonably good when consideration is given to the simplicity of the method of calculation. Curves are presented for computing critical sizes of cubic arrays of 8, 27, 64, or 125 identical units as a function of the albedo of the reflector surrounding the array and of the reactivity of an individual unit. (auth)


A systematic study of natural uranium-graphite lattices was undertaken in the critical facility MAURUS, which was built in 1959 in Marcoule. Integral measurement of lattice properties is carried out by the progressive replacement method. The report describes the experimental methods, the analysis of the experiments, and the results obtained for lattices with pitchings ranging from 192 to 317 mm and fuel elements with cross sections ranging from 6 to 20 cm². The principles of corelation of the results are also outlined. Additional experimental results are also given, pertaining to the determination of the anisotropy, of both the axial and the radial migration areas, and of the age in graphite. (auth)


A review is presented, dealing with heavy-water-moderated assemblies, of measurements of buckling and reactivity; calculations of reactivity; detailed measurements and calculations of lattice parameters; determinations of the effects of voids, heterogeneities, and other special features on the physics parameters; and the incorporation of the measured and calculated physics parameters into the design of heavy water reactors. Examples are given. A bibliography of 103 references is included. (T.F.H.)
parallelipipeds, made of stainless steel containers with a wall thickness of 1.5 to 2 mm and radii or sides of 30 cm, filled with an aqueous solution of 90% enriched $^{235}\text{UO}_2\text{(NO}_3\text{)}_2$. Similar systems were also built with 6-liter capacity glass cylinders with 0.3-cm thick walls. Comparison of the experimental and theoretical values indicated that the latter contain a suitable safety margin and therefore may be used for evaluating the safety of an interacting system; this was proved by recalculating the data of E. Woodcock and U. Paxton (Progr. Nucl. Energy Series Vol. 4, 213 (1961)) concerning a system of $U$ spheres. (TTT)

250.


One-region buckling measurements that were made on a series of $D_2O$-moderated lattices of heavy uranium metal tubes in the Process Development Pile at Savannah River Laboratory are presented. The purposes of these measurements are to provide normalization points for lattice bucklings and to extend the study of natural uranium-$D_2O$ systems. The dependence of buckling on the moderator-to-fuel ratio is studied for two types of lattices. (R.E.U.)


In connection with the development of a heterogeneous zirconium-hydride moderated reactor, several critical and sub-critical measurements on lattices have been performed. These tests were performed in a fixed aluminum lattice by varying the arrangement of rods containing $UO_2$ or zirconium-hydride with equivalent diameters. The time-dependent neutron flux behavior after a neutron pulse was measured. The higher harmonic effects were eliminated through a numerical evaluation method which permitted the determination of the characteristic decay constants, as eigenvalues can also be directly calculated for a given system through theoretical methods, developed and employed for this purpose. The reliability of these methods can be illustrated with an example of a $UO_2$-$H_2O$ experimental core with a large reflector. Measurements were performed on various zirconium-hydride lattices up to $450^\circ C$, in order to determine the temperature dependence of the reactivity. This dependence is of special interest in the case of zirconium-hydride moderation. In this connection several theoretical expressions for the scattering kernel and for the heterogeneous spectrum methods were investigated in order to theoretically describe the experimental results as well as possible. The critical experiments were performed on a split-table assembly of the Argonne "ZPR" type and were limited a maximum temperature of $200^\circ C$. The results of the sub-critical measurements were in essence verified by the critical assembly. In addition measurements with a pile-oscillator, threshold detectors and resonance detectors were made to obtain a very detailed determination of the cell-parameters. These measurements will, in turn, make possible the determination of the temperature dependence of heterogeneous zirconium-hydride lattices. (auth)


Experimental results and theoretical interpretation of a series of twenty uniform lattice critical experiments in which the neutron spectrum is varied over a fairly broad range are summarized. Two types of fuel rods were studied: 4.02% enriched $UO_2$ in stainless steel tubes; and 2.46% enriched $UO_2$ in aluminum tubes. Lattice nonmoderator-to-moderator volume ratios ranged from 0.65 to 1.2. The moderators were mixtures of light and heavy water ranging in composition from zero to 77% $D_2O$, with and without boric acid. Measurements include critical size and composition, $\rho_0/\rho_h$, buckling and reflector savings, thermal disadvantage factor, and cadmium ratios of $U^{234}$ and $U^{238}$. Theoretical methods used to analyze the data are given, and results are compared. (auth)


Critical experiments and theoretical calculations of reactivity and flux perturbations were performed. Three major critical assemblies were studied; the basic cores had 1.206-cm diameter, aluminum clad, 2.46% enriched $UO_2$ fuel rods on a square lattice pitch of 1.511 cm. Mod-
Lattices 255-258

erator compositions for the three major cores had 0, 50, and 72 mole % D₂O. Boric acid was added to keep the core radius constant at 61.11 cm. Data on a zone-loaded core with an inner zone of 93%-enriched aluminum clad UO₂—ThO₂ fuel rods surrounded by an outer zone of 4%-enriched stainless steel clad UO₂ fuel rods are also included. The moderator composition was 81% D₂O. Perturbers were introduced axially by removing fuel pins to create moderator gaps and by inserting perturbing blades and cruciform rods. The blades were 10 inches wide. The cruciform rods were 6 inches from tip to tip. The materials were Boral, cadmium, or aluminum. Most perturbers were fully inserted, but some measurements were made with the Boral cruciform rod withdrawn. Thermal flux distributions were measured with gold- and dysprosium foils. Reactivity was determined by using 172 mole % Th in lattices 255-258. The materials 


The Small Lattice Experiment (SLE) is a technique for measuring kᵣ and other infinite-medium lattice parameters using much less test fuel than is required for equivalent critical or experimental experiments. The theoretical basis for applying such an experiment to epithermal lattices is discussed, an analysis of spectral mismatch error is given, and the facility for the experiment is described. SLE measurements are compared to those in an equivalent critical assembly to test the validity of the technique. In an epithermal lattice of 4.6%-enriched UO₂ fuel rods having a nonmoderator-to-moderator volume ratio of 1.0 and moderated by a D₂O—H₂O mixture containing 70% D₂O, values of kᵣ, ρ₀/∅₀, ρ₂/∅₂ agree within statistical errors (0.5 to 1%). The SLE technique was also used to measure these parameters in two similar epithermal lattices containing (U₂O₃—ThO₂) fuel rods (N₁/N₂ = 15). (auth)


Lattice parameters are studied in natural-uranium-fueled heavy-water-moderated pressure-tube type reactors. The ZED-2, ZEEP, and CANDU Reactors are used in this study and their results are obtained by experimental techniques and analyised theoretically by computer codes LATREP and LATTICE ANALYSIS. The analytical methods are outlined and correlated with the experiments. (N.W.R.)

3186  (KR-50) COOLANT VOID REACTIVITY MEASUREMENTS IN A D₂O-MODERATED LATTICE WITH 7-ROD UO₂ FUEL CLUSTERS. F. W. A. Haberman (Nuclear facilities in the zero power facility NORA with a core of 7-rod UO₂ cluster fuel elements of the Halden Boiling Heavy Water Reactor to investigate the reactivity effect of void formation in a coolant channel. The shroud of one fuel element was subdivided into sections in which the water level could be depressed by air. The reactivity effect of voids created in this way was measured for various combinations of channels. The measured effect was always negative and largest in absolute magnitude in the center of the fuel cluster. The interaction between voided channels was such as to increase the absolute magnitude of the reactivity effects. With a first order perturbation expression based on two-group theory for an anisotropic reactor, curves were fitted to the measured reactivity effects as a function of void depth. For each case two parameters were calculated representing, respectively, multiplication and leakage properties of the lattice. It is then possible to make a comparison with lattice parameter calculations. The comparison is made only for the case of 100% void in a coolant channel by calculating the parameters for this case with the Swedish burnup code. (auth)


Diffusion coefficients and bucklings are measured for UO₂ fuel tube assemblies in D₂O moderated lattices. The effects of H₂O fogs and air in the coolant channels in place of D₂O are studied. (T.F.H.)

LATTICES 259-265


The experimental determinations of critical size and flux distributions in critical and sub-critical assemblies using fuel packs consisting of Al-U and stainless steel plates are described. The results show that few-group models are adequate for calculating the critical size of highly absorbing light water moderated lattices and that material buckling and thermal finite structure measurements can be made in light water moderated assemblies as small as 12 in. square and 24 in. high. (D.C.W.)

263.


An experimental and theoretical program on the physics of heavy water-moderated, partially enriched lattices is reported. Experimental methods were adapted or developed for research on buckling, fast fission, resonance capture, and thermal capture. After being successfully tested on lattices of one-inch-diameter natural-uranium rods in heavy water, the methods were applied to three lattices of 1/4-inch, 1.925% enriched uranium rods, moderated by heavy water. Research programs are also under way to take and correlate data from single-rod measurements, two-region lattice measurements, miniature lattice measurements, and pulsed neutron methods. In addition, a program is under way to measure the effect of control rods in the lattice assembly. A listing of topical reports, generated during the report period, is included. (auth)

264.


It was determined that it is safe to store the SRE Core III elements (both enrichments) in an infinite array when positioned at the spacing dictated by the presently designed storage cell. This safety is independent of the degree of water flooding. However, it is dependent on the presence of the four-inch schedule 40 steel pipe positioned in each storage hole. Although the array is calculated to be subcritical (k_infinity < 1) in the absence of the steel pipe, the accuracy of the method of calculation and the margin of safety are inadequate for assuming it to be safe under these conditions. (auth)

265.


Experimental results on material buckling and extrapolation distances for various lattices and fuel arrangements

259.


The methods commonly used in making buckling, critical loading, and cell parameter measurements are reviewed, and the results of such measurements are summarized. Only simple uniform and multiregion lattice experiments are considered. The experimental work done at each of the laboratories involved is summarized. The methods of interpreting the various lattice measurements are briefly discussed, with emphasis on the sources of systematic errors. (D.C.W.)

260.


A compilation of measurements for light-water lattices is presented. Measurement techniques, interpretation of results, and actual experimental results are described. Bucklings and critical dimensions are tabulated in the appendix. A bibliography of 109 references is presented. (R.E.U.)

261.


Critical experiments are reported for clumped heterogeneous light water moderated lattices using 3 wt % enriched UO2 fuel clad with stainless steel. Reactivity worth utilizing water height and pulsed neutron techniques, temperature and void coefficients, flux and power distributions, and spectral indicator activations were measured for a wide range of configurations. These configurations were typical of boiling water reactor lattices and included assemblies with black and gray absorbing slabs and control rod cruciforms. Differential reactivity and spectrum effects were investigated for various slab, cruciform and void tube perturbations in these experiments. The resonance spectral indicator method was used in particular to study the spatial variation of the neutron spectrum and density in a typical boiling water reactor lattice. The observed spatial neutron spectrum variations show that this is a very useful technique. (auth)

262.

3104 CRITICAL SIZE AND FLUX DISTRIBUTION MEASUREMENTS IN HIGHLY ENRICHED, LIGHT WATER MODERATED LATTICES. I. Johnstone, W. H. Taylor, and S. K. Wallace (Atomic Energy Establishment, Winfrith,
Lattices 266-269

are reported. The results are compared with theoretical values calculated by simple but realistic models. By use of the experimental results on material buckling, the number of neutrons per absorption in fuel and the effective resonance integral is reduced. (auth)


The ratios of epicadmium to subcadmium capture in U238, of U235 capture to U238 fission, of U235 fission to U235 fission, and of epicadmium to subcadmium fission in U238 were measured in a UO2 blanket in a UO2-Zr-H2O seed blanket-seed slab core with a water to nonwater volume ratio of 0.142. The results were compared with calculations by a design model using a one-dimensional few group diffusion program and with calculations by a one-dimensional, multigroup P-3 transport code. (D.C.W.)


A series of lattices consisting of voided rod cluster elements was investigated in the subcritical facility MINOR. This experimental program has formed the basis to check calculation methods for heavy water-moderated and gas-cooled pressure-tube reactors. A detailed analysis was undertaken for an element consisting of 7 uranium metal rods enriched to 0.960%. This rod cluster is supported by a graphite structure, which is surrounded by a pressure tube. An annular air gap separates the pressure from the calandria tube. This cluster presents a physical mockup of the fuel element to be used in the first charge of the Swiss experimental power station presently under construction at Lucens. In order to obtain calculation methods that are of a more general usefulness and have a wider field of application, the experiments specific to the Lucens reactor were extended by varying a number of parameters of the lattice arrangements; the most important change consisted in a replacement of the enriched rods by natural uranium rods, leaving all structural material the same. The void content per cell was decreased by removing the calandria tube and increased by removing the graphite fillers. Some measurements with 4-rod clusters were performed. For all lattices, several moderator-to-fuel volume ratios were investigated. For some or all of the lattices the following parameters were determined: the material buckling; flux disadvantage factors; the epithermal component; Cd-ratio of 198J and axial flux peaking due to Al spacer pieces. Theoretical interpretations of the experiments have been made on the basis of the two group four factor model. (auth)


The kinetic behavior of a neutron transport medium irradiated by a burst of fast neutrons was investigated on the basis of several theoretical models. Expressions were derived for the prompt-neutron decay constant of the asymptotic thermal flux in a subcritical multiplying system. These expressions relate the decay constant of a critical assembly to various parameters of interest. Pulsed neutron experiments were made with subcritical assemblies to measure lattice parameters. The pulsed neutron method was also applied to the measurement of absolute reactivity and the reactivity worth of control rods in far subcritical assemblies. Concurrently with the pulsed neutron studies, steady-state exponential experiments with control rods were also undertaken. Die-away experiments on pure moderator assemblies were made to measure the thermal-neutron diffusion parameters of heavy water at room temperature and the effect of thermally black rods inserted axially in a cylindrical moderator assembly. Pulsed neutron runs on unperturbed lattices were used to evaluate such lattice parameters as k, L, k, \( k_{th} \), f, \( k_{th} \), etc. These values are in agreement, within experimental uncertainties, with the results of steady-state exponential experiments and by calculations based on the THERMOS code. Pulsed neutron experiments on perturbed lattices were made to find prompt-neutron lifetime and the absolute negative reactivity of the assembly. The worths of control rods were also measured. The pulsed neutron and steady-state experiments for the measurement of the reactivity effect of control rods give results that agree within the experimental uncertainties. Two-group theory, with no allowance for absorption in the fast group, is found to underestimate the worth of the rod by a few per cent. The conditions for the validity of control rod experimental studies of exponential assemblies were considered. Suggestions for extending the techniques developed in this work and for refining the results are also included. (auth)


A report is given of a series of exponential experiments performed on the Queen Mary College subcritical assembly. This assembly consists of a 4\( \frac{1}{2} \) ft aluminum core tank in which up to 4 tons of aluminum-clad natural uranium in the
form of 1.2-in. dia. bars, 43.25-in. long, are supported vertically by a series of lattice plates. Initially a 10-c., Po–Be neutron source (2.5 × 10^5 n/s) was used to provide the neutron flux, but later measurements were made using a SAMES, 150 kv neutron generator (5 × 10^5 n/s). Buckling measurements were made for moderator-fuel ratios of 1.73, 2.48, and 3.18 using Indium foils, BF₃ counters and BH₃ fission chambers as detectors. Migration areas were determined for the same ratios of moderator-to-fuel by means of the boron poisoning technique. The novel features of the results were the apparent variation of the material buckling with fuel loading and with the type of detector employed. The reasons for these variations are discussed and their general significance to exponential experiments in hydrogenous media developed theoretically. (auth)

270. 5970 (ORNL-3499(Vol.1)(p.62-3)) CRITICAL EXPERIMENTS AND CALCULATIONS WITH ANNULAR CYLINDERS OF U(93.2) METAL. J. T. Mihalcezo (Oak Ridge National Lab., Tenn.). Critical experiments were performed on annular cylinders of enriched (93.2% U²³⁵) metal to verify the adequacy of the S₄ method of solving the transport equation for this geometry. The measured reactivities agree well with the calculated multiplication constants. Measurements of prompt-neutron decay constants by the pulsed-neutron technique were included. (auth)

The core-tank of UTR-B is separated in six sections. Each section has a length of 83.3 mm in east-west direction, and holds a fuel element that has a length of 76.2 mm in the same east-west direction. It may be expected that the reactivity of the reactor would be changed, if each fuel element is shifted towards the center line of the core-tank or in the opposite direction. The possible reactivity change due to the slight shift of the fuel elements was examined. It was found that if each fuel element is shifted toward the center line of the core-tank, the reactivity of the reactor increases and the critical mass decreases. If each fuel element is shifted outward in the opposite direction, the reactivity of the reactor decreases and the critical mass increases. This effect is smallest near the center line of the core-tank, and the farther the position is from the center line, the larger the effect becomes. But near the side of the core-tank, the effect comes small again. This may be affected by the thermal neutron flux distribution and the importance function. The thermal neutron flux distribution in the core-tank was measured, and the importance function was calculated. The curve of the statistical weight vs position of the core-tank was found to have the same tendency as the experimental results. (auth)
Lattices 275-280

275.

The parameter $g^{(i)}(l = 1/s)$ was experimentally determined on the light-water moderated, enriched-uranium fueled, heterogeneous critical assembly ZR-1 using both the Feynman and the Rossi methods. It was found that the Rossi method can be used in the case of thermal reactors with hydrogenous moderator and enriched fuel provided the average neutron lifetime is not larger than $10^{-4}$ sec. The latter of the two methods proved to be faster and more reliable and even the effect of the delayed neutrons does not disturb the evaluation of the data measured. (auth)

276.

The material buckling of various lattices in the exponential (ZEBRA) and the critical (RO) facilities, resonance integrals of various fuel geometries and materials, and neutron spectra with a fast chopper are discussed. In the exponential experiments, a leakage effect was found. The leakage results in a background that introduces systematic errors when changes in the radial buckling are estimated purely by theoretical calculations. In the critical facility the progressive substitution technique was extensively used. A modified one-group perturbation theory combined with a new cell definition is found to be a powerful tool for the interpretation of substitution experiments. A summary of the work on resonance integrals is given together with a discussion of factors limiting the experimental accuracy. The neutron spectrum in the central channel of the reactor R1, where the resonance integrals were studied, was subject to detailed investigations with a fast chopper. The energy range extended from 0.02 ev up to about 10 kev. (auth)

277.

A critical experiment was performed with 12 BORAX-V superheater subassemblies in a central voidable region plus 1228 to 1555 UO₂ fuel pins (3 wt % enriched) in a peripheral region. Removing water (28% of superheater volume) at room-temperature decreased reactivity by 2.2%. The midplane (two-dimensional) peak-to-average power distribution in the voided superheater was approximately 1.24, mostly attributable to flux depressions within insulated fuel boxes. Cadmium ratios are also reported. The experiment was initiated to supplement computational information which might have affected plans for loading the superheater zone into the BORAX-V reactor. No changes were indicated by the experiment. (auth)
Lattices 281-285


A series of experiments performed to determine the basic parameters of a D_2O-2% enriched-uranium lattice are described. The fuel elements were hollow cylinders, canned in aluminum and wetted by heavy water both on the inside and outside. The experiments were performed on the RB critical assembly of the Borla Kidrič Institute of Nuclear Sciences in Belgrade. The clean geometry of this system enables a rather simple theoretical interpretation of experimental results, and straightforward comparison with two-group diffusion-theory calculations. Measurements performed for ten different lattice configurations included determination of buckling, water-level reactivity coefficient, and disadvantage factors for thermal- and epithermal-neutron flux inside a reactor cell. Techniques for these measurements are described. The experimental data are then used to derive the lattice parameters defined by the four-factor formula and two-group diffusion-theory treatment of the reactor core. The results are compared with the standard calculations of the same parameters for measured lattice configurations. Deviations are found in the value of n and resonance integral for 235U. An analysis is made of the usefulness and accuracy of information obtained from critical experiments for design and exploitation of a research reactor, composed of D_2O and enriched uranium. (auth)


Experimental experiments with light-water moderator were conducted to determine criticality standards for the handling of uranium metal enriched to 3 wt % in 235U. These measurements, made with massive rods 2 and 3 in. in diameter, were combined with Hanford measurements with smaller rods to provide critical bucklings and masses for H_2O-modified lattices over a range of rod diameters from less than 0.15 to more than 3 in. Subcritical buckling measurements are compared with the more conventional approach-to-critical method. (auth)

283. 36779 (HW-81659(p.4-9)) EXPERIMENTS WITH PuO_2-UO_2 FUEL ELEMENTS IN LIGHT WATER. L. C. Schmid, W. P. Stinson, R. C. Liikala, and J. R. Worden (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Approach-to-critical, exponential, and critical experiments are carried out for H_2O-modified UO_2-PuO_2-fueled lattices having moderator/fuel volume ratios of 2.71 to 5.14. The center-to-center spacing of the fuel elements varies from 0.71 to 0.90 in. Experimental results are compared with the predictions of one-dimensional, 4-group diffusion theory. (T. F. H.)


Material bucklings for UC fuel elements, enriched to 3.02% are inferred from critical experiments in which 24 UC fuel elements were surrounded by uranium metal fuel elements. The UC fuel elements were placed on triangular pitches of 9, 11, and 12 in. and the number of uranium metal drivers changed to obtain criticality for each case. Four-group theory was employed to infer the material bucklings of each critical configuration by modifying the thermal absorption cross section of the UC region until the calculated value of k_eff was unity. Comparisons of theoretical calculations and "measured" material bucklings (B_{eff}) indicate that multigroup theory (15 groups) does much better than few-group theory (2 to 4 groups). Because of the inherent simplicity of few-group theory (as compared to multigroup theory), an extensive comparison was made between various few-group theoretical calculations. These calculations determined the effect on the curve of material buckling versus lattice pitch when all flux shapes were calculated in each of the four energy groups (with diffusion theory) and with transport theory. Also, various treatments were employed to obtain cross sections for the thermal energy group. These included calculations with and without flux depressions across each fuel pin, with the fuel pin homogenized in various ways (over the process tube volume, over the rubber band surface of the pin cluster, and into several bands of UC), and a thermal multigroup calculation (using THERMOS). These few group calculations caused significant variations in the magnitude of the material buckling but did not produce the same shape for the curve of B_{eff} versus pitch as was obtained from the measurements. Measurements on subcritical assemblies of B_{eff} for ASCR lattices were found to vary considerably from the values of B_{eff} that were inferred from the critical assemblies. When the subcritical material bucklings are corrected by using calculated values of diffusion anisotropy, most of the difference between the critical and subcritical values of B_{eff} disappears. (auth)

115
Material bucklings for uranium carbide fuel elements, enriched to 3.02 wt% of uranium-235, were obtained from 12 "two-region" critical mass measurements. Material bucklings were calculated by the GAM-1 multigroup program for macro- and micro-volumes. Reasonable agreement was found for material buckling as well as for the measured thermal disadvantage factors, initial conversion ratio, and fast fission ratio. (auth)

Light-water-moderated 10% enriched uranium-fueled lattices were investigated with different H/235U ratios. One of the aims of these experiments was to test the results of calculation methods used. For this purpose the measurements were carried out in a wide interval of the H/235U ratio in order to get significant effects. Fuel elements of the VVR-S-type reactor used were investigated with the two-group model and in most cases triangular lattice patterns applied. Critical mass values were determined over the range 190 to 775 of the H/235U ratio. This range involves nearly equal lengths of under- and over-modерated regions. Temperature dependence of reactivity was determined in the interval 10 to 40°C, in under- and over-modерated, and in a nearly optimum lattice. Temperatures of fuel and moderator were kept nearly the same. In this temperature interval both positive and negative values of temperature coefficients were found in all the lattices. Spatial dependence of the reactivity effect of vertical, cylindrical voids—the lengths of which were identical with that of the fuel elements—was determined in four cores with different lattice pitches. The values of the void-to-cell-volume ratios were kept nearly the same. Positive void effect was reached in a strongly under-modérate lattice. Thermal neutron distributions were determined in lattices containing central water traps of different diameters. Measurements were carried out both in a strongly under-modérate and in a nearly optimum lattice. The value of \( \beta_{HH} \) was measured using both the Fenymann and the Rossi method in a nearly optimum lattice. The two methods yielded nearly the same results. The value of \( \beta_{HH} \) was determined in an over- and an under-modérate lattice by the Rossi method only, which proved to be faster and more accurate. The results of a simple calculation method were compared with the experimental data. Rather good agreement was found for the two-group model when taking into account fusion under slowing down and the intracell dependence of the thermal spectrum in the calculation of the thermal utilization factor. (auth)

For the calculation of D2O-moderated, organic-cooled natural uranium lattices, a calculational procedure is proposed based on the most refined reactor codes available at Ispra. For the calculation of fast group constants, the GAM-1 multigroup program is used. Weighting factors to take into account the heterogeneity of the lattice cell are calculated by the DSN program for macro- and micro-cells. For the calculation of thermal cross sections the THERMOS code is used, and the disadvantage factors for the different materials in the cell are obtained by a one-group calculation using the DSN program. On the basis of this procedure, an analysis of critical experiments for 7-UO₂ rods, organic-cooled fuel elements was performed. (auth)

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tor and reflector. Three different lattice arrangements were used for the measurements which included the determination of the reactivity change caused by Cd, B, Dy and Ag specimens introduced into the assembly as aqueous solutions of Cd(NO₃)₂, H₂BO₃, Dy(NO₃)₂ and AgNO₃ in Plexiglas containers. The reactivity change due to the absorber was calculated from the measured change in the asymptotic period of the system brought to the supercritical state; the concentration of the solution was so chosen that reactivity change values of about 4 x 10⁻⁴ were obtained. The results differed from those of similar experiments carried out with the VVR-S type reactor having a similar spectrum and burnout; this is probably due to differences in the experimental technique. (TTT)


As many as 125 five-liter units of concentrated aqueous uranyl nitrate solution were assembled in critical arrays. The solution, at a concentration of 415 g of uranium per liter, and having a specific gravity of 1.555, was contained in right circular cylinders of methacrylate plastic having a 0.64-cm-thick wall. The U²₃⁵ content of the uranium was 91.2 wt%. The dependence of the number of units required for criticality as a function of the spacing was determined. The critical number, N, as a function of spacing, within the range of these experiments, has been determined to be

\[
N = N_0 \left( \frac{\rho}{\rho_0} \right)^{-s},
\]

where \(N\) is the number of units in the critical array, \(\rho\) and \(\rho_0\) are, respectively, the uranium density in the array and in the unit, and \(N_0\) and \(s\) are constants depending upon the neutron reflector surrounding the array.

For arrays with no reflector \(N_0\) and \(s\) are 2,506 ± 0.014 and 1.928 ± 0.028. Experiments with 8 and 27 unit arrays at lower U²₃⁵ concentration indicate that the value of \(s\) increases and that of \(N_0\) decreases with decreasing concentration. (auth)

292. 5968 (ORNAL-3499(Vol. II) p. 50-71) EFFECTS OF BOROSILICATE GLASS RASCHIG RINGS ON THE CRITICALITY OF AQUEOUS URANYL NITRATE SOLUTIONS. J. T. Thomas, J. K. Fox, and E. B. Johnson (Oak Ridge National Lab., Tenn.).

The natural-boron content of the glass varied from 0.5 to 5.7 wt %, and the volume of the vessel occupied by the glass ranged from 20.9 to 30%. Results from exponential experiments, using a critical layer of solution above the column of solution-ring mixture as a neutron source, provided estimates of the material buckling of the mixture as a function of solution concentration, boron content of the glass, and the glass volume present. It was shown, for example, that the buckling is negative if glass containing 4 wt % boron occupies more than 22% of the mixture volume, whereas the same concentration of glass containing only 0.5 wt % of boron results in positive values of the buckling except for solutions more dilute than about 72 g of uranium per liter. (auth)

293. 5947 (ORNAL-3499(Vol. II) p. 58-62) CRITICAL ARRAYS OF U(93.2) METAL CYLINDERS. J. T. Thomas (Oak Ridge National Lab., Tenn.).

Critical experiments were performed with three-dimensional arrays of individually subcritical units of U(93.2) metal with various thicknesses of paraffin reflector. The units consisted of cylinders containing about 21 and 26 kg of U(93.2) supported on stainless steel rods and separated by stainless steel spacers. In some arrays neutron-moderating materials surrounded the individual units. Within the range of the experiments, the effect of placing a 15.24-cm-thick paraffin reflector around an array of a particular density was to reduce the number of units required for criticality by a factor of ~12. The critical number was further reduced by a factor of ~2 by the addition of 1.270-cm-thick Plexiglas between adjacent units. (auth)


Criticality studies were made of three-dimensional arrays of uranium metal cylinders enriched to 93.2 wt% U²₃⁵. Four weight groups of units, ranging from 10.5 to 26.2 kg of uranium in five sizes, were employed to determine the critical surface separation between units as a function of the number in an array. The critical uranium density (the ratio of the mass of a unit to the volume of the lattice cell it occupies in a critical array) was observed to depend upon a number of factors. Changes in kᵣᵣ of the individual units, caused by altering their shape, produced inverse changes in the density, provided the array shape was unchanged. The density of the array varied with its shape in a manner similar to that resulting from corresponding changes in a single critical unit. Addition of an hydrogenous reflector reduced the density of the array by factors much greater than those observed for similar reflection of a single critical unit. The density was found to be minimal when the units were separated by an hydrogenous moderator 4.9-cm thick. The separate effects of moderation and reflection of arrays are not additive. In several experiments it was observed that an array comprised of one-half of each of two different critical arrays was subcritical. Part I of the study is ORNL-TM-719. (auth)
296.

36782 (HW-81659(p.33-44)) AN ANALYSIS OF PLUTONIUM-LIGHT WATER CRITICAL EXPERIMENTS.
J. R. Worden, R. C. Liikala, and W. A. Reardon (General Electric Co. Hanford Atomic Products Operation, Rich­
land, Wash.).

A detailed analysis of three sets of critical experiments, at 1.8, 2, and 5 wt % Pu in Pu-Al alloys, has been made.
These experiments all involved critical approaches using Pu-Al rods in light water at various lattice spacings. Three
areas of potential difficulty in the analysis of plutonium systems were studied: the basic cross section data; cal­
culation of the space-energy distribution of thermalized neutrons in a lattice cell; and methods of calculating the
neutron absorption rate in the 1.056-eV resonance of $^{239}$Pu.

1965

297.

5477 (HW-80296) APPROACH TO CRITICAL AND CALIBRATION EXPERIMENTS IN THE PLUTONIUM RF-CYCLE CRITICAL FACILITY. R. A. Bennett and L. C.
Schmid (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). July 1964. Contract AT(45-
1)-1356. 54p. Dep.: $3.00(cy), 2(mm) OTS.

The Plutonium Recycle Critical Facility (PRCF) is de­
signed for investigations of reactor phenomena associated with either D_2O or H_2O moderator. The report describes
the initial approach to critical experiment and power level, control rod, and moderator level calibration experiments
conducted in the PRCF moderated with D_2O. Results of measurements of the longitudinal and radial flux distribu­
tion are included separate from the power calibration. Control rod calibration results include integral and dif­
ferential worth measurements as measured by positive periods. Moderator level calibration results include the
integral and differential worth of moderator level changes between 93 and 105 in. In a few cases the experimental
results are compared to analytical results using three-group diffusion theory. The reactor is critical at a moderator
level of 105 in. with 25 UO_2 fuel clusters surrounded by 30 Pu-Al fuel clusters at an 8 in. lattice spacing. Diffusion
theory adequately describes the spatial distributions of the thermal-neutron density. The maximum flux in the
reactor is $4.6 \times 10^7$ neutrons/cm$^2$ sec at 100 w total power. The control rods are worth approximately 2 mk. The
moderator level coefficient of reactivity is 0.18 mk/in. at 105 in. (auth)

298.


The reactors studied are heterogeneous, normal water­
moderated, uranium-dioxide-fueled cores with water-to­
fuel ratios of 1:1, 1.5:1, 2:1, 3:1, and 4:1. The fuel rods
are held in a hexagonal array by upper and lower grid
plates. The control rods are clusters of three poison-fuel elements. Each control element is 0.660 in. in outside di­
ameter and has an upper section of packed boron carbide and a lower section of uranium dioxide fuel. These control
rods are designed to be located anywhere in the core and
are guided by the grid plates; with the control rods at their
upper limits, the core is uniformly fuel and moderator.
The initial experiments measured critical sizes, flux
shapes, cadmium ratios, peripheral fuel rod worths, water
gap worths, and control rod worths. The values and prob­
able errors for these parameters are tabulated. Hand­
computational methods of determining $k$ by a multigroup
 technique, which uses four energy groups, and a multi­
factor technique, which uses a resonance fission factor in
addition to the usual four factors, are shown. Control rod
worths are calculated by both a simple two-group model
and a modified two-group model that takes the flux per­
turbation into account. Agreement between theory and ex­
periment in the determination of $k$ was as good as expected;
k was shown to be within 6% of one for the multifactor
method and within 4% of one for the multigroup method. The
poor agreement found between theory and experiment for
the determination of control rod worths indicates the need
for a better theoretical model. (Dissertation Abstr., 25:

300.

10495 MEASUREMENT OF THE INFINITE MULTI­
PLICATION FACTOR IN A NATURAL URANIUM, LIGHT­
WATER LATTICE. T. W. T. Burnett and T. G. William­
on (Univ. of Virginia, Charlottesville). Nucl. Sci. Eng.,
The infinite multiplication factor, $k_{in}$ is one of the basic
parameters of a subcritical assembly. Usually, these assemblies are designed for maximum k_s; however, it is difficult to conduct laboratory experiments which yield a value of k_s to reasonable accuracy. Common methods, such as the loading technique and exponential experiment, are of doubtful validity or require apparatus not always available. Pulsing techniques are widely accepted, but are difficult to apply to reflected assemblies. An alternative approach is used. It is based on the integration of the thermal-neutron flux over the equivalent infinite medium. Use of variations in the method with poisoned assemblies eliminates the need for accurate determinations of the source strength, the absolute thermal-flux calibration, and the epithermal parameters of the medium. The theory is general and can be applied with a minimum of equipment. The results obtained from this method (and its variations) were checked by pulse measurements on the bare assembly and by a four-factor formula calculation. All results agree to within 2%. (auth)


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

A simple, approximate method is used for the calculation of safe limits for lattices of slightly enriched uranium and uranium oxide in water. The material buckling is calculated by an asymptotic multigroup transport code and the extrapolations distances by a two-group diffusion theory code. Data from exponential and critical experiments on these lattices in water are reviewed briefly. The calculations and experimental data were used to calculate critical and safe dimensions for spheres, infinite cylinders, and infinite slabs. (D.L.C.)


Measurements related to reactor physics parameters were made in three heavy water lattices. The three lattices studied contained .25 inch diameter, 1.03% 235U uranium metal rods in triangular arrays spaced at 1.25, 1.75, and 2.50 inches. The following four microscopic parameters were measured in each of the three lattices studied: (1) the ratio of the average epithermal 235U capture rate in the fuel rod to the average subcadmium 235U capture rate in the fuel rod, (2) the ratio of the average epithermal 235U fission rate to the average subcadmium 235U fission rate in the fuel rod, (3) the ratio of the average 235U capture rate to the average 235U fission rate in the fuel rod, and (4) the ratio of the average 235U fission rate to the average 235U fission rate in the fuel rod. (M.O.W.)

303. 3521 (TID-21258) PROGRESSIVE SUBSTITUTION EXPERIMENTS IN UO, LATTICES MODERATED BY D2O/H2O MIXTURES. H. R. Frazier (Norway. Institute for Atomenergi, Kjeller). Aug. 1964. 29p. (KR-30; NORA-21). Dep. (mn); $2.00 (cy), (1mn) GTS.

Buckling measurements for cores of uranium oxide (3% enriched) in different mixtures of D2O/H2O were per-

formed in the NORA reactor by means of a progressive substitution technique. In order to check the results, some experiments were also carried out by the substitution technique in critical lattices for which the material buckling was already known. Some subcritical experiments were also performed to give additional information about the buckling obtained by substitution experiments. The analysis was done by three-region, two-group theory, and a correction was introduced in order to take into account the effect of the reflector. For a D2O concentration of 99.50% and a lattice pitch of 6.54 cm, the material buckling with void was obtained by three-region, one group theory. All the results were found to agree satisfactorily with the results from critical experiments. Axial flux distributions were measured in some cases by using copper foils and a small fission chamber. (auth)

304. 33573 COMPARISON OF LATTICE PHYSICS EXPERIMENTS IN HEATED GRAPHITE STACKS CONTAINING PLUTONIUM-URANIUM FUEL WITH THEORETICAL PREDICTION. Gibson, M.; Harper, R. G. (Atomic Energy Establishment, Winfrith, Dorset, Eng.). I. Nucl. Energy, Pt. A & B, 19: 343-56 (May 1965). Measurements of buckling and flux fine structure and fission-rate distributions in graphite-moderated lattices fuelled with plutonium-uranium metal rods at temperatures up to 400°C are summarized. The results are of general interest in the development of rigorous methods of calculation for plutonium-bearing systems. The measured values are compared with theoretical calculations which determine a detailed representation of the neutron energy spectrum and incorporate fundamental nuclear data libraries. There is general overall agreement between the measured values and the theoretical predictions. (auth)


A series of critical and subcritical experiments were
performed to determine values of material buckling and $k_{\text{eff}}$ for graphite assemblies fueled with slightly enriched (3.02 wt %) uranium carbide fuel elements. Lattice arrangements were studied which included control channel mockups for triangular lattice pitches of 9, 11, and 12 inches. Experiments were performed to determine critical mass for two-region (uranium-carbide and uranium-metal-fueled) assemblies and $k_{\text{eff}}$ for subcritical uranium-carbide-fueled assemblies. Exponential experiments were carried out to determine material buckling for subcritical assemblies containing uranium carbide fuel elements. Additional assemblies of varying size were used to study diffusion plant, two and one, dependence of material buckling upon assembly size. The exponential studies were corrected for these effects. Material buckling and $k_{\text{eff}}$ values were also inferred from the two-region critical mass measurements using four-group theory, and these values were compared to the results obtained from subcritical measurements. The values inferred from the critical mass measurements provided a more consistent set of results. These inferred values also were compared to calculated results. The agreement with results obtained using multigroup theory is very good while the agreement with results from few-group calculations is poor. (auth)


Consideration was given specifically to arrays whose units are individually safe for diffusion plant operations and to the derivation of optimum spacing criteria for these units. Part I of the report covers that portion of the work concerned with homogeneous arrays of interacting uranium solution units, in air, and also includes results for arrays of high-density uranium metal units. From the mass-volume relations developed from the calculations, it appears that, for equal volume arrays, the minimum critical U-235 mass will occur at an intermediate U-235 enrichment appreciably lower than the 93% level at which the single unit minimum critical mass is normally found. (auth)


Exponential and approach-to-critical experiments were conducted with slightly enriched uranium in light water. The uranium was fabricated into elements of 0.175 to 1.66 in. diam with enrichments of 1.007 to 3.063 wt % $^{235}$U. Exponential measurements were also made with natural uranium rods (0.925 in. diam) in light water. Analytical methods were used to correlate the experimental results and extend the data to include uranium rods containing 4.0 and 5.0 wt % $^{235}$U. (auth)
between the reactivity and separation of units. An examination of this relation for regular arrays of spheres is made in terms of a simple theoretical model which focuses attention on the interaction probabilities between pairs of units, i.e., the proportion of the neutrons emitted by one unit which reach another. Formulation of balance equations connecting the fluxes entering and leaving each unit then allows assessment of the criticality of the array. The interaction probabilities are taken as mean fractional solid angles. Attention is drawn to the possibility of using Monte Carlo methods for the evaluation of interaction probabilities for systems of complex geometry and the GEM Monte Carlo code for the IHE 7000 is used to evaluate cases where other spheres in an array partially intervene between source and target spheres. The values obtained are separately tabulated as being of interest outside the particular arrays considered. A new code incorporating a specially simplified variant of GEM and provision for solution of the neutron balance equations is described as a ready means for estimation of the criticality of arrays for which the unit reactivities are known. A related discussion of factors affecting the criticality of arrays is presented. Factors considered include the size, shape, composition and density of units, proximity to reflectors and the mixing of units of differing types. Comparison is made between the theoretical model and published critical data for both unreflected and reflected arrays. Recently reported data for systems of this kind from both experimental and calculational sources allow a reasonably comprehensive comparison. New results are also given on the effect of moving units out of the plane of a two dimensional array and on the interaction of small numbers of slab tanks. The latter is seen as a special case and is investigated in some detail; a simple empirical correlation is given for air spacing and it is shown that there is little to be gained by attempting decoupling with shielding materials. (auth)

1966

315.


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

Central to an understanding of the criticality of interacting arrays is a knowledge of the quantitative relation

Lattices 313-317

between the reactivity and separation of units. An examination of this relation for regular arrays of spheres is made in terms of a simple theoretical model which focuses attention on the interaction probabilities between pairs of units, i.e., the proportion of the neutrons emitted by one unit which reach another. Formulation of balance equations connecting the fluxes entering and leaving each unit then allows assessment of the criticality of the array. The interaction probabilities are taken as mean fractional solid angles. Attention is drawn to the possibility of using Monte Carlo methods for the evaluation of interaction probabilities for systems of complex geometry and the GEM Monte Carlo code for the IHE 7000 is used to evaluate cases where other spheres in an array partially intervene between source and target spheres. The values obtained are separately tabulated as being of interest outside the particular arrays considered. A new code incorporating a specially simplified variant of GEM and provision for solution of the neutron balance equations is described as a ready means for estimation of the criticality of arrays for which the unit reactivities are known. A related discussion of factors affecting the criticality of arrays is presented. Factors considered include the size, shape, composition and density of units, proximity to reflectors and the mixing of units of differing types. Comparison is made between the theoretical model and published critical data for both unreflected and reflected arrays. Recently reported data for systems of this kind from both experimental and calculational sources allow a reasonably comprehensive comparison. New results are also given on the effect of moving units out of the plane of a two dimensional array and on the interaction of small numbers of slab tanks. The latter is seen as a special case and is investigated in some detail; a simple empirical correlation is given for air spacing and it is shown that there is little to be gained by attempting decoupling with shielding materials. (auth)

316.


Critical experiments were performed using fuel elements of type Ek-10 (these are standard fuel elements used in reactors of type VR-S) and the organic moderators monoisopropylbiphenyl (which is a product of the alkylation of diphenyl) and gas-oil (a straight distillation fraction of natural oil representing a mixture of aromatic naphthene and paraffin hydrocarbons) as well as an aeous moderator. The main results of these critical experiments and a description of the reactor are included. (M.O.W.)

317.


Parameter measurements in a 1.1% enriched UO2 lattice with H:U = 0.42 were performed. These measurements are an extension of an experimental program in the TRX critical facility of the Bettis Atomic Power Laboratory. Earlier
Lattices 318-324

Measurements were made for a wide range of water-to-
uarum (H2O:U) volume ratios (1:1 to 8:1) using 4-ft
(1.2-m)-high slightly enriched, 0.387-in. (0.98-cm)-diam
urium metal or oxide fuel rods clad with aluminum. The new
data were compared with current analytic techniques,
using both P-1 and P-3 multigroup analysis in the epi-
thermal-neutron energy range and Monte Carlo multi-
group methods for thermal neutrons. This extremely undermod-
erated lattice provides a very stringent test for both the
computational methods and the neutron cross sections used.
The quantities measured were: the ratio of epithermal-to-
thermal neutrons; the ratio of capture in 238U (238U); the ratio of cap-
tures in 232U (232U); the modified conversion ratio, 2.; the ratio of 232U fissions to 233U fissions (233U); and the ratio of epithermal-to-thermal 233U fissions (233U).

In addition, activation experiments were conducted with thermal-neu-
tron detectors of widely different spectral response. The results
indicate that the calculation methods predict the param-
ters very well, except for 238U. The discrepancy in 238U may
to be due to inadequate 238U inelastic scattering cross sections,
and the ratio of epithermal-to-thermal 235U fissions
(235U). The quantities measured were: the ratio of epithermal-to-
thermal activations of widely different spectral response. The results
indicate that the calculation methods predict the param-
ters very well, except for 238U. The discrepancy in 238U may
to be due to inadequate 238U inelastic scattering cross sections,
and the ratio of epithermal-to-thermal 235U fissions
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ters very well, except for 238U. The discrepancy in 238U may
to be due to inadequate 238U inelastic scattering cross sections,
and the ratio of epithermal-to-thermal 235U fissions
(235U). The quantities measured were: the ratio of epithermale

LATTICES OF U AND UO2 RODS IN WATER. Clark, Hugh K. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C., Savannah River Lab.). Feb. 1966. Contract AT-(07-2)-1. 60p. (CONF-651103-29), Dep. mn. CFSTI $3.00 cy, $0.75 mn.

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

A survey is given of available critical and exponential data. Critical and exponential data obtained with water-reflected lattices of slightly enriched uranium and uranium oxide rods. Calculations are made for these lattices by an asymptotic multigroup buckling code and by a two-group diffusion theory code employing parameters generated in the first code. Comparison between calculations and experiments is made in terms of a k_eff, which is the ratio of the calculated k to that calculated from the experimental dimensions and flux traverses and from the calculated migration areas and extrapolation distances. For some of the data, comparisons are also made between the present method of calculation and a more highly sophisticated method. Reasonable care is taken in the present calculations to take account of all important effects; but since the experiments are used to normalize the calculations, absolute accuracy in the calculations is not necessary. A least squares treatment is given to k_eff, the parameter relating calculation and experiment, in order to obtain an average value of k_eff as a function of the ratio of water to uranium. As low enrichments, k_eff is also allowed to vary with rod diameter and enrichment. Nearly all values of k_eff lie within ±0.1 of the average curves. Critical masses and dimensions are calculated with the buckling and two-group codes to correspond to values of k_eff lying on the average curves; and safe masses and dimensions are calculated to correspond to values of k_eff lying 0.02 below the average curves. Tables of minimum critical and maximum safe values are presented as a function of enrichment. (auth)
dissipated throughout a light-water-moderated nuclear reactor. The lumped poisons were borosilicate glass rods (12.6\% B_2O_3) of 0.460- and 0.326-inch diameters, silicon glass rods (3.0\% B_2O_3) of 0.326-inch diameter, and aluminum-clad B,C rods. The fuel was 4.24\% enriched UO_2 rods of 0.475-inch OD. Lattice non-moderator/moderator volume ratios were 0.750. Measurements included critical size and composition, poison rod reactivity worth, \(\bar{r}/\bar{h}\) and excess reactivity, gross power distribution, and thermal flux distribution around the central poison rod. Measurements are tabulated, and results are summarized and discussed. (auth)

325.

32793 METHODS OF MATERIAL BUCKLING DETERMINATION IN THE HELENA SUBCRITICAL ASSEMBLIES. Dabrowski; Cyryl; Mikulski, Andrze T. 33p. (CONF-651123-7). ORAU.


A detailed analysis of the material buckling determination in the natural uranium fuelled graphite moderated HELENA subcritical assemblies was carried out by means of the following methods: thermal neutron flux measurement along the vertical axis with an assumption of the point thermal neutron source either in one-region or two-region assembly, measurement of the thermal neutron flux along the vertical axis and in one horizontal plane, measurement of the thermal neutron flux in two horizontal planes, and measurement of the thermal neutron flux in one vertical plane of the assembly. Three assemblies of 24 cm lattice pitch and the following dimensions: 180 x 180 x 237, 180 x 240 x 237 and 240 x 240 x 237 cm were investigated. The following results have been obtained: migration area anisotropy coefficient \(\alpha_m = 1.096 \times 10^{-2}\) and material buckling (73.5 \pm 4.3) \times 10^{-7} \text{ cm}^2. (auth)

326.

3397 REACTOR PHYSICS MEASUREMENTS IN SLIGHTLY ENRICHED URANIUM D_2O LATTICES.


327.

44993 (DP-1054) LATTICE MEASUREMENTS ON TUBULAR FUEL ASSEMBLIES IN D_2O. Pike, Harold R. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Lab.). Aug. 1966. Contract AT(07-2)-1. 16p. Dep. m. CFSTI \$1.00 cy. \$0.50 mn.

Substitution measurements in the Process Development Pile (PDP) were used to determine buckling and diffusion coefficients for D_2O-moderated lattices of tubular natural UO_2 fuel assemblies. Four types of assemblies were investigated at triangular lattice pitches of 9.33, 11.10, and 12.12 inches. Coolants within the different assemblies included D_2O, air, and mockups of H_2O fog and organic liquids. (auth)
critical number in a similar hexagonal lattice was 100. A double tier of 1000 closely packed 8-in.-long slugs was critical when submerged in water. The dia of an infinitely long, unreflected critical cylindrical array of optimum spaced rods was between 15 and 17.5 in.; the dia of a water-reflected cylindrical array was about 12 in. It was shown that randomly spaced slugs produce less source neutron multiplication than do lattices of the same overall density. (auth)

330.

38532 (NASA-TN-D-3555) CRITICAL MASS STUDIES WITH THE NASA ZERO POWER REACTOR II. HETEROGENEOUS ARRAYS OF CYLINDRICAL VOIDS. Fox, Thomas A.; Mueller, Robert A.; Ford, C. Hubbard (National Aeronautics and Space Administration, Cleveland, Ohio. Lewis Research Center). Aug. 1966. Spec. CFSTI $1.00 cy, $0.50 mn.

The ZPR-II was used to determine several critical cylindrical configurations of aqueous fuel solutions that contain symmetrically arranged parallel to the axis of the reactor, and extend the height of the core. A wide range of highly enriched aqueous uranyl fluoride fuel concentrations are studied. Data are also presented on the thermal neutron flux distributions in the central radial planes and on the variation of void reactivity Importance with radial position. (H.D.R.)

331.


A new method for measuring values of the multiplication factor for an infinite medium, $k_{\infty}$, was developed. The method involves addition of neutron absorbers to subcritical assemblies. Measurements of the values of $k_{\infty}$ for three lattices of U metal rods in heavy water were made. Solid copper rods were used as absorbers. The values of $k_{\infty}$ were found to be in agreement, within experimental uncertainties, with values obtained with the four-factor formula. The method used is closely related to the Hanford (PCTR) technique for measuring $k_{\infty}$ which requires a critical reactor. Since it is nearly a measurement of $(k_{\infty} - 1)$, and because the measured values are independent of a number of sources of error in the four-factor formula, the uncertainties in the measured values should be smaller than those obtained with the four-factor formula. The work shows that a 3% uncertainty in $(k_{\infty} - 1)$ may be obtained with the method, comparable with the uncertainty in PCTR measurements. Measured values of the neutron regeneration factor were also determined. They agree well with values computed with the THERMOS code. The results of exploratory experiments indicate that it is possible to do the measurements in two-region subcritical assemblies. (auth)

333.


The reactivity effects of H₂O addition into the moderator of a heavy water moderated reactor with natural uranium are investigated. The values for various radii of cylindrical fuel rods and various water-uranium ratios were calculated as a function of light water concentration in the moderator. The maximum values of the bucking which can be attained by addition of H₂O at various parameters of the lattice are discussed. It appears that the reactivity increases only in systems with a low moderator-to-fuel volume ratio in the cell. (auth)

334.


The critical dimensions of lattices of SPERT-D fuel elements in several non-reactor environments were determined to establish specifications for use in storage, transportation, and chemical processing operations. Each fuel element contained about 300 g of $^{235}$U in 22 aluminum-clad flat plates. In addition to lattices with water moderator and reflector, a dilute aqueous solution of uranyl nitrate was used in some experiments to simulate a dissolver. In still other experiments, varying amounts of boron were added to the uranyl nitrate solution to determine its effect as a soluble neutron absorber in chemical process equipment. It was shown, for example, that a minimum of about 3.5 kg of $^{235}$U is required in a critical lattice moderated and reflected by water. This mass is reduced to 2.8 kg (contained in the elements) when $\mathrm{UO}_2$ solution having a $^{235}$U concentration of 3.93 g/liter was substituted for water. It was increased to 13.6 kg when 1.118 g of natural boron was added to each liter of the uranyl nitrate solution. (auth)

335.


The developmental work on nuclear analytical methods for distributed control pins is summarized. All the calculations utilized a nuclear code package that was selected after a comparative study of available codes. Four critical experiments have been selected as typical of the B poison pins used in the experiments; B concentration, pin size, and pin distribution were considered. The reported error of the calculated multiplication constant indicates the adequacy of the analytical model. Further work required in some areas is discussed. (auth)

336.

14080 PULSED NEUTRON STUDIES OF BeO-NATURAL URANIUM LATTICES. Joshi, B. V.; Nargundkar, V. R.; Subbarao, K. (Atomic Energy Establishment
The use of the pulsed neutron method for determination of the diffusion parameters and multiplication constants of lattices is described. The diffusion parameters and multiplication constants of BeO-natural uranium lattices are obtained by this method. The uranium rods used were 2.92 cm diameter clad in aluminum 0.072 cm thick and were arranged vertically in channels of square cross section 5 x 5 cm, in a square lattice of pitch 15 cm. The neutron bursts were produced from a cascade accelerator by pulsing the ion source and using the Be(d,n) reaction. The detectors were enriched boron trifluoride proportional counters. The space and time harmonics were eliminated. All the diffusion and multiplication constants were determined by the method of least squares fit by three different approaches. The prompt critical buckling was determined by solving the decay constant equation for \( A = 0 \).

(aut)

LATTICES 337-340

339.


The predictions of the ARGOSY method of calculation are compared with buckling and reaction rate measurements on graphite lattices containing plutonium enriched oxide cluster fuels. Most of the measurements were obtained on one lattice pitch giving a graphite-to-fuel volume ratio similar to that in the Windscale A.G.R. Apart from a reduction of 10% in the values of the capture and fission resonance integrals (4 ev to 10 keV) of all nuclides the method as coded in ARGOSY III uses basic nuclear data. It is shown that the buckling predictions of ARGOSY III at room temperature are in agreement with the experimentally determined values within approximately the experimental error, i.e., equivalent to \( \pm 0.5\% \) in reactivity. When systematic errors are removed, however, a linear trend with experimental error, i.e., equivalent to \( \pm 0.5\% \) high for fuel with 80% of fission occurring in \( { }^{239}\text{Pu} \). The calculated reactivity being \( (0.6 \pm 0.3)\% \) high for fuel with 80% of fission occurring in \( { }^{239}\text{Pu} \). The experimentally determined change of buckling between 29 and 390°C is predicted to within the experimental error, i.e., equivalent to \( \pm 0.7 \text{meV/°C} \). Reaction rate ratios and radial power distributions at all temperatures are predicted well by ARGOSY III. (aut)

338.


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

The interaction effect is studied in arrays of fissionable materials. Both experimental and theoretical efforts are reported. The arrays studied experimentally are composed of Pu metal units. Array geometries are simple; the basic units are cylinders and the arrays are cubical. Bare arrays are studied, as well as those with internal moderation or external reflection. 130 basic units are available so that arrays up to \( 5 \times 5 \times 5 \) in size can be studied. The purposes of the study are two-fold: to provide clean, critical data to use in the development and normalization of computational techniques; and the data as such are expected to be useful for interpretation and extrapolation for application to actual systems of interest. The basic array unit is a 3-kg cylinder of Pu metal, 4.62 cm high, 6.53 cm in diameter, and of density 19.54 g/cm\(^3\). The height-to-diameter ratio is chosen so that a double unit consisting of two basic units, one on top of the other, will have the same shape factor as a single basic unit. A brief description of the canning, assembly equipment, and the experimental building is presented. A short discussion of the safety evaluation (startup accident, step insertion of reactivity) is also given. Pertinent calculations are performed using both the GEM and SORIS neutron Monte Carlo electronic computer programs. Experimental results for arrays consisting of two units up to the maximum number are presented. (aut)

339.


From American Nuclear Society Meeting, Denver, Colo.

The lattice characteristics of seed fuel elements containing \( { }^{233}\text{U} \) and \( { }^{232}\text{U} \) were measured in a series of small seed-blanket critical assemblies. The measurements included critical axial bucklings, seed power shapes, the ratio of epithermal to thermal thorium captures in blanket regions, and the ratio of epithermal to thermal \( { }^{233}\text{U} \) fissile through seed and blanket regions. The experimental results are compared with calculations performed with diffusion theory, Monte Carlo, and neutron transport techniques. (H.D.R.)

340.


A series of critical experiments was undertaken in which the reactivity worths of clusters and uniform arrays of Ag-In-Cd absorber rods were measured in water moderated uranium-oxide cores of low enrichment. Experimental parameters included enrichment, lattice pitch, diameter of poison and fuel rods, and absorber rod cladding thickness. Reactivity worths were obtained by measuring the critical water heights of the test and reference cases, and integrating the differential water worth curve between those limits. Some of the experiments had very small worths (e.g., absorber rod cladding), so good core reproducibility was required. These experiments were performed using a variety of fuel including stainless steel clad fuel rods of 2.7 wt %, 3.7 wt %, and 5.7 wt % enrichments, and Zircaloy-4 clad rods of 2.72 wt % enrichment. (M.O.W.)
Lattices 341-348

341.  

342.  

Six critical experiments in PuO₂ lattices performed at Hanford are calculated with different methods. (auth)

343.  

The problems in the neutron calculation in a reactor containing ²³⁹Pu and the difference with ²³⁹U are examined. A general scheme for the calculation is proposed and this plan is optimized for the neutron cross section data used. A comparison is made between theory and experimental values for the Hanford critical experiments. (J.S.R.)

344.  

The results of a series of buckling and microparameter measurements on square lattices of 0.96% enriched U metal rods are given. The experiments were performed in the subcritical facility MINOR. A special chapter deals with approach to criticality experiments. These were necessary in order to determine whether a lattice would reach criticality at finite D₂O levels. The method used gave very reliable extrapolations for the determination of critical D₂O-heights. The experimental bucklings are compared with a buckling codes and show considerable deviations in the undermoderated region. (auth)

345.  

Lattice parameters for assemblies of fuel element bundles of the WWR type with uniform and different ²³⁹U enrichment were calculated by the programs GRUPA and EMMI and compared with the values determined experimentally on critical assemblies. Although the calculational methods are relatively simple, a good agreement resulted between the experimentally determined and the calculated values of the material buckling bₘ, an effective migration area Mₘ and the infinite multiplication factor kₘ of the lattice. (auth)

346.  

347.  

Clusters with three different pin pitches were investigated by means of the substitution technique as regards the material buckling versus lattice pitch and the changes of buckling and diffusion coefficients caused by the introduction of air as “cooler.” The buckling change with void is found to be in approximate agreement with that predicted by a modified version of BURNUP-5 calculations, but the radial diffusion coefficient changes less than expected giving rise to a more pronounced anisotropy. The measured changes of the diffusion coefficients can, however, be well correlated by means of a simple formula combining two extreme definitions of the average diffusion coefficient. Activation studies were performed in one lattice configuration on voided and D₂O-filled clusters. The fine structure was measured by means of copper foils and wires. (auth)

348.  

Neutron spectra were measured both integrally and differentially and then compared with theoretical calculations. Measurements were made in highly absorbing media in the University of Florida Training Reactor (UFTTR) thermal column, in the UFTTR core, and in the subcritical assembly sitting on top of the graphite pedestal of the UFTTR. The subcritical assembly contained some typical D₂O-moderated natural uranium lattices. In the thermal column of the UFTTR, thermal neutron spectra were measured integrally using activation detectors such as ¹⁹⁷Au, ¹⁹⁴Dy, ¹⁵⁵Eu, ¹⁴⁷Eu, ²³⁹Pu. The measurements were made in a stainless steel rod, a natural uranium slab, and an Al can filled with borated water, and results were interpreted using Westcott's formulation. The effective neutron temperature and the epithermal index, γ²γ/T₀, were measured by using activation detectors in the UFTTR core between the two fuel boxes in the North-South center line. The result of the effective neutron temperature measurement was compared with the open-beam differential spectrum and found to be in good agreement. A crystal diffraction spectrometer for differential spectrum measurements in the subcritical assembly was used. The neutron spectrum out of the subcritical assembly was measured for two different lattice pitches using Mark I and Mark V-B natural U fuel elements in D₂O. A neutron beam was extracted from the cell boundary and from the center of the cell for each of the three lattice arrangements. The results were compared with the theoretical spectra calculated with the THERMOS code and found to be in excellent agreement. Integral measurements were also made in the subcritical using ¹⁵¹Eu and ¹⁹⁷Au detectors. The experimental activation measurements were compared with the THERMOS calculations and were also found to be in excellent agreement. Finally, the effective neutron temperature change in a unit cell for four U- D₂O
lattice arrangements was measured differentially and integrally and compared with the theoretical calculations. The results were in good agreement. (A.G.W.)

349.
Relative total neutron densities and hyperfine Mn-wire activation distributions are derived from measurements in the central cell of a triangular array of 55 19-element natural uranium metal assemblies in the ZED-2 reactor. Results are presented for lattice pitches of 20, 24, 28, and 40 cm. "Coolants" used were, (1) D$_2$O of moderator purity, and (2) air to simulate loss of coolant condition. Data indicated that the neutron-density depression in the cell is less upon loss of coolant. (auth)

350.
From IAEA Symposium on Criticality Control of Fissile Materials, Stockholm.
Due to the large amounts of plutonium and uranium stored at LASL, criticality studies were made for stored arrays. Critical masses of air-spaced arrays of 10.5, 15.7, 20.5, and 26.2 Kg cylinders of 55% enriched uranium are presented. Critical masses are given for spheres with core densities reduced by the fraction $p/p_0$; water reflected U(93) metal, U(93)O$_2$, U(93)F$_4$, and unreflected U(93) metal, (M.O.W.)

351.
A comparison between the calculated and measured Laplacian of a heavy water gas-cooled lattice cell with natural U fuel is given. The method of the correlation of the program PARAMETRY II is discussed together with the deficiencies of the calculational method. An estimate of the influence of incorrect values of some physical constants and geometrical characteristics on the Laplacian is made. In the conclusion some recommendations concerning the improvement of the calculational method are added and attention is devoted to necessary extension of the experimental material. (auth)

352.
12492  (AECL-2514) COMPARISON OF CALCULATED AND MEASURED CRITICAL SIZES FOR SOME LATTICES 349-355

353.
Experiments in support of specifications for the safe storage and transportation of fissile materials are described. The effect of hydrogenous reflectors surrounding arrays, as well as the introduction of moderating materials into arrays, are presented. Other factors influencing the number of units in a critical array are illustrated and an estimate of their magnitude is given. (auth)

354.
The quantity of material necessary to effectively isolate a multiplying media is of primary concern to the design of shipping containers and formulation of storage regulations for fissionable materials. The results are given of a series of critical experiments in which the isolating properties of various materials were investigated. Those materials tested were placed at the interface between two arrays of multiplying media that comprised the core of a critical assembly. The critical length of the fuel core was then determined as a function of the test material thickness; the effective isolation thickness of a material was taken to be that thickness for which the fuel core critical length approached an asymptotic value. Comparative data were also obtained on the reflector savings associated with the various test materials, where the reflector savings is defined as the decrease in critical length due to the presence of the reflector. (M.O.W.)

355.
A neutron multiplication experiment was performed with 21 storage capsules containing high isotopic analysis plutonium dioxide. Each storage capsule contained approximately 80 grams of the $^{239}$Pu isotope. The capsules were assembled in a heterogeneous array spaced four inches center-to-center and measurements were made independently in a water and in an air medium. The results indicated that no neutron multiplication was detectable in the fully assembled array in a water medium. However, a slight neutron multiplication of 1.29 existed in the fully assembled array in an air medium. (auth)
Lattices 356-359

356.


The results of an experimental comparison of organic and aqueous moderators under identical conditions on a critical stand are presented. In these studies, monoisopropylbiphenyl was used as the organic moderator, the purpose of the experiment being to determine the critical mass of uranium in systems with organic and aqueous moderators for a given construction of fuel elements. The critical stand used is described, and the distribution of thermal neutrons is analyzed. The results show that the values of the migration area for media with monoisopropyl­biphenyl lie below the values for aqueous moderators by 40-70% for identical values of QH/Qo. Measurements were also carried out at different ratios of the active zone to determine the effect of the geometry of the active zone on the critical masses. These investigations showed that in the region where Dqemu = 1 when QH/Qo = 200 to 300 for monoisopropylbiphenyl and QH/Qo = 300 to 400 for aqueous moderators, the values of the critical masses are essentially independent of the geometry of the active zone. (ATD)

1967

357.


Measurements are described of the nuclear properties of water­moderated, slightly enriched U-fueled lattices, perturbed by boric acid and Ag-In-Cd control pins. Three critical assemblies were studied in this period. In all cases the lattices consisted of 0.475-in.-OD rods of 2.46%-enriched UO2 arrayed on a square pitch of 0.644 in. and moderated by H2O. Only the boric acid concentration in the moderator (and the critical dis) was changed, from zero to about 1.5 g B/l. The critical conditions, reactivity coefficients (Δρ/Δν and Δρ/Δν), buckling, and thermal disadvantage factor are reported. Methods being developed for the measurement of χp, the modified conversion ratio, and the epithermal neutron spectrum are described. (auth)

358.


Bucklings derived from activation measurements in the ZED-2 reactor are given for 28-element natural UO2 fuel assemblies. Measurements were made in D2O-moderated triangular arrays of 55 assemblies at pitches in the range 20 to 40 cm. The three coolants used were D2O of moderator purity, H2 to simulate a voided condition, and HB-40, an organic liquid. Bucklings obtained with H2 as coolant are greater than those for D2O and HB-40 coolants for the pitches investigated. Macroscopic variations of the ln-Cd ratio were measured with D2O and organic coolants within the range of pitches studied. (auth)

359.


From 2nd International Thorium Fuel Cycle Symposium, Gatlinburg, Tenn.

Measurements were made on D2O lattices of natural UO2 rods, 1.2% enriched UO2 rods, ThO2 rods, and their admixtures. The data from the buckling, spectral index, and flux distribution measurements are presented. Conversion ratio measurements were also made. (D.I.C.)
3. CROSS SECTIONS

1956

1. 1058 KAPL-813
Knolls Atomic Power Lab., Schenectady, N. Y.
SOME ACTIVATION MEASUREMENTS IN THE INTER-
MEDIATE ENERGY REGION. I. H. Dearnley, H. E.
Measurements of ratios of isotopic capture cross sections to \( \sigma^{U235} \) fission cross section \( \sigma_{\text{f}}^{U235} \) in an intermediate energy region have been made for a number of isotopes. Materials were activated by photoneutrons from a 2-Mev x-ray Be block source or by neutrons from side hole No. 4 of the Argonne CP-3 reactor. The ratio of \( \sigma_{\text{f}}/\sigma_{\text{U235}} \) was measured for Na\(^{23} \), Mn\(^{25} \), As\(^{75} \), Ru\(^{106} \), Sb\(^{120} \), Sn\(^{122} \), Sn\(^{122} \), Ba\(^{142} \), Pd\(^{110} \), Cd\(^{114} \), In\(^{115} \), Sn\(^{115} \), Sn\(^{115} \), Sn\(^{115} \), Sn\(^{115} \), La\(^{139} \), Pr\(^{141} \), Ce\(^{142} \), Nd\(^{148} \), W\(^{188} \), and Au\(^{197} \). The values obtained were considerably lower than predicted by the statistical theory of nuclei. (auth)

2. 6350 LA-1201
Los Alamos Scientific Lab., N. Mex.
A MEASUREMENT OF THE AVERAGE FISSION CROSS
SECCTIONS OF Pu\(^{239} \) AND Pu\(^{240} \) IN THE FAST REACTOR
(OTS).
A comparison chamber is described which was designed
to be inserted into a reactor port in order to measure fission rates from neutron fluxes which had undergone a minimum of energy degradation. The cross section was inferred from a comparison of the fission counting rates from foils of known Pu\(^{239} \) enrichments with the rates from standard foil containing only Pu\(^{240} \). Incident to the Pu\(^{239} \) fission cross section measurement the experiment also yielded a measurement of the average fission cross section for Pu\(^{240} \). (auth)

3. 4359 KAPL-511
Knolls Atomic Power Lab., Schenectady, N. Y.
DEPENDENCE OF \( \alpha \) OF U-233 ON TEMPERATURE.
J. B. Sampson and H. Hurwitz, Jr. August 14, 1951.
$0.20(OTS).
Two lines of reasoning are used in considering the temperature dependence of \( \alpha \) (ratio of radiative capture to fission) in U\(^{233} \) exposed to a thermal neutron flux. An experiment involving measurement of the ratio of fissions in U\(^{233} \) to capture in U\(^{233} \) with and without a Ag shell surrounding the U foil, is analyzed. The best value indicated from this experiment is \( \frac{\alpha}{\text{U233}} \sim (10^9) \times 10^{-5}$/°C, in contrast to Wigner's result of \( 22 \times 10^{-5}$/°C by a similar analysis. Indications from neutron velocity selector results are that \( \frac{\alpha}{\text{U233}} \) is more apt to be positive than negative, and possibly as large as \( 10 \times 10^{-5}$/°C. There is no experiment which is definitely inconsistent with the assumption \( \frac{\alpha}{\text{U233}} = 0 \), and moreover this zero value is a compromise between the qualitative values indicated by the two lines of reasoning considered. (auth)

4. 1642 ORNL-1992
Oak Ridge National Lab., Tenn.
A DIRECT COMPARISON OF SOME NUCLEAR PROPERT-
IES OF U-233 AND U-235. J. T. Thomas, J. K. Fox,
7405-eng-28.
A comparison has been made of some nuclear properties
of U\(^{233} \) and U\(^{235} \) based on data obtained in a series of
critical experiments. Aqueous solutions of uranyl oxyfluoride containing uranium enriched to about 90% in each of the two isotopes have been made critical in water
reflected spherical reactors having diameters of 26.4 and
32.0 cm. With the hypotheses that the reported nuclear constants for U\(^{233} \) are reliably known and that the neutron leakage spectra for U\(^{233} \) and U\(^{235} \) were equal for the same water reflected critical sphere, the value of \( \eta(U233) \) at
0.026 ev was determined to be \( 2.31 \pm 0.03 \). The critical masses of the two isotopes have been measured over a limited temperature range and corresponding values of the
Cross Sections 5-11

Reactivity temperature coefficient are reported. The results are tabulated. The critical mass of U235 in the larger sphere, unreflected, was 1.15 kg. The delayed neutron yield of the two isotopes were compared by using the period resulting from the withdrawal of a boron poison from the critical spheres. It is shown that the yield from U235 is about one third that from U238, in agreement with other determinations. (auth)

1957

5.

2739 KAPL-M-HB-3
Knolls Atomic Power Lab., Schenectady, N. Y.

6.

12204 AECU-3527
Westinghouse Electric Corp. Research Labs., [East], Pittsburgh, Penn.
Part 1 issued as AECU-3387. Part 11 issued as AECU-3402.
The papers presented are entitled: Resonance Capture IntegraLs as a Check on the Neutron Absorption Cross Section in Multigroup Tabulations; Multigroup Neutron Cross Sections; Cross Section Calculations for Fast Neutron Scattering; IV. Depth and Radius of the Real Potential in a Diffuse Surface Optical Model; and The Construction of Inelastic Scattering Matrices. (T.R.H.)

7.

9440 AERE-NP/R-2140
Five quantities relating to the behavior of fissile isotopes are tabulated for U235, U238, and Pu239. These quantities are the average number of neutrons emitted per fission, the average number of neutrons emitted per neutron absorbed, σf, the absorption cross section, σf, the fission cross section and the ratio (σf - σo)/σf. (M.H.R.)

8.

12504 KAPL-1246
Knolls Atomic Power Lab., Schenectady, N. Y.
The neutron capture-to-fission ratio was measured for a 5000 ev shielded U235 sample and found to be, for a mean fission energy of 15 kev, 0.406. The schematic diagram of the new three-stage mass spectrometer is given. An anode-coincidence scintillation spectrometer, which is being applied to the quantitative analysis of non-fission-product mixtures is described. Plans are being made to build a high-precision magnetic alpha analyzer. Pu239 fission and absorption cross sections and temperature coefficients of reactivity are analyzed. The average of a Breit-Wigner resonance over a Maxwelullan neutron spectrum is made.
An extension has been made to the theory of atomic displacements produced by irradiation in a solid, improving agreement with experiment. The effects of monocentric fast neutron on normal and gold-doped Ge were studied. (W.L.H.)

9.

2636 BNL-1575
Brookhaven National Lab., Upton, N. Y.
The experimental method and results of fast fission factor measurements on reactor fuel rods containing U238 are discussed. (D.E.B.)

10.

11359 KAPL-1793
Knolls Atomic Power Lab., Schenectady, N. Y.
A critical review is made of experimental information concerning the energy dependence of the capture-to-fission ratio, α, of Pu239. A form of this dependence is obtained which is shown to be consistent with nearly all measurements, including these recently reported for EBR-I. This dependence shows a rapid decrease with energy near 100 kev, and is intermediate in form between those of several previous suggestions. (auth)

11.

7171
The average number of neutrons $\nu_{\text{eff}}$ emitted by the isotopes $^{235}\text{U}$, $^{238}\text{U}$, and $^{239}\text{Pu}$ on capture of neutrons with energies from 30 to 900 keV is 714. It is discovered that in this energy region $\nu_{\text{eff}}$ increases substantially as the neutron energy increases. (auth)

1958

13.

9371


On the basis of data obtained from the U.S.A., the U.S.S.R., and England, the most probable values for the neutron cross sections and criticality constants for $^{235}\text{U}$, $^{238}\text{U}$, and $^{239}\text{Pu}$ are tabulated. (J.S.R.)

14.

4573


The ratio $\alpha = \sigma_{\text{c}}/\sigma_{\text{f}}$, where $\sigma_{\text{c}}$ is the neutron capture cross-section and $\sigma_{\text{f}}$ the neutron-induced fission cross-section, has been measured for $^{235}\text{U}$ as a function of neutron energy. A pulsed and collimated neutron beam is passed through a $^{235}\text{U}$ sample at the center of a large liquid scintillator. Captures and fissions are detected by means of their prompt gamma rays; elastic and inelastic scattering events are ignored because of smaller pulse heights. Fissions are distinguished from captures by means of delayed pulses from the capture of thermalized fission neutrons. It is found that in the neutron energy range $E_n = 0.175$ to 1.0 Mev the value of $\alpha$ is given approximately by $\alpha = 0.190 + 0.016E_n$. The accuracy of the determination of $\alpha$ is 10 to 15% in terms of the standard deviation and individual points. (auth)

15.

11726


Measurements of two independent types were made of the reactivity effect in a thermal test reactor of samples of $^{235}\text{U}$, $^{232}\text{U}$, and $^{239}\text{Pu}$. From these measurements average subcadmium values of $\epsilon$ (relative to $\epsilon$ of $^{235}\text{U}$) are obtained independently of other knowledge of the average absorption cross sections. Average absorption cross sections are also obtained from the measurements. Values of $\epsilon$ for $^{235}\text{U}$, $^{239}\text{Pu}$, and $^{239}\text{Pu}$ are respectively $2.23 \pm 0.034$, $1.927 \pm 0.024$, and $2.133 \pm 0.07$. The corresponding value of $\epsilon$ (0.0253 ev) of $^{233}\text{U}$ is found to be 2.025. A presentation of the method and results are given together with a comparison with previous work. (auth)

16.

9382


Analyses of uranium, neptunium, and plutonium isotopes have been carried out on dissolved MTR fuel elements. Effective reactor cross sections computed from these measurements are in good agreement with differential cross-section measurements. In particular, effective cross-section values were founded for $^{235}\text{U}$ and $^{238}\text{U}$ of 34 barns and 23 barns, respectively. (auth)

17.

6362


The degradation of neutron energies in a fast reactor is largely due to inelastic scattering. In a dilute fast system (large $^{233}\text{U}$ to $^{235}\text{U}$ atomic ratio) the neutron spectrum is then primarily determined by a fission spectrum distribution modified by inelastic scattering in $^{238}\text{U}$. In this investigation a set of ten-group fast cross sections for $^{238}\text{U}$ have been prepared with the inelastic cross sections below about 1.35 Mev based upon levels at 45, 150, and 700 kev. The inelastic transfer contributions from unknown higher levels were chosen to be consistent with the gross measurements of Bethe, Beyerler, and Carter, having the three-group energy division consisting of above 1.4 Mev between 0.4 and 1.4 Mev, and below 0.4 Mev. The ten-group fast cross sections were tested by comparing the calculated equilibrium spectrum, diffusion...
Cross Sections 18-22

length, and detector responses in natural uranium with reported experimental values found in the blanket of the Zephyr reactor and in the Snell experiments. (auth)

1959

19. 9257 BNL-483(p.53)
NEUTRON YIELDS FOR U\(^{235}\). R. J. Becley, p.53 [of]
THORIUM-\(^{233}\) SYMPOSIUM, SPONSORED BY THE
UNITED STATES ATOMIC ENERGY COMMISSION AT
BROOKHAVEN NATIONAL LABORATORY,
JANUARY 9-10, 1958. 1p.

Investigation of cross section curves to obtain average values of \(\eta\) for eight energy groups for \(U^{235}\) are reported. (W.L.H.)

19. 11377 CF-59-1-70
Oak Ridge National Lab., Tenn.
ABSOLUTE MEASUREMENT OF ETA BY THE
MANGANESE BATH TECHNIQUE. G. deSausilr and
7405-eng-26]. $3.30(sp), $2.40(mf) OTS.

An experiment is described for measuring \(\eta\) of \(U^{238}\)
and estimating the various errors involved. A value for
\(\eta\) can be obtained with a precision of 1% or better.
(auth)

20. 17438
COMPARISON OF CRITICAL EXPERIMENTS FOR THE
DETERMINATION OF ETA OF U\(^{235}\). D. W. Magnasun
and Reginald Gwin (Oak Ridge National Lab., Tenn.),

The series of experiments to determine \(\eta\) of \(U^{238}\) with
five concentrations of \(U^{235}\)O\(_2\)NO\(_3\) solutions and four
concentrations of \(U^{238}\)O\(_2\)NO\(_3\) solutions in a 69.2-cm-
diam. aluminum sphere is described. Boric acid was
used as a poison in the experiments in order to vary the
uranium concentration. Unpoisoned solutions in a 122-
cm-diam. sphere were also made critical. Experiments
and calculations are summarized. The best value of \(\eta\)
determined was 2.295 \pm 0.036. (W.D.M.)

21. 1444 ORNL-2609
Oak Ridge National Lab., Tenn.
NEUTRON PHYSICS DIVISION ANNUAL PROGRESS RE-
PORT FOR PERIOD ENDING SEPTEMBER 1, 1958.
(OTS).

A relatively low-power reactor research facility, the
Pool Critical Assembly (PCA), was installed in one end
of the pool at the Bulk Shielding Facility. A description of
the facility and the initial critical experiments is
given. Experiments to determine the effect on reactivity
of large voids in the reflector are described. A new
attempt was made to calculate the energy spectrum of
the gamma rays leaking from the BSF reactor by inte-
grating the sources over the entire reactor and applying
an attenuation kernel. Some recent calculations pertain-
ing to the UO\(_2\)-SS Bulk Shielding Reactor II are pre-
sented, as well as a status summary on the project. The
latest design of the Tower Shielding Reactor II (TSR-II),
with its associated controls and 5 Mw water cooling
system, and several studies supporting the design are
presented. The values of \(\eta\) of \(U^{238}\) and \(Pu^{239}\) relative
to the value of \(\eta\) of \(U^{235}\) were determined. Experiments
were performed to determine the critical parameters of
aqueous solutions of 92.2% \(U^{235}\) enriched uranyl fluoride
contained in cylindrical annuli formed by various com-
binations of aluminum cylinders varying in diameter
from 2 to 30 in. Critical experiments were performed
with ORR and BSR fuel elements to determine safe
arrays in which the elements could be handled and
stored. Experiments were performed with 2.0% \(U^{235}\)-
enriched uranium metal plates in a light-water-
moderated and -reflected assembly. The reflecting
properties of water, Styrofoam, and Plexiglas were
studied in critical experiments with 6 in. thick slabs of
aqueous solutions of \(UO_2F_2\) enriched to 93% \(U^{235}\). A
series of experiments with blocks of 2% \(U^{238}\)-enriched
mixtures of \(UF_6\) and paraffin are described. Some data
are presented describing the critical conditions of
spherical volumes of aqueous solutions of \(U^{235}\) and \(U^{238}\).
A study of gamma emission during thermal fission of
\(U^{235}\) was carried out. The slowing down of fission neu-
trons in water was investigated. Results were obtained
for neutron diffusion measurements for beryllium at
room and liquid nitrogen temperatures. A 200-kv parti-
cle accelerator was constructed for the BSR. A detailed
theoretical study was made of resonance absorption of
neutrons in nuclear reactions. The connection between
neutron flux and slowing down density was investigated.
The empirical kernel method for treating bare critical
systems is described and utilized to predict the mater-
ial buckling of aqueous \(U^{235}\) bare reactors in an effort
to clarify the age discrepancy. The infinite medium
multiplication factor was computed for \(\frac{U^{235}}{O_2F_2} + H_2O\)
solutions as a function of neutron energy.

22. 9259 LAMS-2215
Los Alamos Scientific Lab., N. Mex.
NEUTRON CROSS SECTIONS FOR FAST AND INTER-

A neutron transport equation which was solved nu-
merically has been connected with multigroup neutron
cross sections and critical assemblies to display the
area of coverage and range of errors. The cross sec-
tions, critical assemblies, and results are tabulated.
(auth)
group-averaged and digitalized cross sections are defined. The groups over which the cross sections were to be averaged, the averaging procedure, and the digitalization procedure were specified so that the results obtained would be readily usable in the GE reactor calculation procedure. In addition, thermal averages of the neutron diffusion cross sections are reported. (auth)

26.

11384   TNCC(US)-39

[Tripartite Nuclear Cross-Sections Committee.]

REPORTS BEARING ON U235 THERMAL FISSION CROSS-SECTION DISCREPANCY. Mar. 1958. 27p. $4.68(ph), $2.70(mf OTS).

Originally issued as LA-511 and LA-512.

The gamma-ray activity of two samples irradiated in the Clinton and Hanford piles and of the barium and cesium extracted from them was compared with the activities extracted from a sample in which the number of fissions was determined by monitoring during the neutron exposure. The results obtained show good internal consistency and indicate that $(8.9 \pm 0.3) \times 10^{17}$ and $(4.8 \pm 0.25) \times 10^{16}$ fissions, respectively, had occurred in the two samples. These results are compared with mass spectrometer data obtained by Williams and Yuster on the same samples to find the ratio of the capture and fission cross sections of U235. (auth)

27.

934   AERE-R/2620


The fast, epithermal, and thermal neutron flux distributions through the core of LIDO have been measured using absolutely calibrated thin foils. By comparing the reaction rates of sodium, manganese, indium, and gold, absolute values have been inferred for the epithermal and thermal fluxes, and some revised cross sections are suggested for manganese and indium. (auth)

1960

28.

13217   WAPD-BT-17(p.51-4)


MULTIGROUP CROSS SECTIONS FROM DETAILED NUCLEAR DATA. H. J. Amster. 6p.

The cross sections of U238 have recently been tabulated with more than usual detail. They are used here for making up a fifty-four group experimental library for the MUFT
Cross Sections 29-33

Code. The procedure is presented as an example of how one could, in principle, account for fine details in all nuclear data. (auth)

29.
13422

The neutron energy spectrum at the center of the dilute fast core of the coupled fast-thermal reactor ZPR-V was studied by use of fission chambers having electrodes quantitatively electrodeposited with U\textsuperscript{235}, U\textsuperscript{233}, U\textsuperscript{234} and U\textsuperscript{238}. Atomic fission ratios found with these four isotopes determine a four-group neutron energy spectrum which can readily be measured as a function of position in the core by use of suitable drive units. The same fission chamber procedure was used to study the equilibrium neutron energy spectrum in a natural uranium exponential column at Los Alamos. The results of measurements in these two spectra are shown and compared with theoretical predictions. The ZPR-V results are also compared to an analysis of this spectrum made by use of nuclear emulsions for the range 0.2 to 2.5 Mev. (auth)

30.
26188  
CF-60-4-12
Oak Ridge National Lab., Tenn.
CRITICAL EXPERIMENTS FOR REACTOR PHYSICS STUDIES. R. Gwin and D. W. Magnuson. Sept. 16, 1960. 64p. OTS.

The thermal value of \( f \) for U\textsuperscript{233} and U\textsuperscript{238} was determined in a series of experiments involving direct comparison of the critical parameters of unreflected homogeneous aqueous solutions of the two isotopes. Auxiliary experiments establishing limits of error, testing certain aspects of the theoretical model employed, and experimentally determining the parameters in the critical equation were performed. Experiments performed with 27-in.-diameter and 48-in.-diameter spheres and 5-ft-diameter and 9-ft-diameter cylinders yielded consistent values of \( \eta \). Measurements of the nonleakage probability in cylindrical geometry gave values consistent with those predicted by a two-group model in which the theoretical value of the age was used. Within the experimental error no differences were found in the ages of fission neutrons for U\textsuperscript{233} and U\textsuperscript{238}. The average thermal values of \( \eta \) for these isotopes were \( 2.284 \pm 0.018 \) for U\textsuperscript{233} and \( 2.074 \pm 0.018 \) for U\textsuperscript{238}. The 2200 m/sec values are the same since the g-factors for eta are unity. (auth)

31.
18428  
BNL-607
Brookhaven National Lab., Upton, N. Y.
NEUTRON CROSS SECTION EVALUATION GROUP NEWSLETTER NO. 1, JUNE 1960. Rudolph Sherr and Sophie Moore. 8p. OTS.

The discrepancy in the Be\textsuperscript{9}(n,2n) cross section as measured by Fischer and Levin & Cranberg was partially resolved in the 2.5- to 4.1-Mev region by new measurements of the nonelastic neutron scattering. The cross section for differential elastic neutron scattering for Be\textsuperscript{9} was determined at 2.5 to 6.0 Mev, together with that for Be\textsuperscript{9}(n,\alpha) at 3.9 to 8.6 Mev. \( B \textsuperscript{10} \) was determined to be present in natural B to the extent of \( 19.8 \pm 0.1 \% \), and the cross sections for \( B \textsuperscript{10}(n,\alpha) \) and natural B (absorption) were found to be 3840 \( \pm 10 \) and 782 \( \pm 3 \) barns, respectively. The cross sections for \( B \textsuperscript{14}(n,\alpha) \) and \( B \textsuperscript{10}(n,2\alpha) \) were determined in the Mev range. Cross sections are reported for \( O \textsuperscript{16}(n,\alpha) \) at 5 to 7.3\% Mev, \( O \textsuperscript{16} \) differential elastic neutron scattering at 3.0 and 8.0 Mev, \( A \textsuperscript{235}(n,\alpha) \) at 1.3 to 5.5 Mev, and \( A \textsuperscript{235}(n,\alpha) \) at 5.75 to 8.94 Mev. Neutron capture cross sections in the keV energy range are reported for Nb, Mo, Rb, Pd, Ag, Cd, In, Sn, W, Pt, and Au. Thermal neutron total cross sections at 0.02 to 0.20 ev are reported for \( U \textsuperscript{233}, U \textsuperscript{234}, U \textsuperscript{235}, \) and \( U \textsuperscript{238} \), with the results for 0.02 to 0.04 ev being given at 2200 m/sec. Neutron multiplication data are given for \( U \textsuperscript{233} \) and \( U \textsuperscript{235} \). (D.L.C.)

1961

32.
22862  

The multiplication of 14-Mev neutrons in uranium shells was studied. The measurements lead to a value of \( \eta \), the average number of neutrons produced per inelastic collision of a 14 Mev neutron, of 3.30 \( \pm 0.15 \). Neutron interactions in a thick uranium shell have also been investigated and effective cross sections for the reactions \( U \textsuperscript{233}(n,f), U \textsuperscript{234}(n,f) \), and \( U \textsuperscript{235}(n,\gamma) \) have been obtained for the inelastic neutron spectrum in the shell. The value of \( \eta \), combined with other nuclear data, leads to the following cross section data for \( U \textsuperscript{233} \) for neutrons in the energy range 13.4 to 14.8 Mev: 1.1 \( \pm 0.2 \) barn and 0 \( \pm 0.2 \) barn. (auth)

33.
24295  

The value for the capture-to-fission ratio \( \alpha \) for \( U \textsuperscript{235} \) for intermediate-energy neutrons was determined by analysis of depleted uranium fuel from a Be-moderated and -reflected lattice. The radiochemical method of analysis is outlined. The results are compared with the 1947 KAPL results and with the work of S. Oleska. (D.L.C.)
34.

6766 (ORNL-3016(p.81)) DETERMINATION OF THE THERMAL VALUE OF $\eta$ OF $^{233}$U AND $^{235}$U BY DIRECT COMPARISON OF CRITICAL PARAMETERS OF AQUEOUS SOLUTIONS. R. Gwin and D. W. Magnuson (Oak Ridge National Lab., Tenn.).

An experimental determination of the thermal value of $\eta$ of $^{233}$U and $^{235}$U, based on the direct comparison of the critical parameters of dilute aqueous solutions of the uranyl nitrates of $^{233}$U and $^{235}$U, was completed. The average thermal values of $\eta$ obtained are $2.284 \pm 0.015$ for $^{233}$U and $2.074 \pm 0.015$ for $^{235}$U. The 2200-m/sec values are the same, since the $g$ factor for $\eta$ is unity in each case. (auth)

35.


The absolute thermal value of $\eta$ for $^{235}$U and $^{235}$U was measured directly by a method of total absorption which involves relative counting of manganese bath activations and some minor corrections. A thermal neutron beam (defined by cadmium difference) is introduced in the center of a one-meter-diameter sphere filled with a dilute solution of manganese sulfate in water. The beam is first made to activate the bath directly, then it is totally absorbed in the fissionable sample whose fission neutrons then activate the bath. The ratio of the two activities is equal to $\eta$ except for small corrections. The results obtained for $\eta$ corrected to 2200 m/sec were, for $^{235}$U, $2.296 \pm 0.010$; and for $^{233}$U, $2.077 \pm 0.010$. (auth)

36.


Capture-to-fission ratios in $^{233}$U, $^{233}$Pu, and $^{239}$Pu are given as a function of neutron energies from 0 kev to 1 Mev. Excitation functions of fast-neutron-induced reactions are given for $^{234}$I(n,2n)$^{234}$I from 8.4 to 15.1 Mev. Reduced neutron widths, $\rho_r$, are tabulated for calcium from 254 to 595 kev and lead-208 from 352 to 718 kev, as determined from the neutron total cross section resonance parameters. Measurements of neutron radiative cross sections for 45 nuclides are tabulated relative to indium at 30, 65, and 167 kev. (For preceding newsletter see BNL-654.) (B.O.G.)

1962


The $^{235}$U(n,p)Co$^{35}$ reaction cross section in a fission spectrum was re-measured and found to be 102 $\pm$ 3 millibarns. (auth)

38.


The correlation of integral experiments and high-energy cross sections is discussed. The importance of integral data where cross-section measurements are inadequate is pointed out. The sensitivity of estimates of fast fission of $^{235}$U to inelastic cross sections and energy degradation in the Mev energy range is shown by comparison of integral data with Monte Carlo calculations. It is shown that the Snell experiment is a sensitive index to the absolute values of inelastic cross sections above 1.4 Mev. The results of attempts by the Brookhaven Cross-Section Evaluation Group to reconcile measurements of inelastic cross sections of $^{233}$U are given. Other areas where integral data and critical experiments can be used to reduce computational uncertainties are the fast effect in beryllium, and $\eta$ of $^{235}$U at intermediate energies. Critical experiments can reduce the present uncertainty in Be (n,2n) cross-sections and in intermediate energy values of $\eta_{23}$. (auth)

39.


The Yiftah, Okrent, and Moldauer cross-section set (ANL Set 135) was modified to include new measurements of $\nu$ and $\alpha$ for $^{235}$U, and to include, in an approximate manner, Hummel, Rago, and Meneghetti's corrections for flux depression at resonance energies in Al and stainless steel. This modified set, ANL Set 635, was used to compute values of $k$ for 22 ZPR-III assemblies of widely varying composition. The DSN neutron transport code was used in spherical geometry and the $S_4$ approximation; shape factors were used to convert from cylindrical to spherical geometry. Seventeen of the calculated values of $k$ lie within $\pm 1\%$ of a mean value of 1.003, and the remaining 3 lie within $\pm 2\%$. In terms of prediction of critical
Cross Sections 40-45

mass, it appears that the procedure used here can achieve an accuracy of 5% to 10% for a wide range of U\textsuperscript{233}-fueled assemblies. (Auth)

40.

3862 (ORNL-3193) THE MEASUREMENT OF \( \alpha \) AS A FUNCTION OF ENERGY. G. deSaussure, L. W. Weston, et al. (Oak Ridge National Lab., Tenn.).

Experiments leading toward a precise measurement of \( \alpha \), the capture cross section to fission cross section ratio, as a function of neutron energy are completed. Determinations of \( \alpha \) are discussed at 30 and 65 kev, and over the range from 1 ev to 150 kev for the principal fissile isotopes. A fission chamber is designed and tested, and various problems involving background radiation and counting techniques are investigated. Electronic circuitry is also designed, built, and tested. (Auth)

41.


The ratio of neutron capture to fission cross sections was measured for U\textsuperscript{233}, U\textsuperscript{235}, and Pu\textsuperscript{239} at 9 incident neutron energies from 30 to 1000 kev. A pulsed and collimated neutron beam is passed through the target at the center of a large cadmium-loaded liquid scintillator. Captures and fissions are detected from their prompt \( \gamma \); scattering is biased out since corresponding prompt pulses are small. Fission neutrons are thermalized in the scintillating solution and provide delayed pulses that identify a fission event. Capture events are not followed by delayed pulses from the scintillator. Corrections are applied for the fissions not followed by delayed neutron pulses and for the effect of background. This experiment yields values of \( \alpha \) to an accuracy of 1 or 2%. (Auth)

42.


The thermal value of \( \eta \) for U\textsuperscript{233} and U\textsuperscript{235} is determined in experiments on unreflected homogeneous aqueous solutions of the two isotopes. These experiments also yield a value for the neutron age and the limiting concentrations of the fissile isotope in the aqueous solutions for infinite volumes. Auxiliary experiments, establishing limits of error, testing certain aspects of the theoretical model employed, and experimentally determining the parameters in the critical equation, are also performed. Experiments performed with 27-in.- and 48-in.-diam spheres, and 5-ft- and 9-ft-diam cylinders yield consistent values of \( \eta \). Measurements of the nonleakage probability in cylindrical geometry give values consistent with those predicted by a two-group model in which the theoretical value of the age is used. Within the experimental error no differences are found in the ages of fission neutrons for U\textsuperscript{233} and U\textsuperscript{235}. The average thermal values of \( \eta \) determined are 2.292 \pm 0.015 for U\textsuperscript{233} and 2.076 \pm 0.015 for U\textsuperscript{235}. The 2200 meters/sec values are the same since the \( g \)-factors for \( \eta \) are unity. The value of the neutron age to the indium resonance energy for U\textsuperscript{235} fission neutrons in water is found to be 25.6 \pm 1.3 cm\(^2\). The minimum U\textsuperscript{233} and U\textsuperscript{235} critical densities for these nitrate solutions are found to be 11.25 \pm 0.10 and 12.30 \pm 0.10 gm/liter for U\textsuperscript{233} and U\textsuperscript{235}, respectively. (Auth)

43.

12501 DETERMINATION OF \( \eta \) BY COMPARISON OF \( \eta \) FOR U\textsuperscript{232} AND Pu\textsuperscript{239} WITH \( \eta \) FOR U\textsuperscript{235} IN A FLUX TRAP CRITICAL ASSEMBLY. R. Gwin and D. W. Maguson (Oak Ridge National Lab., Tenn.). Nuclear Sci. and Eng., 12: 359-63 (Mar. 1962).

The values of \( \eta \) for U\textsuperscript{232} and Pu\textsuperscript{239} were determined by a reactivity measurement. An aqueous solution of each isotope was introduced axially into a critical cylindrical annular flux trap reactor, and the resulting reactivity change was measured by period determinations. From these data the ratios \( \frac{\eta (U\textsuperscript{232})}{\eta (U\textsuperscript{235})} \) and \( \frac{\eta (Pu\textsuperscript{239})}{\eta (U\textsuperscript{235})} \) were obtained. Using measured values of \( \eta \) for U\textsuperscript{235} and the absorption cross sections in this ratio, the thermal values of 2.317 \pm 0.040 for \( \eta \) of U\textsuperscript{233} and 2.032 \pm 0.053 for \( \eta \) of Pu\textsuperscript{239} were obtained. Correction to a neutron velocity of 2200 meters/sec by using the appropriate \( g \)-factor gives a value of 2.317 \pm 0.040 for \( \eta \) of U\textsuperscript{232} and 2.082 \pm 0.054 for \( \eta \) of Pu\textsuperscript{239}. (Auth)

44.


Six-group neutron cross sections are listed for the more common fissile isotopes for study of fast neutron critical assemblies and sixteen group cross sections of the more common reactor material for study of intermediate neutron critical assemblies. Data sources and averaging scheme used for the development of these multigroup parameters are also given. (Auth)

45.

12498 NEUTRON CAPTURE TO FISSION RATIOS IN U\textsuperscript{233}, U\textsuperscript{235}, Pu\textsuperscript{239}. J. C. Hopkins and B. C. Diven (Los Alamos Scientific Lab., N. Mex.). Nuclear Sci. and Eng., 12: 189-77 (Feb. 1962).

The ratio of neutron capture to fission cross sections, \( \alpha \), was measured for U\textsuperscript{233}, U\textsuperscript{235}, and Pu\textsuperscript{239} at 9 incident neutron energies from 30 to 1000 kev. A pulsed and collimated neutron beam is passed through a target placed at the center of a large, cadmium-loaded, liquid scintill-
Capture and fission events are detected by means of their prompt gamma rays; elastic and inelastic scattering events are discarded because of their smaller pulse height. Fission is identified by the delayed pulses produced by capture in the scintillator of the fission neutrons. Corrections are applied for the fission events not followed by delayed neutron pulses and for the effect of background counts. This procedure yields values of 1 + α to an accuracy of 1 or 2%. (auth)


The ratio of neutron capture to fission cross sections, α, was measured for 235U, 239U, and 241Pu at 50 to 1000 keV. A pulsed and collimated neutron beam is passed through the target at the center of a large cadmium-loaded liquid scintillator. Captures and fissions are detected by their prompt gamma rays; elastic and inelastic scattering events are ignored because of their smaller pulse heights. Fission events are identified by the neutrons that accompany the fission process. These neutrons are thermalized in the scintillating solution and provide delayed pulses which characterize a fission event. Capture events are not followed by delayed pulses from the scintillator. Corrections are applied for the fission events not followed by delayed-neutron pulses and for the effect of background counts. The capture, capture-plus-fission, and background events are recorded simultaneously. This procedure yields values of 1 + α to an accuracy of 1 or 2%. (auth)


Critical masses of representative ZPR-III fast assemblies containing resonance scatterers are calculated using the SILENE transport code. The IBM-704 ELMOE code of Hummel and Rago was used to evaluate group transport and group elastic-transfer cross sections in the core in conjunction with the 16-group cross section set of Yiftah, Okrent, and Moldauer. By using hundreds of neutron energy groups and the detailed elastic scattering matrices for the resonance scatterers, ELMOE carries out a fundamental mode analysis. It thereby obtains material buckling and the detailed fine structure flux dependence upon energy. It then re-evaluates the group cross sections for transport and elastic transfer. Consideration of the resonance scattering effects caused by aluminum and stainless steel increase the calculated critical masses of ZPR-III assemblies 23, 31, and 32 by about 15 kg, 21 kg, and 21 kg, respectively. Corresponding reactivity decreases are about -1%, -0.7%, and -2% \( \kappa_{\text{eff}} \), respectively. (auth)


The sensitivity of the calculated critical masses of simple systems to changes in the basic neutron scattering data were investigated. The systems considered are spheres of 29% 235U and 93.7% 239Pu, both bare, and reflected by thick natural uranium. The calculations were carried out using the Carlson SM method with 4 energy groups, and the percentage changes in the calculated critical masses of the different systems, due to specified changes in the aspects of the neutron scattering data, were obtained. The results are presented and discussed with particular reference to the adjustment of the basic data to agree with experimental critical sizes. (auth)

(EUHAEC-390)

The effective resonance integral was measured for \(^{235}\text{U}\) metal and oxide rods of various diameters. Three different experimental methods — activation, static reactivity, and oscillation — were used, and the following consistent results were obtained: \( ^{235}\text{U} \), \( \mathbf{R}_1 = 2.4 + 2.6 \times 10^{-3} \) and \( ^{238}\text{O} \), \( \mathbf{R}_1 = 0.8 + 2.8 \times 10^{-3} \). The Doppler broadening effect was measured to 6000°C for \(^{235}\text{U}\) and up to 1000°C for \(^{232}\text{O} \). The results are in good agreement with the previous measurements of Hellstrand. Measurements of the Dancoff effect for both \(^{235}\text{U}\) and \(^{238}\text{O} \)-moderated lattices are consistent with the Bell approximation. (auth)


The potential of plutonium as a fuel in near-thermal converter reactors is investigated. Over certain ranges of fuel loading and/or moderation, it is shown that the effective absorption cross section (averaged over the entire neutron spectrum) of Pu\(^{240}\) decreases with fuel burnup; i.e., decreases with the associated softening of the neutron spectrum. The plutonium, therefore, behaves as a self-stabilizing or self-compensating fuel with the decrease in Pu\(^{240}\) cross section balancing fissionable material burnup and fission product buildup. Thereby long core lives are attainable with nominal shim control requirements. The
strong neutron temperature dependence of the effective Pu\textsuperscript{239} absorption cross section also results in a highly negative temperature coefficient of reactivity and thereby in the feasibility of spectral shift shim control. Economics evaluation indicates that fuel cycle costs of between 1.5 and 2.5 mills/kW-hr may be attainable with these plutonium fueled systems. (auth)

53.


Alpha, the ratio of neutron capture to fission, for Pu\textsuperscript{239} was measured at seven different points along a flux scanning tube of the PLUTO reactor. Highly-enriched Pu\textsuperscript{239} samples were also irradiated to make allowance for the destruction of this nuclide. Experimental values of \( \alpha \) for Pu\textsuperscript{239} varied from 0.458 to 0.470. Correcting all results to 2200 m/sec neutrons gave a mean value for \( \alpha_{9} \) of 0.370 ± 0.008, a result that is in excellent agreement with present recommended values, but does not include uncertainties in the \( g \) and \( s \) values for neutron capture and fission. (auth)

54.


A critical comparison of measured and calculated central fission ratios for 18 ZPR-III fast reactor assemblies was made with the object of examining the accuracy of computation of spectra and of the fission cross sections used. This comparison uses fission ratios measured with Kink absolute fission chambers and computed with ANL cross section Set 635. The Kink chambers and experimental technique are described. It is shown that ratios measured with threshold detectors must be corrected for the effects of inelastic scattering in the chamber walls. Possible sources of error in the experimental technique are discussed, and experimental evidence for the validity of the method is presented. The derivation of ANL Set 635 is described. It is shown that Set 635 is a modification of the Yiftah, Okrent, and Moldauer ANL Set 135 and that the central spectra and fission ratios calculated with the two sets are generally similar. The \( \phi_{1}^\text{c} \) and \( \phi_{2}^\text{c} \) fission cross sections are not given in Sets 135 and 635; these were taken from ANL Set 179. The measured and calculated fission ratios obtained with \( \phi_{1}^\text{c} \), \( \phi_{2}^\text{c} \), \( \phi_{3}^\text{c} \), \( \phi_{4}^\text{c} \), \( \phi_{5}^\text{c} \), and \( \phi_{6}^\text{c} \) were compared, both to search for trends which might occur with progressive changes in spectra, and also to determine the accuracy of prediction of ratios. It was found that the calculated relative fission rates of Pu\textsuperscript{239} and Pu\textsuperscript{240} are within about ±1\% of the measured values, and the calculated rates for Pu\textsuperscript{241}, Pu\textsuperscript{242}, and Pu\textsuperscript{243} relative to either Pu\textsuperscript{239} or Pu\textsuperscript{232} are within ±3\% to ±5\% of the measured values. However, calculated fission rates for Pu\textsuperscript{241} and Pu\textsuperscript{242} are about 6\% low and 8\% high, respectively, relative to those of the other five isotopes. (auth)

55.

12773 (ORNL-3360(p.51-63)) THE MEASUREMENT OF \( \alpha \) AS A FUNCTION OF ENERGY. G. deSaussure, L. W. Weston, et al. (Oak Ridge National Lab., Tenn.).

The ratio, \( \alpha \), of the capture cross section to the fission...
cross section of $^{235}$U was measured at neutron energies of 30 and 65 kev. The results, $\alpha = 0.372 \pm 0.026$ at 30 $\pm$ 8 kev and $\alpha = 0.315 \pm 0.06$ at 64 $\pm$ 20 kev, are in agreement with previously measured values. The capture cross section of $^{238}$U was also determined — $0.331$ barn at 30 kev and $0.340$ barn at 64 kev, values which are in agreement with published data. (auth)


The nuclear characteristics of gadolinium in septa form were predicted and compared to experiment in three critical assemblies. These cores were designed to provide different spectral environments for sheets of 4.3 wt % gadolinium in zirconium. Two diffusion theory models were examined. A variational method predicted the $k_{eff}$ to within $+0.3\%$ and $-1.4\%$, whereas an iterative treatment was $1\%$ to $2\%$ low. (auth)

57. 19569 (HW-75007(Paper 14)) THE PHYSICS OF PLUTONIUM IN FAST REACTORS. D. Okrent (Argonne National Lab., Ill.) and F. W. Thalgott (Argonne National Lab., Idaho Falls, Idaho). 33p

A review of the Pu behavior physics associated with medium and large fast reactors is presented. The relative reactivity worth and breeding potential of the various Pu isotopes are deduced from examination of pertinent cross sections. This information is used as a basis for examination of a broad range of fast reactors. Other discussion is concerned with Doppler effect, Na reactivity coefficient, selected kinetics, and reactor safety. (J.R.D.)


Nine and sixteen group cross sections for use in transport theory codes are listed. Elements included are H, Be, B, C, O, Al, Ti, Cr, Mn, Fe, Ni, Co, Y, Zr, Nb, Mo, Eu, Gd, Ta, W, Re, Th, U, and Pu. A discussion of the format and the method used to process these is included. (auth)

Cross Sections 56-61

1964


Multigroup constants for fast and intermediate neutrons are compiled, and new data on neutron interactions with nuclei are included. The principle of utilizing multigroup constants, selection of energy groups, averaging of macroscopic cross sections, averaging effective cross sections, considerations of resonance effects, and determination of moderator cross section are discussed. A review is given on fission cross sections mean numbers of secondary neutrons, spectra of fission neutrons, capture cross sections and elastic and inelastic scattering. Tables of group constants and fission spectra for reactor materials are included. 578 references. (R.V.J.)


Developments in methods used to study the physics of fast reactors are reviewed. The status of experimentally obtained cross section data and of cross section theory is included. Some representative values and the methods used to obtain multigroup constants based on the latest microscopic data are presented and described. Experimental results for a representative set of ZPR-III assemblies were compared with calculations based on these latest multigroup constants. The sodium void coefficient and the Doppler effect are discussed. (M.C.G.)


The materials bucklings of a series of plutonium-aluminum alloy-fueled, graphite-moderated assemblies were measured, the carbon/plutonium ratio ranging from 14,520 to 2420. Fission ratios of $^{235}$Pu, $^{233}$U, and $^{235}$U in the assemblies were obtained using small fission chambers. On one of the systems, experiments were extended up to a temperature of 370°C. The experimental data were compared with the predictions of a 43-group diffusion-theory code. Agreement between predicted and measured
Cross Sections 62-67

bucklings is good over the whole range of composition, within the limits of experimental error, which are equivalent to an uncertainty of about 2 per cent in reactivity. The spectrum-sensitive $^{239}$Pu/$^{235}$U fission ratios were 6 to 9 percent higher than those calculated from the multigroup spectra. Comparison with further results from an assembly loaded with $^{233}$U in similar configuration, leads to a ratio of the 2200 m/sec $n$-values of $^{239}$Pu and $^{235}$U of 1.02 ± 0.02. (auth)

62.


The ratio ($\alpha$) of the neutron capture to the neutron fission cross section for $^{238}$U was measured for neutron energies from 4 ev to 2 kev using a pulsed neutron source. A multiplate fission chamber was placed in the center of a large liquid scintillator. The gamma rays emitted by the II of this chamber, following a neutron absorption, were detected with high efficiency by the scintillator; fission fragments were detected by the fission chamber. After various background and efficiency corrections, $\alpha$ was obtained as the ratio of the count rate of the scintillator in coincidence with the fission chamber to the count rate of the scintillator in coincidence with the fission chamber. The $\alpha$'s obtained were used to compute $\eta$ from 4 to 24 ev, in good agreement with direct measurements. (auth)

63.


Multigroup (18-group) diffusion equation cross sections are rederived using updated numerical techniques and recent basic cross-section measurements. The basic series of Spade (BeO—moderated) and Songy (graphite—moderated) critical assemblies are calculated using these group constants, and the results are compared with those from earlier compilations. (auth)

64.


The accurate integral measurement of $\alpha$, the capture-to-fission ratio, for $^{235}$U was experimentally determined for samples containing only 0.095 ppm $^{238}$U. (R.E.U.)

65.

16582  DISTRIBUTION OF THE RATIO OF CAPTURE TO FISSION CROSS SECTION FOR $^{239}$Pu ACCORDING TO HEIGHT IN THE BR-5 REACTOR. V. I. Ivanov, N. N. Krot, and G. N. Smirenkin. At. Energ. (USSR), 16: 497-500(June 1964). (In Russian)

A study was made of the distribution of the capture to fission ratio $\alpha_{c}/\alpha_{f}$, for $^{239}$Pu according to height in the BR-5. The distribution of capture reactions was measured by determining $\alpha_{c}$ as a function of the spontaneous fission rate in plutonium samples irradiated by an integral flux of $10^{14}$ to $10^{15}$ n/cm$^{2}$. Almost isotopically pure $^{239}$Pu ($45 \times 10^{-5}$) was used as the initial material. Values of $\beta_{c}$ as a function of increasing distance from the reactor center increased from 0.1 to 0.8. Data corresponding to an equilibrium neutron spectrum in the active zone and in an external region of the reflector agree with results of measurements of $\alpha_{c}$ and $\alpha_{f}$ for monoenergetic neutrons. (tr-auth)

66.


The average total neutron cross sections for Na, Mo, Rh, Ag, Cd, and In are measured at 10 to 100 keV, and the values obtained are used to calculate the optical-model parameters of the nuclei. The neutron energies used fall in the region of the P-wave giant resonances in these nuclei. (T.F.H.)

67.


Methods are presented for the measurement of $\rho^{18}$ and $\rho^{17}$. The methods for $\rho^{18}$, the ratio of epicalcium to subcadmium capture of neutrons in $^{238}$U, is based on the chemical separation of neptunium from uranium and fission products in two fuel samples, one irradiated in the lattice under study and one in a pure thermal neutron flux. The irradiations are normalized by means of auxiliary dysprosium foils. A description of the separation procedure and measuring technique is presented together with some experimental values found for D$_{2}$O-modified lattices studied in the NORA zero-power reactor. The method for $\rho^{17}$, the ratio of the fission rate in $^{239}$U to that in $^{235}$U, is based on the chemical separation of $^{18}$O from uranium foils of different enrichment, irradiated in the lattice under investigation. A description of the irradiation and separation procedure is presented together with experimental results obtained in D$_{2}$O-modified lattices studied in JEEPNIK, a miniature exponential facility. (auth)

An experimental study was made of some problems of fast-neutron reactor physics. The studies continue a series of experiments on fast reactor physics that has been carried out at the Physical-Energy Institute since 1950. The following problems are considered: refinement and development of multigroup cross-section sets for nuclear reactors calculation; studies of neutron propagation in large blocks of material performed on BR-1 reactor are described; the BFS reactor for large fast-neutron critical assembly investigations; and radiative-capture cross-section measurements on fissionable isotopes for neutron spectra of the BR-5 core and reflector. (auth)


As part of a general program of work on Intermediate reactor assemblies, a series of measurements were made in a subcritical plutonium-fueled, graphite moderated system in which S, the ratio of moderator to fissile atoms, was 920, and in a 1H3f fueled system in which S was 440, to determine the flux fine-structure, the material buckling, and the relative 1H3f/Pu239 reaction rates. Predictions of material buckling and the shape of the neutron energy specrum were made using a Parks scattering kernel, a Sheffield scattering kernel, and a Free Gas Model together with what is basically OCUSOL. (See ANL 5800) group-averaged nuclear data, and also with the point tabulated data from the Winfrith Nuclear Data File. Unfortunately, the latter does not give a good representation of the cross section between 4 and 60 ev. The 1H3f/Pu239 relative reaction rates predicted by the Parks model using OCUSOL group-averaged data are in good agreement with the measured values in both assemblies although the values predicted by the other models using this data are not very different. With the Winfrith Nuclear Data File, discrepancies of as much as 20\% arise between predicted and observed values of this reaction rate ratio. With OCUSOL data, satisfactory agreement is obtained between predicted and observed values of Bn for the 1H3f fueled assembly, but for the 1H3f fueled assembly better agreement was found when the n value for the 1H3f was increased by 25\% at energies above 10 ev. For both assemblies using the Winfrith Nuclear Data File leads to values of Bn up to 6\% lower than those predicted using OCUSOL data. (auth)


The capture-to-fission cross section ratio, \( \alpha \), for 1H3f was determined for two samples of 1H3f-Al alloy irradiated to 2.03 and 2.26 \times 10^{21} \text{neutrons/cm}^2 in the NRU reactor. The changes in isotopic and total uranium content with irradiation were determined by chemical and mass spectrometric analysis. The value of \( \alpha \) obtained from these measurements is 0.1718 \pm 0.0006, which is 2.5\% lower than the weighted mean of other recent measurements. (auth)


An exponential assembly is described, and buckling and diffusion length measurements with it are reported. The device consists of a graphite cube 240 cm on a side with 100 fuel channels 60 x 60 mm, lattice constant 240 mm, and 25 measurement channels 30 x 30 mm. The cube is covered with cadmium foil. There is 2088 kg of natural U in rods 27.2 x 300 mm in a can 1-mm thick. The value of L obtained is 45.4 cm. From the buckling experiment the dimension of a cubic critical reactor is 734 \pm 14 cm. (T.R.H.)


The pulsed-neutron technique was used in different critical experiments. This technique is particularly interesting since it can be used without altering the reactor structures. Its application requires only a time selector and pulsed sources which, because of their low dosrty, are introduced on a small scale into the reactor or its reflector. The technique can be used to measure the neutron lifetime and calibrate control rods quickly and in perfectly safe conditions. Two different Rubbele cores were studied in a critical experiment with a beryllium-oxide moderator and slightly enriched uranium-oxide fuel, and the negative importance function was measured in different arrays: (a) reduction in the volume of the nonreflected core; (b) introduction of cadmium control rods; (c) unloading of peripheral fuel; and (d) fuel replacement. Other experiments reported (Aite: light water and uranium enriched to 90\%; Aquilon: heavy water and natural uranium; Marius: graphite and natural uranium) show that the technique is applicable to natural-
Cross Sections 73-77

uranium reactors. Using a nonportable neutron generator, the same technique was employed to measure graphite diffusion parameters. The buckling range was between 7 m⁻² and 155 m⁻² and the following results were obtained: diffusion coefficient \( D_0 = (2.12 \pm 0.03) \times 10^6 \) cm² sec⁻¹, and graphite anisotropy \( D_4/D_0 = 1.017 \pm 0.008 \). These results are compared with those obtained by others. (auth)

73.


Starting with the Boltzmann equation in the diffusion approximation, the scattering by hydrogen was treated separately from the scattering by all other nuclei, which were considered as "heavy" for this purpose. The resulting equations were then put into the multigroup (in lethargy) form including inelastic scattering and using the fission spectrum as the source. The final results treated hydrogenous media in either the \( P_1 \) approximation or in a "transport" approximation. The connection with the "moment method" was developed. The criticality problem is discussed formally using the matrix method. A discussion of the extrapolation length as a lethargy group quantity is presented. Results of fifteen-group calculations are presented in both approximations for the critical radius of an assembly based on a core consisting of a \( \text{U}_3\text{O}_8 \)-water solution with enriched uranium and compared with the experimentally determined dependence of the radius on hydrogen concentration. Similarly in a nine-group calculation the results from the two approximations for the age to 1.4 ev from fission in water-iron mixtures of various compositions are compared with experimental results. Twelve tables giving the group cross sections for several elements, especially hydrogen, are included in the appendix. (1 f't)

74.


By minimizing the \( \chi^2 \) function, improved values are obtained for the parameters of three formulas for nuclear temperatures (the formulas of Weisskopf, Bethe and Lange-Couteur). A comparison of the improved formulas shows that all fit the experimental data equally well and yield lower values for the nuclear temperature. On the basis of the improved Weisskopf formula, new multigroup inelastic cross sections are calculated and used in the calculation of several fast assemblies. The calculated critical masses are generally in slightly better agreement with experiment. An explanation of this effect is given. (auth)

75.


For neutrons above 1 or 2 Mev, inelastic scattering is described to a good approximation by an evaporation formula. It is necessary to find fitting values for the nuclear temperature at various energies and for a wide range of nuclides. The formulas for the nuclear temperature, \( \theta \), are obtained as derivatives of the formulas for the average level spacing, \( D \). Usually, first parameters appearing in the formula for \( D \) are fitted to the experimental data, and then the formula for the nuclear temperature is derived. It is advantageous first to derive the formula for \( \theta \), in which only one parameter is involved. The fit with experiment is done for parameters in the formulas resulting from the three most common formulas for \( D \)—those of Weisskopf, Bethe and Lange-Couteur. This is done by minimizing the \( \chi^2 \) function. Comparison of the improved formulas shows that all describe equally well the experimental data and yield lower values for the nuclear temperature. On the basis of the improved Weisskopf formula, new multigroup inelastic cross sections are derived and used in the calculation of several fast assemblies. The calculated critical masses are generally in slightly better agreement with experiment than previous calculations. (auth)

76.

4832 (EANDC-331525-128), NEUTRON CAPTURE IN \( \text{U}^{238} \) AND THE RATIO OF CAPTURE TO FISSION IN \( \text{U}^{235} \). L. W. Weston, G. de Saussure, and R. Gwin (Oak Ridge National Lab., Tenn.).

Large liquid scintillator tanks have been applied to the measurement of the capture cross section of \( \text{U}^{238} \) at 30 and 64 kev, and the measurement of the ratio of capture to fission in \( \text{U}^{235} \). The ratio of capture to fission in \( \text{U}^{235} \) is studied over a wide neutron energy range (4 ev to 700 kev) by two different methods. (auth)

77.


The ratio of the neutron-capture cross section to the fission cross section (\( \alpha \) for \( \text{U}^{235} \)) was measured for incident neutron energies from 12 to 690 kev by a large gadolinium-loaded liquid scintillator technique. Additional measurements at 30 and 64 kev were made by a method using a liquid scintillator and a fission chamber. The experimental values of \( \alpha \) can be approximately described by a linear decrease from 0.374 at 10 kev to 0.177 at 210 kev, followed by a less rapid linear decrease to 0.095 at 700 kev. The results of these experiments are consistent and in reasonable agreement with other reported values of \( \alpha \) in this energy range. (auth)
Cross Sections 78-84

neutrons emitted in the fission of $^{235}$U by 2.086 and 4.908 MeV incident neutrons were measured with a time-of-flight technique in conjunction with a multi-plate fission chamber. The experimental arrangement is described and possible errors discussed. The fission neutron spectra obtained from the slow neutron fission of $^{235}$U and $^{239}$Pu were also measured. These results and other published data are summarized and compared with the predictions of statistical theory. (auth)

82.


83.


Methods for computation of spectra in light water are available, and it is interesting to carry out at the same time experimental studies of simple media (such as solutions of fissionable salts) which allow quite direct comparisons with computed values. The spectral indices measurements were made with two small fission chambers, one containing deposited plutonium, the other deposited uranium-235. Their response, when neutron spectrum is modified, allows the epithelial part of the flux to be studied. The media studied with these chambers are fissionable solutions of plutonium or 90 percent enriched uranium) which were made critical in bare cylindrical geometry in the ALECTO reactor. If the ratio of the chambers is normalized to unity in a Maxwell spectrum, then the noted variation of the ratio of the counted Pu chamber/U-235 chamber reaches 1.4 in the range of the studied concentrations. (auth)

84.


In order to compare the breeding capabilities of the major nuclear fuels in the spectrum of a fast-breeder reactor, integral measurements were made for the ratio of their capture and fission cross sections in the third loading of the First Experimental Breeder Reactor (EBR-I, Mark III). The capture-to-fission ratio was determined as a function of position in the reactor for $^{235}$U, $^{239}$U, and $^{233}$Pu. In addition, for $^{235}$U the ratio of (n,2n) and fission cross sections was determined. Further, for $^{233}$Pu the following cross-section ratios were determined: $\sigma_c/\sigma_f(\text{U})/\sigma_f(\text{X})$ and $\sigma_c/\sigma_f(\text{Si})/\sigma_f(\text{X})$, where $\sigma_f(\text{X})$ refers to the fission cross sections of $^{235}$U, $^{239}$U, and $^{233}$Pu. The capture-to-fission ratio results for the three primary fissile species were compared with calculations based upon 15-group neutron diffusion theory using two different sets of monoenergetic neutron cross sections, and the agreement is good. The
Cross Sections 85-93

present data show that of the three major fissile species, 235\textsubscript{U} has the highest value of \( \alpha \), the \( n \rightarrow f \) ratio, and hence indicates that there are no gross errors in the assumed average microscopic values of the 235\textsubscript{U} capture cross section and the 235\textsubscript{U} fission cross section. (auth)

85.


An activation technique was developed for the measurement of the ratio of the capture cross section of 235\textsubscript{U} and the fission cross section of 235\textsubscript{U} in zero-energy fast reactors. This work was initiated because of the long-standing discrepancy between calculated values of this ratio and radiochemically measured values. The technique is a direct counting method which does not involve chemical separation in any way. Measurements were made in four ZPR-3 fast reactor assemblies, two with hard spectra and two with soft spectra. In all cases the measured ratio was slightly (4%) higher, on the average, than the calculated value. This is in good contrast with the past radiocentrical measurements in ZPR-3 assemblies, which gave values 16% less than calculation. The present measurements, therefore, support the general correctness of the calculated ratio, and here indicate that there are no gross errors in the assumed average microscopic values of the 235\textsubscript{U} capture cross section and the 235\textsubscript{U} fission cross section. (auth)

86.


The ratio of the neutron capture cross section to the fission cross section, \( \alpha \), for 235\textsubscript{U} was measured for incident neutron energies from 3.25 ev to 1.8 kev. A pulsed and collimated neutron beam was passed through a 235\textsubscript{U} fission chamber placed at the center of a large liquid scintillator, and both capture and fission events in the chamber were detected in the scintillator by means of their prompt gamma rays. A fission event was distinguished from a capture event by a coincidence of the scintillator signal with a signal with a signal from the fission chamber. The values of \( \alpha \) obtained, after various efficiency and background corrections were applied, are in good agreement with data derived from other experiments. (auth)

87.


Self consistent neutron cross sections for the energy range 1 to 15 Mev are tabulated. The cross section for the (n,2n) reaction in the important energy region 2-4 Mev is still not well determined but the shape was adjusted to give agreement with a recent integral measurement. The angular distribution of elastically scattered neutrons, and the energy distribution of neutrons from the (n,2n) reaction are not reviewed. (auth)

88.


89.


90.


The effect of using the approximate Breit-Wigner single-level formula for representing resonances on criticality calculations was studied by considering the fission/absorption ratio in a pure 235\textsubscript{U} fuel element. Absorption cross sections and fission/absorption ratios obtained with the single-level formula were compared with results obtained with a multilevel formula. (D.C.W.)

91.


Experimental and analytical neutron cross sections for 235\textsubscript{U} are tabulated. The data are incorporated into the GAMMA-II slowing-down program and the GATHER-II thermalization program. The neutron energy range studied is from 1 Mev to 15 Mev. The results are also presented in graphical form. (T.F.H.)

92.


A group collapsing code for condensing fast reactor cross section data is described. The code uses the fundamental mode spectrum as a weighting function in computing few-group cross sections. The input cross sections can have up to 26 energy groups with 10 downscattering elements. The collapsed set can contain any number of groups (1 \leq n \leq number in input set). A FORTRAN IV source deck listing, input information, and a sample case are given. (auth)

93.

94.

**3540** CALCULATIONS OF ZPR-III FAST ASSEMBLIES USING A TWENTY-SIX-GROUP CROSS-SECTION SET.

95.


96.

**4484** (AECL-2148) A DETERMINATION OF THE RATIO OF CAPTURE-TO-FISSION CROSS SECTION OF $^{239}$Pu

The neutron capture to fission cross section ratio ($\alpha$) for $^{239}$Pu has been determined for two samples of $^{239}$Pu-Al alloy irradiated to 2.29 and $9.5 \times 10^{13}$ neutrons/cm$^2$ in the NRU reactor. The changes in isotopic composition and total uranium content with irradiation were determined by mass spectrometric and isotopic dilution analyses. The average value of $\alpha$ obtained was $0.0949 \pm 0.0004$ for the irradiation in a neutron spectrum having $r = 0.023 \pm 0.004$ and $T = 40^\circ$C. The $2200 \text{ m/sec}$ value of $\sigma_c = 0.00939 \pm 0.0009$ was derived from the measured $\alpha$. (auth)

97.


The generation of a set of 26-group neutron cross sections for analysis of fast reactors is discussed. The evaluation of cross-section data for individual inelastic level excitation, total inelastic cross sections, fission, nuclear temperature, and capture is outlined. Cross sections for both reactor fuel and structural materials are included. (D.C.W.)

98.

**3374** TWENTY-SIX-GROUP CROSS SECTIONS.

99.

**4555** (TID-2141), pp 50-66) NEUTRON CROSS SECTIONS. (O. Schael, Polytechnic Inst., Troy, N. Y.). The capture-to-fission cross section ratio of $^{235}$U was measured, and preliminary values of the neutron resonance parameters for Hf isotopes were obtained by analysis of total cross section measurements on Hf. Neutron scattering and total cross sections of $^{235}$Pu, $^{237}$Pu, and $^{239}$Pu were measured from 10 to 100 ev. The analysis of the cross section data for enriched samples of $^{235}$Pu, $^{237}$Pu, $^{238}$Pu, and $^{239}$Pu was almost completed; resonances were identified and assigned by isotope's from 100 ev to 4 kev using a computer program. (D.C.W.)

100.


A nineteen group extension to the first fourteen groups of the Yiftah, Okrent and Moldauer cross-section set was produced. The thirty-three groups cover the energy range 0.414 ev to 10 Mev, the highest energy group being of unit lethargy width and the remainder of half-lethargy width. Resonance self-shielding factors at 300K for $^{235}$U, $^{239}$U, $^{238}$Pu and $^{239}$Pu are included. They are cast as a simple formula relating the shielded cross section in each group $g$ to the cross section at infinite dilution via $\sigma_g$, the total cross section per atom, and two parameters, $\alpha_g$ and $\beta_g$. (auth)

101.


Neutron cross sections are tabulated and shown graphically for elements having $Z = 88$ to 98. (T.F.H.)

102.


An Addendum to CRFP-913 and CRFP-1090. New pseudo-fission product yields have been calculated for thermal-neutron fission of U-233, U-235, Pu-239, and Pu-241, and fast-neutron fission of U-238, based on new input cross section and fission yield data. The input values were obtained from a review of published yields and cross sections, including data published since 1960. The effect on reactor poisoning of three $\beta$-active nuclides with rela-
Cross Sections 103-110

tively short half-lives are demonstrated. In the case where these are secondary to a non-saturating primary fission product (Pm-148 and Pm-148m) a simplified decay scheme has been used to include their contribution in the pseudo-fission product calculations. (auth)

103.


From Symposium on the Physics and Chemistry of Fission, Salzburg, Austria.

Level parameters, mean fission cross sections, and radiative capture in neutron reactions with 232Np nuclei at 0.002 to 3 kev were measured by a time-of-flight method using pulsed fast reactor as a resonance neutron source. A 2048-channel analyzer with channel width 32, 16, and 8 µsec was used for recording energy intervals at 0.002 to 0.005, 0.005 to 1,5, and 1.5 to 30 kev, respectively, resulting in a resolution of Δ/λ = 0.04 µsec/m.

(R.V.J.)

104.


105.


See also A/CONF.28/P.310.

Some revisions to the previously published 16-group YOM cross section set are presented. An extension is made from 16 to 20 groups by additions in the 2 to 14 MeV range. Both sets take account of new information on neutron angular distributions, modified inelastic scattering matrices, modified capture cross sections for fissile nuclides, modified ̅, and modifications for errors and omissions. The materials studied are the same as in YOM, namely: 241Pu, 242Pu, 243Pu, 244Pu, 248Pu, 235U, 238U, 232U, 22Na, 22O, 22C, 22N, 23Th, 23Bi, 23Pb, 23Ta, 23Mo, 23Nb, 23Zr, 23Ni, 23Fe, 23Cr, 23V, 23Ti, 23K, 23Al, 23Na, 23O, 23C, and 23B. (R.E.U.)

1966

106.


107.


Additions and modifications to the sixteen group cross sections listed in the first edition of GEMP-173 dated January 18, 1963, are reported. (auth)

108.


Measured values of eta, ̅ and ̅ derived values of ̅ are given for 241Pu from 0.03 to 200 ev. The averages of the cross sections and of eta and alpha have been calculated for a number of energy groups together with various ratios and integrals of these quantities which are of interest for reactor design studies. Comparisons are made between the results obtained in the present experiment and those from other laboratories. (auth)

109.


Highly enriched samples of 241Pu were irradiated in five different positions in a flux scanning tube of the PLUTO re­actor so that ̅ (the ratio of neutron capture to fission) could be measured for this nuclide. Highly enriched samples of 242Pu and 243Pu were also irradiated at the same time so that allowance could be made for the effects of their destruction. Experimental values of ̅ for 241Pu varied from 0.352 to 0.395, with a mean value of 0.371 ± 0.016. Correcting all results to 2200 m/sec neutrons gave a final value of 0.390 ± 0.023. Where a comparison was possible with previously published data, good agreement was found. (auth)

110.


The spectra of real and adjoint fluxes at the sample position at the center of the Sodium Graphite Reactor Critical Assembly (SGR-CA) were calculated with a one-dimensional diffusion code using 16 energy groups. Fifteen­group sample cross sections were calculated for a number of sample materials and sizes. These cross sections, together with the calculated spectra, were used in a pertur­bation theory calculation of sample reactivities and Doppler coefficients. Some sample cross sections were recalculated by using the cross section definition appropriate to unper­turbed fluxes. Perturbation theory calculations made by using these cross sections were in good agreement with most experimental results. Work on Monte Carlo calcula-
tion of resonance integrals and Doppler coefficients has started. Self-shielding of the I/v contribution to the resonance integral and reactivity effects due to sample thermal expansion were estimated with the aid of Wigner's rational approximation to the escape probability for a lump. The indicated thermal expansion effect is quite large for samples not under cadmium. The first set of radial flux maps using tungsten and gold foils was completed. The reactivity of a natural-tungsten slug was measured as a function of temperature up to 120°C, both bare and under cadmium. The measured points lie on a smooth curve with little scatter. Each reactivity is determined with a precision of about 0.002 to 0.004 cents; the maximum reactivity change from room temperature is about 0.12 cents. Auxiliary measurements were carried out that demonstrate the insignificance of the effects of scattering by heavy elements in the sample. Preliminary calibration of the epithermal sensitivity of the oscillator was carried out with gold and uranium samples.

111.

The final 15-group calculation of parameters of the 10.6-in. SGR-CA lattice was completed. Multigroup effective cross sections were calculated for the actual isotopic compositions of the 192W and 194W samples. From these (and previous) cross sections and the multiplet and adjoint fluxes, expected reactivities were computed for samples of natural tungsten, 192W, 194W, and gold. Doppler coefficients were calculated for the three tungsten samples. Agreement of these quantities with experiment is as good as could be expected for absolute reactivity calculations; however, measured reactivity ratios among the three samples are significantly different from calculated ratios. An existing Monte Carlo resonance-escape code was modified so that only those neutron histories resulting in passage through the sample are followed. This relatively minor modification has greatly increased the efficiency of the code. Activation measurements of the 194W resonance integral were completed at two temperatures, 293 and 473°C. Reactivity worths and Doppler coefficients of samples enriched in 192W and in 194W were measured in the cadmium sleeve at the center of SGR-CA. Doppler measurements were made at several temperatures from room temperature to above 1000°C. Reactivities of gold samples with widely varying surface-to-mass ratio were measured to allow absolute calibration of the reactivity measurements.

112.

Cross Sections 111-116

113.

A short description is presented of the assembly start-up, critical experiments in different core configurations and measurement of various operational characteristics. The data obtained during operation period are summarized. Some safety aspects are considered. Present and future programs of experiments are included. (auth)

114.

115.

The published cross sections of 235U, 238U, 239Pu, and 240Pu from 1 kev to 10 Mev were studied to select best cross sections for fast reactor analysis. Emphasis was placed on determining the reference data used and, where necessary, revising the measurements to agree with currently accepted cross sections. Some checks were made by comparing calculations based on the selected cross sections with integral measurements in the fission spectrum and in 19 fast reactor critical assemblies. The study shows a clear need for more work on precise fission section measurements, particularly that of 235U since it is widely used as a reference. (auth)

116.

The cross section of 235Pu for low-energy neutrons was measured between 0.16 ev and 5 kev by the time-of-flight method, with a gas scintillator containing 310 mg of 235Pu. Under the best conditions and at high energy, the resolving power of the measurement was 3 ps/m. The Saclay linear accelerator was used as a pulsed neutron source. The measurement consisted in the successive determination of the fission rate (by means of a BF3 counter whose efficiency was proportional to E^-4). The background was determined by interposing black resonance screens in the neutron beam. The quantity q(E) is proportional to the ratio of the fission rate to the counting rate of the BF3 detector. It was calibrated by reference to the resonance at 7.8 ev, the parameters of which are known. At the discrimination threshold, adjusted to an alpha-ray build-up rate of 1 count/s, an efficiency of 6% for the detection of fissions was obtained. The fission cross section was measured in several sequences to observe the various energy ranges with sufficient resolving power. A detailed description is given of the experimental apparatus, the tests with the
Cross Sections 117-122

scintillator and the experimental conditions in which the various measurement sequences were carried out. A curve showing the fission cross section as a function of energy is included. The results of an analysis of the resonance parameters are also reported. (auth)

117.


The available data for $^{23}Pu$, $^{24}Pu$, $^{25}Pu$, and $^{26}Pu$ were compiled and evaluated for neutron energies between thermal and 15 Mev. The data obtained from this evaluation are presented in the form used in the General Atomic data library. A best set of single level resonance parameters was obtained, over the resolved resonance region for each of the above nuclides, in order that some estimate of the Doppler coefficient of reactivity can be made. (auth)

118.


In an enriched under moderated reactor resonance absorption of epithermal neutrons must be considered. A new set of neutron cross sections of $^{235}U$ are presented. In particular resonances are included in the experimentally unresolved energy range from ~50 ev to 30,000 ev. Resonance self-shielding and Doppler broadening in this range is handled by a picket fence model for the resonance structure. Account is taken of the fact that within this statistical range there are in general two different distributions for each partial width of a resonance corresponding to different spin states of the compound nucleus. In the unresolved region use is made of the cross section measurements reported by Utley and those in BNL-325. At low energies, $E < 1.86$ ev, the data of Shore and Sailor is used and at high energies, $E > 3.18 \times 10^4$ ev, data in the BNL-325 report is used. For the experimentally resolved energy region resonance parameters obtained recently by the Saclay group and also parameters given in the BNL-325 1964 supplement are used; smooth background cross sections due to the negative energy resonance are included. Values for the average number of prompt neutrons per fission are obtained from the empirical formulas based on measurements by Most, Mather and Fieldhouse, and the fission energy spectrum is obtained from the experimental results of Cranberg and Nereson. The variation with energy of the resulting cross sections are shown to compare favorably with various experimental cross sections. Also results of calculations with the GAM, QUERY, and TNS codes are compared with results of integral experiments, viz., the fission integral, the absorption integral, and measured values of $(\phi)$. A comparison is also made of the calculated hydrogen worth with the measured values for a particular experiment. (auth)

119.


The integral $^{10}Be(n,\gamma)$ cross section was determined by using $Be$ and $BeO$ samples irradiated in the core of the Battelle Research Reactor. The value obtained was $60$ mb for neutrons having energies above 2.7 Mev and a fission spectrum. Fast-neutron dosimetry was accomplished using iron, nickel, and titanium threshold detectors. The results are based on measurement of the total amount of helium produced during irradiation. Corrections were made for the small contribution from $(n,\alpha)$ reactions on $^{10}B$ and $^{15}O$. (auth)

120.


A new data library is available for use with the DTF-II $S_n$ theory transport code in computing neutron fluxes in shield systems and reactor criticality. Two group structures are available: a 21-group structure with nine energy groups in the 0.41 to 14.9 Mev range (primarily for use in shielding problems) and a 16-group structure (suitable for use in criticality calculations). The epithermal data for both group structures were obtained by group-averaging pointwise data over a typical SNAP spectrum using the GAM-II code (these calculations were performed by personnel at Oak Ridge National Laboratory). Data for the thermal range were taken from previously generated data libraries at Atomics International. A complete listing of the library data is also included. (auth)

121.


A new data library is available for use with the DTF-II $S_n$ theory transport code in computing neutron fluxes in shield systems and reactor criticality. Two group structures are available: a 21-group structure with nine energy groups in the 0.41 to 14.9 Mev range (primarily for use in shielding problems) and a 16 group structure (suitable for use in criticality calculations). The epithermal data for both group structures were obtained by group-averaging pointwise data over a typical SNAP spectrum using the GAM-II code. Data for the thermal range were taken from previously generated data libraries at Atomics International. The available materials, the generation of the library data, and information for those wishing to use this library in DTF calculations are described. A complete listing of the library data is also included. (auth)

122.


From American Physical Society, Conference of Neutron
Cross Section Technology, Washington, D. C.

The development of accurate computational methods for thermal reactor lattices has reached the point at which neutron cross section uncertainties are a major limitation to the confidence with which theory can be applied to practical situations. This sensitivity to nuclear data is readily seen in the analysis of extremely simple lattice experiments, although it is still present, of course, in calculations for power reactors of greater geometrical complexity where errors in describing spatial effects may effectively hide basic discrepancies. The basic lattice data and computational methods are briefly reviewed and the connection between uncertainties in cross sections and in predicted reactor characteristics is traced for some typical reactor lattices. The effect during reactor life of changes in the cross sections assigned to both fertile and fissile isotopes in the uranium and plutonium chains is described for some light and heavy water moderated reactors. (auth)

123.


A survey of data was made in order to discuss in detail the problem of neutron cross section data from the experimental and theoretical points of view. The data and their phases involved were discussed for the following: reactions leaving the discrete states of final nucleus, reactions leaving the continuous state of final nucleus, $(n,\gamma)$ reactions and resonance phenomena, and nuclear fusion. (M.O.W.)

124.


Group cross sections for 68-group structure of the GAM-1 code for absorption, fission, and $(n,2n)$ processes for the structural solution, and breeding materials of the Mosel Reactor concept are given. The source of the basic data are compiled. (tr-auth)

125.


The preparation of multigroup data, typically 18 groups, for input to codes such as CRAM and DSN for systems in which the low-energy resonances dominate is considered. (J.F.P.)

126.

Cross Sections 131-138

131.  
Group cross sections for aluminum and nickel reflectors, which include effects of energy-dependent leakages into and out of reflector regions, are considered. Leaks are introduced as absorptions in fine-group EMOE calculations. Iterations are made between coarse-group regional fluxes from criticality calculations and the leakage-modified EMOE analyses. (auth)

132.  
The general applicability of computational surveys in establishing criticality limits and identifying optimum criticality conditions for nuclear safety control is briefly reviewed. Recent studies made with I-group theory, 16-group transport theory, and a first-order solid-angle method for interacting arrays include: the infinite-medium multiplication factor, kmin of homogeneous hydrogen-moderated U of all 235U enrichments, critical spacings of arrays of individually subcritical units in air, and the effects of composite metal-water reflectors on the minimum critical cylinder dimensions of U systems enriched to 4.98 wt. % 235U. Good correlation with applicable experimental data is noted. Specific items discussed: the significance of kmin found for high-enrichment systems of undermoderated U, the 235U-enrichment and concentration limits derived from kmin determinations, and the inverse 235U-enrichment phenomena found for both individual units and interacting arrays. (auth)

133.  

134.  
From American Physical Society, Conference of Neutron Cross Section Technology, Washington, D. C.  
Cross sections useful for determining the production of transuranium elements are evaluated using the Breit-Wigner single level formula. Thermal cross sections and resonance integrals for neutron capture and fission are presented for 238U, 239U, 239Nd, 240Pu, 241Pu, 242Pu, 243Pu, 244Pu, 241Am, 242Am, and 243Cm nuclides. (auth)

135.  

136.  
A novel method which makes possible a direct measurement in low-power reactors of the effective capture-to-fission ratio, \( \sigma \), has been developed. It involves a comparison of reactor response to oscillated samples of a fissile material, an absorber, and a spontaneous fission source, augmented by an experimental determination of the respective fission rate, capture rate, and neutron-source strength. These experimental results, combined with the number of neutrons per fission of the fissile material, yield a value for the quantity \( 1 + \sigma \). Applications for this technique are illustrated by the results for measurements with \( \text{Pu}^239 \) in both an undermoderated critical assembly of moderately enriched \( \text{Pu}^239 \) in light water and an epithermal, \( 1/E \) neutron spectrum. (auth)

137.  
From American Physical Society, Conference of Neutron Cross Section Technology, Washington, D. C.  
Methods used to calculate intermediate neutron spectra in fast reactors are subject to inaccuracies in cross section data and as well as approximations used in preparing (group averaging) cross section data for transport codes. Direct measurement of neutron spectra in bulk media provides a means of evaluating discrepancies. The status of spectrum measurements is reviewed, preliminary results of spectrum measurements are discussed, and some new measurement concepts are described. (auth)

138.  
The fast pulsed reactor was used to measure the total cross section, the fission cross section and the radiative capture cross section of \( \text{Pu}^239 \) by the time-of-flight method with a resolution of \( \sim 0.04 \mu \)s. The flight length was 1000 m. The time spectra were recorded by 2048-channel time analyzers. To measure the total cross sections by the transmission method, a resonance-neutron scintillation
counter with lithium glass was used. The fission, radiative-capture, the total cross sections were measured by the auto-indication method using a large liquid scintillation detector with cadmium added to the solution. The sample to be investigated was placed in the axis of a cylindrical opening in the vessel, in a geometry close to 4π. The volume of the detector was scanned by 32 FEU-24 photomultipliers. The fissions were identified by the delayed coincidences between the scintillations corresponding to the recording of the prompt gamma rays and the moderated prompt fission neutrons. Radiative capture is not accompanied by a delayed pulse. The efficiency of recording fissions and radiative capture was ~50 and ~25% respectively, while the background (expressed as a percentage of the strong resonances) was ~1 and ~5%, respectively. The area method was used to obtain the parameters $g_\Gamma$, $\Gamma_\pi$, and $\Gamma_\sigma$ for a number of low-lying levels. The average widths of $\Gamma_\sigma$ and $\Gamma_\pi$ over all the levels were (42 ± 3) meV and (51 ± 6) meV respectively. The levels were divided into two systems with $\alpha_1 < 1/2 (\alpha_1 \sim 0.5$) and $\alpha_2 > 1/2 (\alpha_2 \sim 1.5$) and $\Gamma_\sigma \approx 72$ meV and $\Gamma_\pi \approx 25$ meV respectively, which may be connected with the two spin values 3 and 4 of the compound nucleus. The energy dependence of the ratio of radiative-capture to fission cross sections was obtained in the neutron-energy range from 2 ev to 30 keV. Measurements using a sample of $^{238}$U instead of $^{235}$U made it possible to take into account experimentally the contribution of the radiative capture of $^{238}$U to $\alpha$. (auth)

139.


A knowledge of accurate and reliable nuclear data for the materials in any fuel cycle is essential in assessing its technical feasibility and economic potential. The nuclear parameters, in which a reactor physicist is usually interested, are discussed for the heavy elements associated with the thorium cycle, such as $^{233}$Th, $^{235}$U, $^{239}$Pu, and $^{238}$U and fission products. The existing data for these materials and the gaps that have to be filled up are described. Temperature dependent effective group capture cross sections for thorium, which were evaluated with the available resonance parameters, are also tabulated. (auth)

140.


The absolute value of eta, the number of fission neutrons per absorption, was measured for $^{235}$U, $^{238}$U, and $^{239}$Pu using monochromatic neutrons from the crystal spectrometer at the Materials Testing Reactor. Measurements were made on all three isotopes at 0.025-ev neutron energy and on $^{235}$U and $^{239}$Pu at 0.057 ev. The Bragg beam from Be (002) was passed through a mechanical monochromator to remove higher-order neutrons and yield a truly monochromatic beam. The neutron detector was a manganese sulphate bath, which absorbed in turn the Bragg beam and then the fission neutrons produced when the beam was completely absorbed in a fissionable sample. The ratio of the levels of $^{14}$Mn activities produced in the two types of irradiation yielded the value of eta for the fissionable material of the sample, after the application of a few small corrections. The method of least squares was used to extract the values of eta from the experimental data. (auth)

141.


$^{235}$U neutron cross sections are tabulated for 1156 energy points from 0.0001 ev to 10 MeV, and the sources of the values are given. The file is intended for use with a Fortran program in obtaining group average cross sections for dilute $^{235}$U systems. In addition, 1/E weighted averages for a 68-group set are given. (auth)

142.


The production cross sections $\dot{E}_p = (\nu - 1) L_n$, where $\alpha = d_o / d_v$ for $^{239}$Pu and $^{238}$Pu, were compared in the fast neutron flux at the center of a bare spherical critical assembly of $^{235}$Pu. These quantities averaged by the fast-neutron spectrum indicate nearly the same properties for the even-even $^{238}$Pu nucleus as for the odd-even $^{239}$Pu nucleus. The ratio measured in a neutron flux peaked at 0.25 Mev with an average neutron energy $E = 1.67$ Mev is $\dot{E}_p^{238}$Pu/$\dot{E}_p^{239}$Pu = 1.01 ± 0.06, and $\dot{E}_p^{238}$Pu = 3.76 ± 0.23 b. The results of this study indicate that $^{238}$Pu metal probably has a critical mass of the same order of magnitude as $^{239}$Pu metal. (auth)

143.


From American Physical Society. Conference of Neutron Cross Section Technology, Washington, D.C. Experiments were performed with the KAPL Solid Homogeneous Assembly. The core built on the assembly are leakage dominated and were designed to have an enhanced sensitivity to high energy cross sections. Sensitivity calculations were made using evaluated microscopic cross sections for oxygen, zirconium, carbon, and uranium-235. (M.O.W.)

144.


Measurements were made of the fission cross section of $^{235}$U, $^{238}$U, $^{239}$Pu, and $^{238}$Pu at several neutron energies between 40 and 500 keV. Measurements in this energy range are of importance in reactor calculations especially in fast dilute systems where the neutron flux is
Cross Sections 145-149

high in the 10–100-kev energy range. Recent measurements of the 235U fission cross section gave absolute values slightly lower than previous data. The present series of measurements are made relative to the new values of the 235U fission cross section using back-to-back ionization chambers. The fission foils were assayed by α-activity, direct weighing and coulometry. Good agreement was obtained between these assays. The fission measurements have an estimated accuracy of between 1 and 2% and, combined with the error on the 235U fission cross section, give a final error of about 2% in the fission cross sections. The results together with those of previous measurements are given, and the corrections for fission fragment absorption, backgrounds, and scattering are discussed. (auth)

145.

The capture-to-fission cross section ratio of 235U may be measured by determining the relative intensities of two lines of the γ spectrum of a neutron-irradiated specimen; one of these lines belongs to an isotope formed by capture, such as 235U or 239Pu, the other to the fission product 106Xe. Evaluation of these data requires knowledge of the absolute yields of these lines, of the probability of formation of 106Xe, and of the efficiency of the γ spectroscopy which depends largely on the properties of the detector. Even though all these factors are not known with the desired precision, valuable data were obtained. The ratio may be determined more conveniently by calibrating the specimen before the actual measurement in a thermal flux, assuming that the 106Xe is formed only by the fission of 235U. This method was used in experiments carried out on the BR-1 reactor; an accuracy of 0.5% was reached. The method is considered suitable for determining other reactor parameters, such as the initial breeding ratio, fast-neutron multiplication factor, etc.; it may be used for systems ranging from fast assemblies to thermal reactors. (TTP)

1967

146.
1208 (LA-DC-7962) COMPARISON OF MULTIGROUP CROSS SECTION SETS USED IN REACTOR CALCULATIONS. Baugh, M. E.; LaBarge, R. J. Los Alamos Scientific Labs., Univ. of California, N. Mex. (1965). Contract W-7405-eng-36. 12p. (CONF-651019-1). Dep, mn, CFST $1.00 cy, $0.50 mn.

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

In order to provide a common basis for comparing various neutron multigroup cross-section sets, calculational models of two well defined critical experiments were specified. The critical assemblies chosen for this study were the Jezebel (Pu) assembly and the 610-liter ZPR VI (Assembly No. 2) UC core. The Hansen-Roach 16-group and Russian 26-group cross-section sets were used for the initial comparisons. Calculations were made to determine the critical core volumes and central reactivity worths using the DTF-IV transport code in the S6 approximation and assuming spherical geometry. In addition, two types of perturbation theory calculations were made to arrive at central reactivity worths for small samples. The first of these is used to compute values in terms of simple atomic perturbation cross sections. The second is based on transport perturbation theory, using real and adjoint currents and fluxes, with results expressed in terms of α/γ-atom. Wherever available, experimental data are presented for comparison with calculations. Central reactivity worths have been calculated for the following materials: 239Pu, 240Pu, 241Pu, 242Pu, 243Pu, 244Pu, 245Pu, 246Pu, 247Pu, 248Pu, 249Pu, and 250Pu. In the case of ZPR VI, the reactivity effect of voiding the core Na was also calculated. (auth)

147.

From IAEA Conference on Nuclear Data, Paris.

Use of the mass-spectrometric method for the measurement of the ratio of neutron capture to fission, for the four nuclides 235U, 239Pu, 241Pu, and 243Pu is discussed. In order to obtain useful results as rapidly as practicable, the high neutron fluxes present in the lattice of a reactor are usually employed for the irradiations. Under these conditions the large epithermal contents of the neutron spectra make the computation of the 2200 mev/sec values, from the measured values, less certain. More recently, similar measurements have been made in the thermal column of a reactor for 239Pu and 241Pu in an attempt to reduce the uncertainty from this source. This technique and its limitations are discussed. (S.F.L.)

148.

Mixtures of enriched samples of 239Pu, 241Pu, and 243Pu were irradiated for 205.6 days in the DIDO Reactor near the core. Neutron doses were monitored by cobalt metal wire. Neutron temperature was (118 ± 4)° with an unperturbed Maxwell-Boltzmann distribution. Results are given for mass analyses of the samples, both before and after the irradiation. (S.F.L.)

149.

The published cross sections of 232Th, 233U, 235U, 238U, 239U, 240Pu, 241Pu, 242Pu, and 243Pu from 1 keV to 10 MeV were carefully studied to select the best cross sections for fast reactor analysis. Emphasis was placed on determining the reference data used; and, where necessary, the published data were revised to accord with more accurate, currently accepted cross sections. Thus, it is believed that a consistent set of cross-section data was derived. Some cross checks were made by comparing calculations based on the selected cross sections with integral measurements in broad fast-neutron spectrums. This study shows the great importance of the 235U fission cross section in deriving other cross sections and emphasizes the necessity of re-evaluating nearly all fission cross sections, if it proves necessary to revise the 235U data. (auth)
4323 DIRECT AND ABSOLUTE MEASUREMENTS OF AVERAGE FISSION NEUTRON YIELD FROM $^{235}$U AND $^{239}$Pu.


From IAEA Conference on Nuclear Data, Paris.

A recent survey by the IAEA has shown that significant discrepancies exist among absolute values of $P$ and $\sigma_i$ for $P(235)$ only one measurement, made in 1958 with a quoted precision of 1.5%, is considered independent, and the Westcott group [Atomic Energy Rev. 3: 1965 reports that the original value has been lowered so that it is now 2.5% below the recommended least-squares average. There are four accepted values for $P(235)$, one of which is also about 2.5% lower than the least squares fit; another is 1.6% low, while the average of all measurements is 1% below the recommended values derived from the multiparameter fit. A similar situation exists with regard to 2200 m/s fission cross-sections, wherein the discrepancies far exceed the precision quoted for each experiment. Since these measurements represent the cornerstone of a strongly interrelated structure of nuclear data utilized in reactor physics, it is important that they be independently and accurately evaluated, despite the fact that there exists strong confidence in $\sigma_i$, $\eta$, and $\omega$ values which provide an over-determined set of parameters. An experiment is reported which has been designed to circumvent certain plausible systematic errors which might be responsible for the discrepancies. The total neutron yield was measured for thermal neutron fission of $^{235}$U and also independently for $^{235}$U with a improved manganese bath apparatus. Evaluation of probable error sources has led to the adoption of a sequence of precision techniques subjected to extensive verification. The neutron yield of a fission counter was determined with the manganese bath; the accuracy of the bath system was independently corroborated with 0.7% precision against the U. S. National Bureau of Standards secondary neutron source. Absolute beta-gamma and relative gamma-gamma coincidence techniques are important facets of this calibration. The fission rate of the fission counter was found in a separate prompt fission-neutron coincidence experiment, borrowing well-established methods from beta-gamma coincidence work. In the course of this calibration it was discovered that angular anisotropy in fission neutron emission is much more a problem than universally realized; it is possible that some of the discrepancies in reported $P$ and $\sigma_i$ measurements result from discounting this correction too readily. This conclusion is supported with a series of angular traverses and by the contemporary $P$ results reported. (auth)

150.

4276 CROSS SECTIONS 150-155.

absorption cross section, and fission cross section of $^{239}$Pu. The measurements were performed in a thermal and epithermal flux. (D.C.W.)

153.

5460 EVALUATION OF NEUTRON CROSS SECTIONS: CALCULATIONAL METHODS AND EVALUATED LIBRARIES.

From IAEA Conference on Nuclear Data, Paris, France.

Two theoretical models are outlined which are useful in conjunction with experimental data, for providing neutron cross sections for reactor and shielding calculations. Several recommended sets of derived data are also given. (S.F.L.)

154.

4276 MEASUREMENT OF AVERAGE CROSS-SECTION RATIOS IN FUNDAMENTAL FAST-NEUTRON SPECTRA.


From IAEA Conference on Nuclear Data, Paris, France.

Spectral indices (energy-sensitive average cross-section ratios) are presented for the fast neutron spectra at the center of bare and natural-uranium-reflected spheres of $^{235}$Pu, $^{233}$U, and $^{235}$U. By means of energy-sensitive nuclear reactions, the six fast critical assembly spectra are compared (1) among themselves, (2) to corresponding fission-neutron spectra, and (3) to neutrons of a single energy. Experimental errors depend upon reproducibility of detecting systems, spectral distortions due to the presence of detecting systems, and purity of the monoenergetic and fission neutrons. Recommended excitation functions and estimated uncertainties are presented for each detector reaction in coarse group structures for computational checks. The observed spectral indices and detector excitations, together with their uncertainties, provide a set of values that must be met by the microscopic cross sections $^{239}$Pu, $^{237}$U, and $^{235}$U via transport computation. (S.F.L.)

155.

5558 EVALUATION OF CROSS SECTIONS AND RESONANCE PARAMETERS FOR $^{235}$U, $^{237}$U, $^{239}$Pu, AND $^{241}$Pu BETWEEN Cd-CUTOFF AND 10 keV.

From IAEA Conference on Nuclear Data, Paris.

Fission and capture cross sections between Cd-cutoff and 10 keV have been evaluated for the fissile isotopes, based on the available experimental differential cross sections, capture-to-fission ratios, and total fission and capture resonance integrals. In particular, the recommended cross-sections contained in the 1965 edition of BNL-325 were compared with some other evaluations and with more recent results of cross-section measurements using nuclear explosives in space and underground. As a result of this evaluation, it was found that even for the most intensively studied nucleus, $^{239}$Pu, the uncertainties in the fission cross-section $\sigma_f$ are about ±5% and in the capture cross-section $\sigma_c$. For $^{239}$Pu, $\sigma_i$ is known to ±10% and $\sigma_a$ to ±25%. For $^{235}$U and $^{237}$Pu, the uncertainty in $\sigma_i$ is ±20% and in $\sigma_a$ ±20%. Different sets of resonance parameters for the resolved resonances, which are recommended in BNL-325 or by other research laboratories, were compared by calculating infinitely dilute total partial resonance integrals. Calculations were performed with the resonance integral code TRIX-1 which was developed at Atomics International and is based on the single-level Breit-Wigner model. The well-known limitations of this model for nuclei, such as the
Cross Sections 156-161

Fissile nuclei, with narrowly spaced interfering levels, result in calculated fission resonance integrals which are 10 to 20% too small when compared with integrated measured differential cross sections. This result was found for energy regions where the cross sections are sufficiently well-known to make such a comparison possible. Average parameters for the unresolved resonances have been extrapolated from resolved resonance parameters. Calculated partial resonance integrals, using these parameters, were compared with integrated measured cross sections where these are available. As no published capture cross-section measurements exist in the resonance region for $^{235}$U, $^{239}$Pu, and $^{241}$Pu these cross-sections were calculated with the determined average parameters and appropriate statistical distributions. (auth)

156.

5535 (OINL-P-2599) RATIO OF CAPTURE TO FISSION IN $^{235}$Pu AT keV NEUTRON ENERGIES. Lottin, A.; Weston, L. W.; de Sauvage, G.; Young, Y.; Yee-Yak Huig (Hollidaysburg, Pa., Tenn.), [1966]. Contract W-7405-eng-26. 17p. (CONF-661019-9). Dep. mn. CFSTI $2.00 cy, $0.65 mn.

The neutron capture to fission ratio, $\alpha$, was measured for $^{239}$Pu for neutron incident energies from 20 to 600 keV. A pulsed beam of neutrons was collimated on a sample of plutonium placed in the center of a large hydrogeous gamma-ray scintillator poisoned with gadolinium. A capture event in the sample was characterized by a single pulse of the scintillator due to the cascade of capture gamma rays, while a fission event was characterized by a pulse due to the prompt fission gamma rays followed, a few microseconds later, by additional pulses due to the gamma rays produced when the thermalized fission neutrons were captured in the gadolinium of the scintillator. Below 100 keV neutron energies were measured by the time-of-flight technique with a resolution of 7 nsec/m. Above 100 keV approximately monoenergetic neutrons were used. Similar measurements were also performed on $^{235}$U and the results are included for completeness. The uncertainties in the values of $\alpha$ obtained from these measurements have been extrapolated to the values by other laboratories. (auth)

157.


The contents of a library of revised Hansen-Roach 16-group cross-sections are summarized and are discussed. Modified Be and Zr cross sections are included together with $^{232}$Pu cross sections and cross sections of other elements used in SNAP reactors. (D.C.W.)

158.


From IAEA Conference on Nuclear Data, Paris.

The exact formulation of a method is given to improve the microscopic cross-section evaluation by means of integral data. Considering a function $\sigma(E)$ and a set of functional forms for this function $\gamma_i(E)$, it is proposed that the function $\tilde{\sigma}(E)$, which minimizes the quadratic form $\int [\sigma(E) - \tilde{\sigma}(E)]^2 dE + \sum_1^n \gamma_i(E) - \tilde{\gamma}_i(E)]^2 dE$ is the best estimate of $\sigma(E)$, where $\tilde{\sigma}$ is the estimate obtained from the analysis of direct measurements of $\sigma(E)$, $\Delta \tilde{\sigma}$ is the standard deviation of a single measurement, and $w(E)dE$ is the significance attributed to the estimate in the interval $dE$. The $\gamma_i$ are the exponential values of $\gamma_i(E)$, and $\Delta \gamma_i$ are their standard deviations. Explicit expressions for $\tilde{\sigma}(E)$ and $\Delta \tilde{\sigma}(E)$ are derived, and various special cases and applications are discussed. To illustrate the method, data from a few simple systems are used to correct the appropriate cross sections. (auth)

159.


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

The pulsed-source, time-of-flight method was used to measure neutron angular flux spectra to 15 MeV from a spherical $^{235}$U metal critical assembly and spheres of tungsten. The object of the $^{235}$U experiment was to test calculation methods for fast reactor spectra and the model for neutron inelastic scattering. The tungsten spectra were measured at 30°K and 800°K, as part of a current program to measure the effect of Doppler broadening on spectra. The principal features of the experimental technique are discussed, particularly time resolution, beam extraction, and detectors. The $^{235}$U sphere measurements are presented and are compared with GAPLSN calculations. The spectrum is very close to that in a critical reactor, but there is a discrepancy between theory and experiment. The tungsten results show a clear difference between hot and cold, even to high energies. The difference can not be explained by the density change, and it is apparently the result of Doppler broadening scattering resonances. Calculations do not take this into account. (auth)

160.


The latest available nuclear data were collected for $^{232}$Th, $^{235}$U, and $^{239}$U, and recommended values of thermal cross sections and resonance integrals are given for these isotopes for use in thermal-reactor calculations in the analysis of thorium fuel cycles. Sixteen-group cross sections for studying fast systems based on the same fuel cycle were prepared. Accuracies in the data and the need for further measurements and improvements were specified. (auth)

161.


The F.D.2 is a 33-group cross-section library produced for sodium-cooled fast reactor calculations. The data were derived mainly from the UKAEA Nuclear Data Library using the GALAXY program and a smooth weighting spectrum, which approximates that of a large sodium-cooled fast reactor. Other sources were used to provide: resonance shielding factors for $^{235}$U, $^{232}$Th, $^{232}$Th, and $^{235}$Pu; sodium elastic moderation and transport cross sections and the capture cross section for the 3-keV resonance; elastic
moderation cross sections for oxygen, chromium, iron, and nickel; and data for substances that are not included in the Nuclear Data Library, or for which more recent evaluations are available. With the exception of the highest energy group, the group boundaries are at half-lethargy intervals based on an origin at 10 MeV. The top group covers energies above 3.68 MeV and has a $^{235}$U fission-spectrum weighting. The remaining 32 half-lethargy groups cover the energy range down to 0.414 eV. The scatter matrix was limited to ten down-scatter groups with scattering to lower energies included in the scattering to the tenth down-scatter group. (S.F.L.)
4. REACTIVITIES

1956

1. 10534
An unreflected, spherical U-235 critical assembly has been in operation at the Los Alamos Scientific Laboratory since August, 1951. A remotely controlled mechanical system is used to assemble subcritical components of the sphere, and reactivity is adjusted with U-235 control rods. The maximum power level during sustained operation is about 1 kw. Investigations with the assembly include studies of the neutron spectrum, observation of the changes of reactivity produced by inserting foreign materials into the assembly, and determination of parameters such as the temperature coefficient of reactivity. In addition, experiments at reactivities above prompt critical have been carried out.
The assembly has also been used as a source of short, high-intensity bursts of neutrons in the study of delayed neutrons following fission. (auth)

1957

2. 13161
KAPL-M-PLI-16
Knolls Atomic Power Lab., Schenectady, N. Y.
$1.80(ph OTS); $1.80(mf OTS).
Reactivity coefficients of samples of Hf, Cd, and B of various thicknesses have been calculated by means of perturbation theory at the center and core edge of the clean PPA-18. Calculated and experimental results are compared. (auth)
Reactivities 5-10

1958

5. LA-1525
Los Alamos Scientific Lab., N. Mex.

The perturbation theory for material replacement experiments in a bare assembly is given through second order, thus permitting corrections for sample size. Computed flux and adjoint distribution functions are tabulated for the Lady Godiva assembly, enabling the observed danger coefficients for U\textsuperscript{238} and U\textsuperscript{233} to be compared with corresponding predicted values. Consistency of this data is checked by its use in three independent combinations, each yielding the same value for the effective fraction of delayed neutrons from fast fission. Reactivity contributions associated with inelastic scattering were calculated and comparison made between central danger-coefficient ratios and ratios for the Topsy assembly. Evaluation of transport cross sections by means of replacement measurements in Godiva is illustrated for the several elements C, Cu, and Au. (auth)

6. HW-27851
Hanford Atomic Products Operation, Richland, Wash.

Reactivity measurements to allow determination of the amount of poisoning produced by the Cd and Ta eves and to determine the relative loss of Pu\textsuperscript{238} when removing Pu\textsuperscript{238} are reported. (F.S.)

1959

7. NP-7011
Iowa State Coll., Ames.
CHANGES IN REACTIVITY IN A SUBCRITICAL ASSEMBLY OWING TO LOCALIZED PERTURBATIONS. (thesis). Donald Miller Beck. 1959. 72p.

The reactivity changes in a subcritical assembly due to localized perturbations were investigated. The perturbations considered were the addition of fuel, poison, and the introduction of a void. Experimental and theoretical results were correlated to ascertain the feasibility of predicting reactivity changes in a subcritical assembly. (W.D.M.)

8. DNL-485(s.75-6)
Los Alamos Scientific Lab., N. Mex.
REACTIVITY COEFFICIENT DATA ON U\textsuperscript{233} IN FAST CRITICAL ASSEMBLIES. G. E. Hansen. p.75-6 [of] THORIUM—U\textsuperscript{233} SYMPOSIUM, SPONSORED BY THE UNITED STATES ATOMIC ENERGY COMMISSION AT BROOKHAVEN NATIONAL LABORATORY, JANUARY 9-10, 1958. 2p.

Data are presented on reactivity coefficient ratios obtained at the centers of Topsy (U reflected Oy), Godiva (Bare Oy), and Jezebel (bare Pu). (W.L.H.)

9. HW-27851
Hanford Atomic Products Operation, Richland, Wash.

Reactivity effects of large voids in the reflector of the Pool Critical Assembly (PCA), an enriched-uranium, light-water-moderated and reflected, pool-type reactor, were investigated experimentally and theoretically. The four principal effects measured were the variations of reactivity of voids at the center of a core face, void position on core face, and separation distance between void and core face. Superposability of the reactivity effects of multiple voids was studied. The effect of the largest void on the thermal neutron flux distribution was measured. (W.D.M.)

1960

10. AECU-4891
(Oak Ridge National Lab., Tenn.).

Submitted to Massachusetts Inst. of Tech.

Reactivity effects of large voids in the reflector of the Pool Critical Assembly (PCA), an enriched-uranium, light-water-moderated and reflected reactor, were investigated experimentally and theoretically. The three principal effects which were studied experimentally were: the variation of reactivity with the size of a void located at the center of one face of the core (including a void covering the entire face), the variation of reactivity with void position on the core face, and the superposability of the reactivity effects of voids. The effect
11.  
10876  KAPL-M-MLS-12  
Knolls Atomic Power Lab., Schenectady, N. Y.  
Measurements of the core distributed reactivity coefficients of moderator and structural materials that were made in a polyethylene moderated and reflected slab reactor (PMA-6) are described. In addition, the experiments are analyzed with the use of various cross section routines currently employed in nuclear design studies at this laboratory. The adequacy with which the WOX-type and 3-Group MUFT-SOFOCATE cross-section codes predict the experimental reactivity coefficient data is demonstrated. (auth)  

12.  
Reactivity effects of 93.2% enriched oralloy, molybdenum, iron, niobium, 310 stainless steel, Inconel X, FeCrAl, zirconium, nickel, nichrome V, and chromiunm clad with FeCrAl, in the form of foils in the center cell of the BEM-II B minimum void, beryllium moderated critical experiment are reported. Nineteen-energy-level diffusion calculations with cell corrections from both coarse and fine energy detail were correlated with the measurements. The fine energy detail improved the correlation, mainly because resonance self-shielding and flux depression are more adequately treated, and residual discrepancies for the most part can be attributed to inadequacies in the cross section data. (auth)  

13.  
7011 (ORNL-3016(p.3-5)) REACTIVITY WORTH OF THE CENTRAL FUEL ELEMENT IN THE BULK SHIELDING REACTOR I. G. deSauesse, K. M. Henry, Jr., and R. Perez-Belles (Oak Ridge National Lab., Tenn.).  

14.  
An accumulation of reactivity contribution data is brought together for the various critical assemblies at Pajarito. Corresponding values of effective absorption and transport cross sections are derived, and relations between critical mass and volume fraction of divulents are obtained in terms of these cross sections. In some favorable cases, inelastic scattering contributions to the effective absorption cross sections are estimated. (auth)  

15.  
The first-order perturbation theory, or equivalently the interpretation of reactivity coefficients, is illustrated for fast neutron critical systems. A second-order perturbation theory for small size sample replacements is given and illustrated especially as it pertains to measurements of reactivity coefficients. (auth)  

16.  

17.  
A listing and description is given of the experiments associated with the HFIR Critical Experiment-2. The primary experiments concern the reactivity of the bare core, reactivity worth of "gray" control plates, core-power
Reactivities 18-22

distribution, reactivity worth of "black" control plates, temperature coefficients of reactivity, and the island void coefficient of reactivity. The secondary experiments concern the reactivity of the fuel, and the reactivity worth of a "partial" gray plate. (auth)

18.


The reactivities of several U-Nb-Zr alloys in the Sodium Reactor Experiment are compared with that of 2.8% enriched U fuel. The alloys considered have the following compositions: 15% Nb - 7% U - 15% Zr; 12% Nb - 82% U - 6% Zr; and 6% Nb - 82% U - 12% Zr. Each alloy is studied at U enrichments of 3, 7, and 11%, and the alloy enrichments necessary to give the same reactivity as the 2.8% enriched U fuel are found by interpolation. (T.F.H.)

19.


Results of reactivity coefficient and flux distribution measurements in the unpoisoned PMA-40 slab core are described. The PMA-40 core is the first in a series of slab cores to be constructed in the KAPL Plastic Mockup Assembly (PMA) for evaluating analytical techniques and cross-section schemes. In addition to the distributed reactivity coefficients of $^{238}$U, aluminum, and polyethylene, the spatial variation in the reactivity coefficients along the short 12-in. dimension of the core is presented. The description of the core, measurements, and results presented is complete enough so that calculations can be performed without additional information. (auth)

20.


The results concerning neutron flux distributions, distributed reactivity coefficients of $^{238}$U, polyethylene, and zircaloy as well as the spatially dependent reactivity coefficients of zircaloy, aluminum, and zirconium-boron in PMA-47 are reported. PMA-47 is the second of a series of cores constructed in the Plastic Mockup Assembly (PMA) to provide reactivity coefficients and criticality data in cores with simple geometry. It differs from the first core of the series (PMA-40) in that it contains a higher fuel density and utilizes zirconium as its structural material. The reactivity data will be used to test analytical techniques and to provide a comparison with related values for subsequent similar aluminum-uranium cores. (auth)

1962

21.


Experimental Breeder Reactor-I. Three methods were used to determine the reactivity worth of the EBRI lead cup reflector to be 5.4%. The feedback mechanism calculated for the resonant instabilities of Mark-II core is discussed in some detail. ZPR-III. Criticality and reactivity measurements for a wide variety of materials are reported for Assemblies 29 to 31 (uranium oxide) and 32 to 33 (stainless steel-uranium). BOHAX-V. Current development and fabrication work carried out on the BOHAX-V facility are described. Transient Reactor Test Facility. Results are presented for a 1010-Mw-sec temperature-limited transient in the facility and for the integrated and peak power as a function of reactivity addition. A temperature rise vs power graph was derived for calculating the energy release of transients. Gamma flux measurements are discussed. (D.L.C.)

22.


Two spherical assemblies of earlier cylindrical assemblies were used for a critical study of shape effects for fast reactor cores with volumes of 300 to 400 liters. Assemblies 24 (cylindrical) and 38 (spherical) had a high-density metallic U blanket, whereas the set of assemblies numbered 31 (cylindrical) and 39 (spherical) had a low-density U-fueled core (with steel and Al diluents) with a high-density blanket of depleted U. The main features of these assemblies are summarized. Reactivity coefficients of a small number of fissile and nonfissile materials were measured in both assemblies (38 and 39). In Assembly 38 the effects of environment, etc., upon fission rates measured with absolute fission chambers were investigated. Radial fission rate traverses in different directions were made in Assembly 39 to reveal any flux asymmetry due to heterogeneity of the core; no such effect was detected. (auth)

The Doppler coefficient, \( \langle 1/\bar{d} \rangle dI/dT \), was measured for several HTGR fuel elements in the HTGR critical assembly at temperatures up to 650°K by a reactivity oscillation technique. The correction required because of a small effect upon neutron temperature caused by heating the central fuel element (the thermal base effect) was also determined experimentally. The coefficient, \( \langle 1/\bar{d} \rangle dI/dT \), is about 3.6 \( \times 10^{-4}/\circ C \) at 300 to 650°K. It increases slightly with thorium loading. (auth)

25


A modified version of the inhour equation, which relates the reactivity to the rate of change of power of a reactor, was used. It takes into account the different probabilities of leakage from the reactor core during the slowing down of prompt and delayed neutrons. Using this equation the reactivity absorbed by a control element in a reactor was deduced. Good agreement was found with other methods of estimating the element effectiveness including critical mass measurements and sub-critical measurements with distributed absorbers. (auth)

26


The NASA zero-power reactor, consisting of an unreflected cylinder containing a solution of uranyl fluoride salt in water, was used to study simple heterogeneous effects. This heterogeneity was introduced into the reactor as tubes or rods of aluminum, iron, and tungsten. For a given hydrogen- to uranium-235-atom ratio of the fuel solution, criticality was achieved by varying the height of the fuel solution contained in a 12-inch-inside-diameter tank. Criticality measurements of cylindrical void regions located on the axis of the reactor were also made. These measurements were analyzed by using 19 energy groups with a one-dimensional diffusion-theory code written for the IBM 704 computer. (auth)

27


Critical studies were performed on a dilute, metallic, fast reactor core. The fuel was 38.6% enriched uranium and reduced-density aluminum was used to simulate sodium coolant. The measured critical mass was 463 kg of U 235 in a 425-liter core. The main part of the experimental program consisted of measurements of the standard fission ratios, a number of central reactivity coefficients, and the prompt neutron lifetime. An additional series of experiments were performed to obtain the worths of aluminum, aluminum oxide, and sodium at various radial positions in the core. (auth)

28


Results of studies with ZPR-III Assembly 29, a mockup of a typical, dilute UO 2 -fuelled fast reactor, are reported. The assembly consisted of a 454-liter cylindrical core, blanketed with depleted uranium, with a critical mass of 421 kg U 235. Experiments included measurements of fission rates, material reactivity worths, and Rosal alpha. The results of multigroup calculations using the present Argonne cross-section sets are presented, and discrepancies between experimental results and calculations are discussed. (auth)
Reactivities 29-34

29.

3837 (ORNL-3193[p.3-4]) THE REACTIVITY EFFECT OF AN AIR-FILLED CAVITY WITHIN A POOL CRITICAL ASSEMBLY CORE. E. B. Johnson (Oak Ridge National Lab., Tenn.).

It is shown experimentally that a positive void coefficient of reactivity exists in the core center-line of the Pool Critical Assembly. The assembly is fueled with BH蒸汽 fuel elements, and the void is introduced in a Lucite container in order to keep the metal/water ratio constant. (T.F.H.)

30.

18612 (DC-59-7-700) REACTIVITY MEASUREMENTS DURING PHASE II TESTS WITH THE HOTCE REACTOR.


The results of reactivity measurements made with the HOTCE reactor during phase II tests are given along with the significant test procedures that evolved during the experiments. The reactivity corresponding to the asymptotic periods were determined from the in-hour measurement or physical properties, static reactor parameters, and parameters which are necessary for the operation of the reactor. (M.C.G.)

31.


(In German)

In this survey report on reactivity measurements, the kinetic behavior of a reactor is first considered, then the methods used for the measurement of the reactivity are reviewed. The application possibilities discussed are measurement of physical properties, static reactor parameters, dynamic reactor parameters, and parameters which are necessary for the operation of the reactor. (J.S.R.)

32.


(In German)

In this survey report on reactivity measurements, the kinetic behavior of a reactor is first considered, then the methods used for the measurement of the reactivity are reviewed. The application possibilities discussed are measurement of physical properties, static reactor parameters, dynamic reactor parameters, and parameters which are necessary for the operation of the reactor. (J.S.R.)

33.

28438 REACTIVITY EFFECTS OF MODERATOR EXPULSION IN AN ENRICHED, LIGHT WATER REACTOR.


An experimental and analytical investigation of the effect of removal of part of the water moderator on the reactivity of a small pool-type reactor is made. The experimental technique consists of bubbling air through one element of the core at a time. The air volume (which is considered to be a void) is determined by calibrating measurements made in an out-of-core gamma ray densitometer. Density perturbations made in individual (3" × 3") core elements produce void coefficients (ν s/v/k/cm/dv) varying from -0.316 at the center to -0.110 at the outer corner of a roughly square core. The average void coefficient produced by volumetrically averaging all the local coefficients is -0.190. Linearity of the reactivity effect with void concentration is observed up to the maximum void fraction introduced of 11 per cent. This result is compared with a diffusion-theory analysis that includes a comparison of two-group perturbation, using one-dimensional and two-dimensional eigenvalue-variation methods. This comparison shows that for uniformly distributed density changes, perturbation theory is simple and accurate within 5 per cent. It is also apparent that variation of the geometry (e.g., cylindricalizing a parallelepiped) has very little effect on either uniformly distributed or localized reactivity effects. Effects of neutron streaming, neutron temperature changes, and changes in flux depression are shown to be negligible in the average water-density range (ρ > 0.75 gm/cc) examined. The most important feature not adequately accounted for by applying density-changed cross sections to one- or two-dimensional codes is the variation of transverse bucklings. This feature is examined with the aid of the Buckling Iteration subroutine of the MAGNUM code. The combination of buckling iteration with any of the two-group techniques considered adequately solves the distributed void coefficient problem. However, for localized voids this technique is not accurate since it assumes no radial variation of axial buckling (separability). The interior void coefficient is overestimated while the outer void coefficient is underestimated (both by about 15 to 30 per cent). On the basis of this evidence suggestions are made for modification of the buckling iteration procedure to allow radial variations of axial bucklings.

34.


The purpose of this report is to compare calculated criticality, reactivity coefficients, and flux distributions with related quantities measured in PMA-47. PMA-47 is
the second of a series of cores constructed in the Plastic Mockup Assembly (PMA) to provide criticality data and reactivity coefficients in cores with simple geometry. A comparison of results is made to PMA-40 which was the first in this series. The two cores differ in that PMA-47 contains a higher fuel density and utilizes Zr as its principal structural material. The calculations were performed using the DIG~WOXX and MUFT-KATF cross section routines with the KARE diffusion theory code. (auth)

1963

35.

4224 (ANL-6635(p.8-27)) LIQUID METAL-COOLED REACTORS. (Argonne National Lab., Ill.)

Experiments were conducted with ZPR-III assembly 41 which consists of 20% enriched U diluted in metal. Data on central fission ratios and sample-size effects in measurements of central- and radial-edge-reactivity coefficients are presented. Other experiments are reported on blankets and reflectors and a rotary control rod mockup containing boron carbide in the radial blanket. In developmental work on the ZPR-VI and IX, installation of the control wiring for the reactor-cell air conditioning and installation of the Arr storage system were completed. A summary of foil irradiation activities is included. Developmental programs concerning EBR-I, EBR-II, and FARET are summarized. (J.R.D.)

36.


37.


The reactivity of the Advanced Test Reactor Critical Experiment (ATRCE) was evaluated with the core in the reference condition. Control cylinders were calibrated using determinations of positive reactor periods interpreted by the in-hour equation. The pulsed neutron technique was used to measure subcritical reactivities. Described are the reactivity evaluations for the hafnium neck shim rods, the outer shim control cylinders, and the borated Ni/Fe tape used in the core region. The results of perturbation measurement of the reactivity coefficient of U^{235} are included. (H.D.R.)

38.


The theory of the source-jerk technique for the measurement of subcritical reactivity was refined to take account of harmonics present in the neutron flux distribution. The method was applied to the measurement of the subcritical activity in a large graphite-moderated uranium-fueled reactor. The subcritical reactivity was found to be [-2.63 ± 0.21] per cent ΔK/K. This value is about 10 per cent higher than estimates based on the subcritical approach to critical and on the measured interaction of shutoff rods. (auth)

39.

12765 (ORNAL-3360(p.12-16)) EXPERIMENTS AND CALCULATIONS ON THE REACTIVITY EFFECTS OF SMALL FUEL DISPLACEMENTS. E. G. Silver, Z. M. Bartolome, et al. (Oak Ridge National Lab., Tenn.).

The reactivity effects associated with the motion of reactor fuel are being investigated in a calculational and experimental program at the BSF. Results indicate that negative reactivities result from all perturbations that decrease the homogeneity of the core, except when the perturbation adds fuel to high-importance regions, such as those adjacent to interior water volumes. The calculation methods appear to be inadequate to cope with the kind of small-dimension, large-magnitude perturbations represented by the experiments; however, experiments are being designed which can be more accurately represented in the calculations. (auth)

40.

29733 NEUTRON LIFETIME AND REACTIVITY MEASUREMENTS IN THERMAL REACTORS WITH THE HELP OF THE ROSSI-α-METHOD. Theodor Stribio (Siemens-Schuckertwerke AG, Garching, Ger.). Nukleonik, 5: 170-2(June 1963). (In German)

Rossi α-measurements on two thermal reactors, the Siemens Instruction Reactor (SUR) and the Siemens Argonaut Reactor (SAR), are reported. Methods and equipment are described and the results presented. For the SUR an average neutron lifetime of 48.75 x 10^{-4} sec (with β_{eff} = 7.4 x 10^{-3}) was obtained, for SAR a lifetime of 173.6 x 10^{-4} sec (with β_{eff} = 8.5 x 10^{-3}) was determined. Reactivity measurements on the basis of the α-method agreed satisfactorily with those obtained by the "rod-drop" method. The statement concerning the size of the time-dependent part of the frequency distribution, which is supplied by theory in the energy and place independent region, could not be confirmed. (tr-auth)
Reactivities 41-46


Measurements are made of the prompt neutron lifetime and of the equivalent reactivity worths of control rods and the gap between two halves of the SHE-1 (Semi-Homogeneous Critical Assembly Core-1) by pulsed neutron technique. The decay constant $\alpha$ can be measured in the range from 7.19 to 147.7 sec$^{-1}$ within 2.1%. The prompt neutron lifetime of this assembly is determined to be $1.21 \pm 0.03$ ms (assuming $\lambda_{eff} = 0.0067$), and the equivalent reactivity worth of a Cd control rod to be $4.10 \pm 0.13$. However, the reactivity found from the value of $\alpha$ by the conventional formula, $\delta = (\alpha - \alpha_0)/\alpha_0$ is found to have a larger error than experimental one below about $-9\%$ for SHE-1, because the change of the value of $k_{eff}/\lambda$ with the degree of subcriticality is neglected. The theoretical validity of reactivity measurement in the represented subcritical system by pulsed neutron technique is discussed. (auth)


Reactivity coefficients, criticality data, and flux distributions were measured in the Plastic Mockup Assembly for an unpoisoned, slab reactor. The core was constructed in two sections, one stationary and the other on an elevator platform. The fuel was 20 wt % uranium enriched to 93% $^{235}$U in a zirconium matrix plate. Reactivity coefficients were measured on both a position dependent and core distributed basis. The reactivity change associated with a particular material substitution was evaluated by means of doubling time measurements and were in the range from 10 to 20%. The results were tested with two cross-section schemes. (M.C.G.)

1964

43.


44.


Pulsed neutron source measurements were made for a series of subcritical to critical loadings on a bare $^{235}$U water assembly in slab geometry. For each loading, the assembly was repetitively pulsed, and the time distribution of thermal neutrons in the core was recorded on time analyzers. From these data, the reciprocal die-away time ($\alpha$), the delayed neutron background, and the reactivity of the system were determined. Measurements at various positions in the assembly verified that spatial modes do not affect the reactivity determination. (auth)


45.


The reactivity change due to an increase in the radius of an empty hole was measured in a heavy water-moderated reactor, and some results differing from experiments with ZEEP were obtained. It was concluded that the streaming in a hole is not so effective for reactivity. In measuring the neutron flux in a void, a flat thermal neutron flux distribution was obtained; and it was concluded that the neutrons leaking through the empty hole or the void do not consist of thermal neutrons, but of fast neutrons for the most part. The experimental result of reactivity change due to the void location in the core indicates that the relation between the void location and the reactivity change is independent of the neutron flux distribution. (auth)
47.


The relative worths of six PuO₂–UO₂ fuel elements and a Pu–Al fuel element in the center of the Plutonium Recycle Critical Facility were determined from positive period data. Five of the PuO₂–UO₂ elements will go into the PRTR; the sixth will be retained as a standard. The worths of the five elements relative to the standard, D₂O, and Pu–Al were obtained. The variation in worth relative to D₂O of the five elements was ±0. A consistency within 1% between the measured worths and the PuO₂ content of the elements was observed. The use of total reactivity worth as a criterion indicates that the stable PuO₂–UO₂ elements were very uniform. (D.C.W.)

48.


Fuel blocks of compacted ZrO₂ and UO₂ powder in paraffin were added axially to two unreflect ed assemblies (20 x 16 in. and 20 x 24 in. cross sections). The separation of the assembly halves when criticality occurred was determined, and the reactivity worth at a given critical gap size was obtained by bringing the halves slightly closer together and observing the reactor period. The core with the larger cross section had a maximum gap twice that of the smaller core. Diffusion theory treatment agreed with the data reasonably well for small gap thickness. (D.C.W.)


49.


Reactivity change measurements were carried out in the graphite-moderated reactor HECTOR on plutonium–uranium fuel elements with a low Pu-content. The aim of the work was to develop a method in which measurements on small fuel samples (typically 18 in. long) could be used to check the validity of proposed methods of calculation for plutonium-bearing lattices. The experiments were carried out in a variety of lattices and demonstrated that at a lattice pitch of 8.5 in., which was used for the majority of the measurements, the results were not significantly affected by the lattice environment. In particular, the results showed that it is not necessary to use plutonium–uranium fuel in the lattice surrounding the sample but that uranium fuel with similar properties is adequate. The accuracy obtained corresponded to an error of 0.5% in kₐ for a lattice containing the plutonium–uranium fuel. The results of the measurements were compared with the predictions of the TRACER method of lattice calculation. It was found that, at room temperature, there was no significant difference between the measured and calculated values of kₐ but that, at temperatures in the range 400 to 450°C, the calculated value of kₐ was about 1½% lower than the measured value. (auth)

50.


Results are given of reactivity gains in PRR-I due to fuel addition or removal at various core positions. (R.E.U.)

51.


Some experimental results for NORA in-pile tests of enriched fuel clusters for the second loading of the Halden Boiling Water Reactor are presented. Measurements were made of thermal-neutron flux distributions, moderator level coefficient of reactivity, core importance function, flux depression by an SSJ-himble, and void effects. (R.E.U.)

52.


Reactivity effects of various materials of interest to epithermal and fast reactors were measured in a series of neutron spectra with median energies of neutrons causing fission ranging from 422 kev to 8.5 ev. An oscillator technique capable of sensitivities of 10⁻⁶ A/m was used.

In addition, a power-history or reactivity-vs-time method was developed for fast, but less sensitive reactivity measurements. It was used to obtain radial reactivity traverses and rod calibrations and for other routine reactivity measurements. (auth)
Reactivities 53-63


For understanding the characteristics of a boiling water reactor measurements were carried out of various coolant void reactivity effects in the zero power facility NORA at room temperature and in the Halden BWR between 55°C and 220°C. The measurements in NORA were carried out with a core of 7-rod UO₂ cluster fuel elements of the HBWR. The shroud of one fuel element was subdivided in sections in which the water level could be depressed by air. The temperature dependence of the void effect has been determined by removing coolant from one UO₂ cluster fuel element in the HBWR. The coolant channel was voided to various depths by depressing the D₂O level in it by means of argon under pressure. (auth)


It is shown that the minimum critical enrichment of uranium in U²³⁵ in aqueous homogeneous systems is independent of density. It is concluded that it should be possible to handle homogeneous or nominally homogeneous uranium salts without regard to density, so long as the minimum critical U²³⁵ enrichment is not exceeded. (T.F.H.)


It is disclosed that the most widely accepted method for determining sodium reactivity effects employs a standard synthesis technique using diffusion theory in one-dimensional cylinder-and-slab geometries. The majority of multigroup diffusion theory codes used employ an option whereby group-and-region-dependent bucklings may be specified as input quantities, facilitating a relatively accurate iterative procedure. Effective group cross sections are derived from many sources of data; the high energy groups usually employ the cross sections of Yiflah, Okrent and Moldauer. Whenever possible the most recent energy-dependent data for cross sections, u(E), and the capture-to-fission ratios of fissile isotopes were used. A refined treatment of elastic scattering resonances (the ELMÖE code) has established the necessity of using composition-dependent cross sections. Computations using several cross section sets indicate trends in the sodium void effect for a variety of conditions. Conclusions from these results are presented. (auth)
which are not returned to the core are nearly all absorbed for the core sodium can be attained. The blanket to the core by decreasing the return probability of neutrons from the core and blanket.

\[ \text{(L.B.S.)} \]

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### Reactivities 64-69


Reactivity worths of various metal plates were measured in a quasi-homogeneous enriched uranium--beryllium oxide subcritical assembly by means of the pulsed neutron method. The materials studied included nickel, iron, cobalt, gold, Hastelloy R-235, and René-41. The equivalences were determined between the various absorbers and fueled core material. Based upon the measurements it was concluded that the prompt lifetime is relatively insensitive to massive localized absorbers. (auth)


Reactivity comparisons were made in Aquilon II between geometrically identical lattices differing only by the composition of the fuel. The fuel elements consist of metallic uranium single rods with either slight differences of the isotopic composition (0.69--0.71, 0.83--0.86 per cent of uranium 235) or slight additions of plutonium (0.043 per cent). Five lattices pitches were used in order to produce a large variation of spectrum. Two additional sets of plutonium fuels are prepared to be used in the same conditions. The double comparisons: natural-enriched versus natural-enriched plutonium, are made in such a way that a very precise interpretation is permitted. The results are perfectly consistent, which seems to prove that the calculation methods are convenient. Further it can be inferred that the usual data, namely for the ratio of the \( \eta \) of \(^{235}\text{U}\) and \(^{239}\text{Pu}\) seem reliable. (auth)


It is shown that a permanently voided gap between the core and blanket in a fast reactor substantially increases the leakage effect when sodium is removed from the core by decreasing the return probability of neutrons from the blanket to the core so that a negative voiding coefficient for the core sodium can be attained. Since the neutrons which are not returned to the core are nearly all absorbed in the blanket, there is no loss in overall breeding ratio. (L.B.S.)
Reactivities 70-75

70.


The spatial distribution of reactivity coefficients in the epithermal region was measured in a graphite-moderated critical assembly (SHE). Unlike the full activation method, the method used permits free selection of a sample regardless of the activation of nuclei, the mode of decay, and the workability of a foil. If the distribution of neutron fluxes is measured with foils of the same sample, the accompanying function (a) can be estimated from the measurement of the reactivity coefficient. To improve the precision of measurements and shorten the time required for measurement, the sample contained in a Cd filter was pulled out of the core of the assembly by a manipulating system and the reactivity coefficient was measured radially from the core towards the periphery by the period method. In the investigation of the change in the reactivity coefficient due to the neutron absorption in the various energy regions, the following elements were used as samples: Au and In for the low energy region, I and Br for the medium energy region, and Co and Mn for the high energy region. The observed values of reactivity coefficients were compared with the theoretical. Both agreed very well in the core of the assembly, but in the reflector, the theoretical values were smaller than the observed. (Nucl. Sci. Abstr. Japan, 3: (Sept. 1964))

71.

8602 (ORNL-3041(p. 59-98)) REACTIVITY EFFECTS OF SMALL FUEL DISPLACEMENTS FOR PLATE-TYPE REACTORS. E. G. Silver and S. B. Johnson (Oak Ridge National Lab., Tenn.).

A slab-geometry plate-type critical experiment was constructed to measure the reactivity effects of small perturbations introduced by fuel-element displacements. These perturbations were calculated by both diffusion-theory and transport-theory codes, with careful attention to sources of error in the calculation model, in order to test the relative abilities of the two calculation methods to predict the magnitude of such perturbations. The transport-theory results were in rather good agreement with the experiment, whereas the diffusion theory was far less accurate in its predictions. (auth)

72.


The variation of neutron importance with energy and position was investigated in a multiregion critical assembly having a series of test regions typical of slightly epithermal to fast power reactors. Values of neutron importance at the center of the test regions were measured using neutron sources and a reactivity oscillator. The variation of neutron importance with position was determined using neutron sources in conjunction with a dynamic reactivity measurement technique. Analysis of data from similar beryllium- and carbon-moderned test regions indicates the significance of the Be(n,2n) reaction. The neutron sources used were Po-Be, Po-210, Po-232, and Pu. Published source spectra were used in the analysis of the data; the Mock-Fission-source spectrum was determined by gamma spectroscopy. Relative yields of these sources were determined by calibration in a manganese sulfate bath. (auth)

73.


The structure of explosively shocked magnesium oxide single crystals was investigated. Although ionic crystals with the rock-salt structure, such as MgO have been used as models to study the basic processes involved in plastic flow at low deformation rates, little is known about their response to high-velocity deformation. In the structural changes induced in MgO crystals by plane shock waves were studied by etch-pit metallography and by electron-spin resonance. The experimental shock-loading procedures were designed to induce high dynamic pressures in MgO crystals for subsequent structural studies. (auth)

74.


The discrepancy between the accepted theoretical conditional detection probability for prompt neutrons and the experimental behavior lies in the experimental observation that the multiplying coefficient associated with the time-dependent portion of the prompt neutron conditional detection probability is independent of the prompt neutron population decay constant. An equation is derived for the prompt neutron conditional detection probability that expresses the dependence of the initial count upon the occurrence of the ancestor fission, thus directly relating the counting events at the times of interest to their common antecedent. Experimental verification of this equation was obtained from the results of a series of experiments performed on the Cornell University zero power reactor. (J.S.R.)

75.


The coefficient of reactivity of Na in Na-cooled fast Pu reactors is considered. The two factors most affecting the reactivity coefficient are the concentration of the fissile material and the presence of neutron capturing materials. The behavior of structural materials with respect to Na cooling is discussed. The nuclear and mechanical characteristics of stainless steel, Ti, Zr, Mo, W, Ts, Nb, and V are discussed. (J.S.R.)
77.Cross for avantaje as and the recent termination slclcred.

The use of real and adjoint fluxes calculated with the cadmium moderated slrnnics, 1965. Contract 14012 (General source num and mined. The reactivity are negligible. Measurements with and without cadmium are consistently higher than those measured: however, the ratios of experimental worths of various samples, including natural tungsten, gold, lead, depleted uranium, aluminum, stainless steel, and lattice. The results indicate that a sensitivity of better than 0.01 cents is obtained for a single reactivity measurement taking 100 sec. This sensitivity is adequate for the resonance-integral measurements, and can readily be improved as required for Doppler measurements by taking data over a longer period of time. Calculated reactivities are consistently higher than those measured; however, the ratios of various calculated worths agree reasonably well with the corresponding ratios of experimental worths. The use of real and adjoint fluxes calculated with the cadmium sleeve taken into account should improve agreement with the expectations. These results also indicate that reactivity effects associated with scattering and structural materials are negligible. Measurements with and without cadmium end caps have demonstrated that end effects are quite -small. (auth)

78.


Measured and calculated thermal neutron decay constants are reported for various S enriched zirconium hydride moderated lattices, using UO2 and ZrH2 rods in an aluminum matrix. By performing measurements in the critical and in near-critical states, values of \( \beta/1 \) could be determined. The resulting \( \beta/1 \) values were used to calculate reactivity worths of different absorber rods and absorber arrangements from the measured decay constants. Several measurements were evaluated applying the modified pulsed source technique of Carellis and Russell. (auth)

79.


Four methods of measuring the reactivity of subcritical systems are discussed and compared. (D.C.W.)

80.


81.


82.

Reactivities 84-90


The reactivity effects of small samples of various materials were measured by the period method at the core center of Assemblies 1 and 3 of the fast zero power FB6. For some materials the reactivity change as a function of sample size was determined experimentally. The core of Assembly 1 consisted only of uranium enriched to 20% whereas the core of Assembly 3 was diluted with 30% graphite. The results were compared with calculated values obtained with a second-order transport-theoretical perturbation model and using differently shielded cross sections depending upon sample size. Qualitative agreement was generally found, although discrepancies still exist. The spectrum perturbation caused by the experimental arrangement was analyzed and found to be rather important. (auth)

86.


The reactivity of single homogeneous systems of zirconium-water-U-235 has been previously developed. This model and using a second-order transport-theoretical perturbation model and using differently shielded cross sections depending upon sample size. Qualitative agreement was generally found, although discrepancies still exist. The spectrum perturbation caused by the experimental arrangement was analyzed and found to be rather important. These measurements can be made either by an oscillator method, in which the amplitude of reactor power modulation is determined while the sample is regularly moved in and out of the reactor, or by period measurements. (J.F.P.)

88.


The reactivity of single homogeneous systems of zirconium-water-U-235 has been previously developed. This information is quantitatively extended by assessing the reactivity effect that results when homogeneous regions are separated by water gaps, that is, heterogeneous systems. Several fuel regions which are stored adjacent to each other under water do not necessarily represent a maximum reactivity. Since these regions are separated by a water gap, which is often necessary in fuel storage, a maximum peak reactivity is reached at a certain spacing between the regions. It is necessary that this maximum value always be determined before the fuel is handled unless it is mechanically impossible to vary this water gap. (auth)

89.


The spectral indices or detector ratios and buckling values characteristic of the equilibrium neutron energy spectrum in a 4.29% 235U enriched bulk of uranium metal were investigated. The 21-inch-diameter Los Alamos outdoor exponential column was utilized with the column and "hydro" source reactor elevated some 11 feet above ground level. (auth)

90.


The reactivity loss caused by a hole introduced in a reactor is calculated by using the two group perturbation theory based on neutron balance. It is shown that better agreement with experimental results is obtained by the use of a more reasonable probability function than those of Marti and Schaefer. The reactivity loss by the presence of a void is calculated by the same method, and results are obtained for several cases: When void volume is below...
a certain limit, $\Delta k(z)$ is independent of $z$ and proportional to the void volume; in the limit of small void, $\Delta k(z)$ is proportional to the square of flux. From these results the mechanism of reactivity loss is discussed. In order to justify the discussion, a comparison is made of the theoretical results with experiments on JRR-2. (auth)
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3. ill7
Deel. Feb.

SEARCH FOR DOPPLER EFFECT (OTS).

2.

Los U 235 • 8758 TID-5183
described. (L.M.T.)

In some limited and defined temperature coefficient or per

4920 BNL-1344
Brookhaven National Lab., Upton, N. Y.
TEMPERATURE COEFFICIENTS OF REACTIVITY OF
Experimental and theoretical investigations of the uniform
temperature coefficient of reactivity (the reactivity change
per degree rise in temperature of all the reactor constitu­
ents in some limited and defined temperature range) are
described. (L.M.T.)

5. DOPLER

1956

1.

4920 BNL-1344
Brookhaven National Lab., Upton, N. Y.
TEMPERATURE COEFFICIENTS OF REACTIVITY OF

Experiment and theoretical investigations of the uniform
temperature coefficient of reactivity (the reactivity change
per degree rise in temperature of all the reactor constitu­
ents in some limited and defined temperature range) are
described. (L.M.T.)

1957

2.

13563 HW-47884
General Electric Co., Hanford Atomic Products
Operation, Richland, Wash.

DOPLER COEFFICIENT OF A DILUTE SYSTEM OF
3p. Contract W-51-109-Eng-52. $1.80(ph OTS); $1.80
(mf OTS).

Experiments conducted in the Hanford Test Pile yield
the result that cylinders containing U^{235} at densities less
than 0.35 gm/cm^2 will have a zero Doppler coefficient in
the temperature range from 50 to 600°C. (auth)

3.

8758 TID-5183
Los Alamos Scientific Lab., N. Mex.
SEARCH FOR DOPLER EFFECT IN THE HEATING OF
U^{235}. David B. Hall and Jane H. Hall. Dec. 9, 1948.
(OTS).

Several experiments were undertaken in order to detect
temperature effects on the Doppler broadening of fission and
absorption resonence lines in U^{235}. It is pointed out that
rapid temperature increases in a reactor fuel rod could
cause a nuclear explosion if an increase in reactivity oc­
curred concurrent with high local heating of the fuel rods.
The effect was studied by heating a rod of U^{235} and an Al
rod, noting differences in the reactivity of a Pu reactor. In
all cases, the effect was not found. (K.S.)

4.

6100
A DIRECT MEASUREMENT OF THE URANIUM METAL
TEMPERATURE COEFFICIENT OF REACTIVITY. R. M.
Pearce and D. H. Walker (Atomic Energy of Canada Ltd.,

The U metal temperature coefficient of reactivity has
been measured in ZEEP. A U sample was oscillated in the
reactor and the resulting modulation of reactor power was
measured as a function of the sample temperature. The
temperature coefficient of uniformly heated U rods, 3.25
em. in diameter, immersed in a constant temperature mod­
erator (moderator-to-U volume ratio 22) is deduced from
this experiment. Over the range +20°C to +230°C the coef­
cient is dk/dT = -(1.25 ± 0.08) × 10^{-4} per °C. Over the
range +5°C to -140°C the coefficient is dk/dT = -(1.58 ±
0.18) × 10^{-4} per °C. (auth)

5.

17752 AERE-R/M-168
United Kingdom Atomic Energy Authority. Research
Group, Atomic Energy Research Establishment,
Harwell, Berks, England.
A MEASUREMENT OF THE CONTRIBUTION OF THE
DOPLER EFFECT TO THE TEMPERATURE COEFFI­
CIENT OF REACTIVITY IN A FAST REACTOR. A. R.

The Doppler effect in U^{235} and U^{234} in a fast reactor
was studied by measuring the changes in reactivity when hot and cold samples were placed alternately at
the center of Zeus. The results for U^{235} are consistent
with zero Doppler effect. For natural uranium a small
negative temperature coefficient was observed, corre­
sponding to increased capture in U^{235} with increasing
temperature, given by: (dR/dT) = 0.0005 ± 0.0002
mb/°C. The estimated Doppler temperature coefficient
for the whole reactor Zeus was zero within experi­
mental error. An upper limit could be placed on the mag­
itude of the Doppler coefficient (which might be positive
or negative) of 10 × 10^{-7} per °C, which is negligible
compared with the magnitude of the total negative tem­
perature coefficient of fast reactors (namely about
150 × 10^{-7} per °C). (auth)
Doppler 6-10

6.
15072 A/CONF.15/P.1777
Knolls Atomic Power Lab., Schenectady, N. Y., and
Argonne National Lab., Lemont, 111.
MEASUREMENT OF DOPPLER TEMPERATURE CO-
EFFICIENT IN INTERMEDIATE AND FAST ASSEM-
BLIES. R. T. Frost, W. Y. Kato, and D. K. Butler,
12p. $0.50 (OTS).
Prepared for the Second U. N. International Con-
Measurements made of the Doppler temperature effect
for material samples in intermediate and fast spectrum
reactors and the experimental technique applied are de-
scribed. The materials studied in the De-moderated
intermediate reactors were highly enriched U$^{235}$, U$^{238}$
and Hf. In the fast reactors, constructed alternatively
of highly enriched U and Al or Pu and Al, experiments
were performed with samples of enriched and natural U
and Pu. Because of the extremely small magnitudes of
the Doppler coefficient in the fast spectrum, only upper
bounds could be measured for the effect in the latter ex-
periments. (M.H.R.)

7.
15021 A/CONF.15/P.936
TEMPERATURE COEFFICIENT MEASUREMENTS OF
LIGHT-WATER MODERATED HETEROGENEOUS
CRITICAL ASSEMBLIES. S. H. Levine, B. H.
$0.50 (OTS).
Prepared for the Second U. N. International Con-
A series of critical experiments on heterogeneous,
zirconium, light water moderated assemblies in simple
geometry is summarized. The assemblies consist of
both uniform highly enriched media and seed-blanket
slab arrays containing a natural uranium alloy blanket
in plate form. A review is given of the typical experi-
mental techniques and experimental data obtained in the
process of validating power reactor designs involving
these core media. The discussion is divided into two
parts; the first covering experimental techniques and
the second covering certain of the data obtained. (auth)

8.
2503 HW-51008
General Electric Co., Hanford Atomic Products
Operation, Richland, Wash.
TEMPERATURE COEFFICIENT OF A GRAPHITE-U-
RAIUM LATTICE. R. C. Lloyd and C. R. Richey. June 18,
52. $4.80 (ph OTS); $2.70 (mf OTS).
Material buckling measurements were made in a
graphite-U exponential pile at temperatures ranging from
24 to 406°C. The large (101-inch cube) exponential pile
was loaded with 1.36-inch diameter fuel elements of natural U.
The temperature coefficient, over this range is

\[
\frac{1}{B_m} \cdot \frac{\partial B_m}{\partial T} = -9.62 \times 10^{-4}/°C \text{ and } \frac{1}{\kappa_m} \\
\frac{\partial \kappa_m}{\partial T} = -4.58 \times 10^{-5}/°C.
\]
Lattice parameters were calculated and the values of \(\Delta n/\eta\) were determined which
permit a fit to the experimental results. (auth)

1959
9.
445 KAPL-M-WS-3
Knolls Atomic Power Lab., Schenectady, N. Y.
ON THE TEMPERATURE DEPENDENT PROPERTIES
OF AN 8 1/4 INCH SLAB REACTOR. J. A. Bielke,
R. G. Lace, and W. Skolnik. Sept. 3, 1958. 30p. Con-
tract W-31-109-Eng-52. $4.80 (ph OTS); $2.70 (mf OTS).
Experiments to determine the critical load of an
8 1/4 x 30 x 32 inch reactor as a function of moderator
temperature are described. The reactor contained
water, highly enriched uranium, and Zirconoy. The
metal-water ratio was 1.63 to 1 and a temperature
range of 79.5 to 613°F was covered in five steps.
(W.D.M.)

1960
10.
23481 APDA-139
THE DOPPLER EFFECT IN FAST NEUTRON REACTORS.
OTS.
Submitted to Cornell Univ.
A new analysis of the Doppler effect in fast neutron re-
actors is made. The effect is evaluated for U$^{235}$ and U$^{238}$ at
several neutron energies between 1 and 200 kev. A new
method is presented for U$^{238}$ at neutron energies below 30
kev where the previous method is shown to be invalid. The
new method is based on the assumption that the individual
resonances are isolated and do not overlap. The impor-
tance of inelastic scattering is evaluated and the effects of
other neglects previously made without complete justifica-
tion are studied in some detail. The newer data on low
energy resonance parameters and high energy cross sec-
tions are examined to determine the best choice for the
statistical distributions of neutron and fission widens and
the distribution of resonance spacing. A crude numerical
calculation for the Doppler temperature coefficient of re-
activity in the Fermi Fast Breeder Reactor gives about
\(-2 \times 10^{-4}/°C\) at 566°C. The results indicate that the Doppler
effect from U$^{235}$ can almost always be neglected as small
but in some of the larger reactors such as a large fast ox-
icide breeder U$^{238}$ may contribute a negative Doppler effect
of the order \(-10^{-4}\) which would be of considerable im-
portance, especially with respect to reactor safety. (auth)
1961


Experimental and theoretical work on the Doppler effect in thermal reactors is reviewed for U metal, UO₂, Th metal, and ThO₂. The experimental values of α, the fractional increase in resonance capture per °C, have a spread many times the quoted errors. The use of different slowing-down spectra contributes to the discrepancies. For U metal, approximate corrections are made to obtain the coefficient α₀ appropriate to a 1/E spectrum. The spread in the corrected values α₀ is smaller than that for α, but remains unsatisfactory. Other experimental difficulties arise in reactivity normalizations, in obtaining the statistical weight of samples, and from spurious temperature effects. Theory and experiment agree on an increase of α₀ with increasing surface-to-mass ratio and that this is caused by an increase in the contribution of lower-energy resonances to the Doppler effect. It is also in agreement with the theoretical interpretation of the radial dependence of the Doppler effect in a lump. However in the region of practical interest where the surface-to-mass ratio is small, α₀ is almost constant. Experimental evidence on the temperature behavior of α₀ is unsatisfactory but indicates that α₀ decreases with increasing temperature. Theory predicts that α₀ will vary approximately as T⁻¹ where T is the Kelvin temperature. In the case of non-uniform temperature distribution in a fuel element, both experimental and theoretical effort is needed. (auth)

1962


Doppler 11-17

Declassified version of NPCC/RPWP/P 66; AERE/R/M-132.

The temperature coefficients of reactivity of the BEPO, Windscale, and Calder reactors are calculated. The results are compared with experimental values. (auth)


The temperature coefficient of the Reactivity Measurement Facility was found to be 49 ± 1 µk/°C (1 µk = 10⁻⁶ Δk/k) in the range 15.4 to 17.8°C. The change in the net reactivity of a standard sample was -0.48 ± 0.02, -0.66 ± 0.03, and -0.78 ± 0.02 µk/°C in three measuring positions. These low values generally make temperature corrections insignificant. The above results are compared with previous determined values. This information developed in the RMF should be generally applicable to flux-trap-type reactors such as the Advanced Reactivity Measurement Facility (ARMF) and ARMF-II, now under construction. RMF was dismantled in April 1962. (auth)


The Doppler effect of Pu²³⁹ in a fast spectrum was studied. Following Goertzel, Lane and Bethe, and the formulation proposed by Nicholson, the Doppler effect of Pu²³⁹ at 1 to 200 kev was evaluated. At these energies, the total macroscopic cross-section μ(E) scarcely deviates from its mean value ⟨μ(E)⟩, and the contribution to the Doppler effect by the group of neutrons centered on energy E can be evaluated by Nicholson’s method “A.” The function Q(E) is calculated on the basis of findings regarding the resonances of Pu²³⁹. (auth)


The low energy total neutron cross section of Pu²⁴¹ was measured from 0.02 ev to 2 kev with the use of the Materials Testing Reactor (MTR) fast chopper. Curves and tables of the total cross section as a function of energy are given. Pronounced asymmetries were observed in several of the resonances, indicating the existence of interference in fission. The data were analyzed below 12 ev with the use of the Reich-Moore multilevel formula, under the assumption that the observed resonance asymmetries are due to interference in a small number of fission channels. A study was
conducted to determine the extent of the Doppler distortion of the resonances below 11 ev, at temperatures ranging from 0 to 2000 degrees centigrade. An estimate of the neutron strength function \( \langle t_r^4/D \rangle \) obtained from the average cross section in the key energy region is \( (0.85 \pm 0.15) \times 10^{-8} \). (auth)

18.

27734 (HW-73116(p.71-81)) THE FUEL TEMPERATURE COEFFICIENT OF \( k_a \) FOR PLUTONIUM-ALUMINUM FUEL. R. I. Smith (General Electric Co., Hanford Atomic Products Operation, Richland, Wash.).

The change in the reactivity of the Physical Constants Testing Reactor (PECTR) was measured as a section of Pu-Al fuel rods in the center test cell of the core was heated from room temperature to more than 400°C. Measurements were made with 1.88 wt % low exposure Pu-Al, with 2.07 wt % high exposure Pu-Al, and with dummy Al rods in the test cell. The signs and the magnitudes of the fuel temperature coefficients of \( k_a \) were obtained from the data. The effects of heating the fuel on the neutron multiplication of the lattice cells fueled with Pu-Al were found to be small in magnitude and negative in sign. The magnitude of the negative effect was seen to be a function of the concentration of Pu-Al in the fuel. The higher PuAl density material yielded a larger negative effect. The results indicate that for Pu-Al fuels of low Pu density, i.e., \( \approx 2 \) wt % Pu in Al, the major contributor to the fuel temperature coefficient is the Doppler effect in PuAl. (auth)

19.


The Doppler coefficient in thorium dispersed in graphite was measured in a reactor spectrum developed in the HTGR critical assembly, using a reactivity technique that compares the reactivity of a single cold element and a single hot element heated to 700°F. The considerable correction due to the small perturbation in neutron temperature resulting from the heating has been experimentally determined by auxiliary measurements. An activation technique has been used to check the results. In this activation technique, the thermal-base effect has also been experimentally subtracted by measuring the \( 1/v \) component of the captures at each temperature in vanadium. Measurements of the Doppler coefficient obtained by these two different techniques agree to within 10%. This agreement, demonstrated by applying both techniques for the first time in the same laboratory, shows that the previously reported spread of results and apparent disagreement between results of these two techniques are avoidable. The average Doppler coefficient for the temperature interval from 300 to 700°F, for a carbon-to-thorium ratio of about 60 in the thorium-bearing region, was measured to be \( 3.6 \times 10^{-4}/°C \) to within about 10% uncertainty. The measured results are in satisfactory agreement with calculated values. (auth)

20.


21.


22.


Isothermal temperature effects were measured in an ATR Critical Experiment core containing two strips of borated mylar film on each plate, no neck draw rods, and control cylinders at an initial position of approximately 100 degrees. Two sets of measurements were made because the first set of data indicated uncertainties in both temperature and reactivity measurements. Both measurements indicated slightly positive temperature coefficients initially. The first measurement indicated a change of sign at higher temperatures, while the second indicated a slightly less positive effect at higher temperatures. An additional measurement of isothermal temperature effects was made in an ATRCE Core with no borated plastic in the fuel region. Core reactivity was held by two fully inserted safety rods in Lobe Positions 1 and 9, and an additional measurement was made with a third safety rod inserted in Lobe 5. A reactivity loss of approximately 67 cents was obtained by increasing core temperature from 21.7 to 39.7°C with two safety rods inserted; a reactivity loss of approximately 67 cents was obtained by increasing core temperature to the same degree with the three safety rods fully inserted. (auth)

23.


Temperature effects on reactivity caused by heating various regions of the Advanced Test Reactor Critical Experiment Core were measured by pumping heated water separately through isolated flux trap and fuel regions. The data necessary to plot reactivity as a function of tempera-
ture were obtained, and isothermal temperature coefficients were measured to correct for bulk temperature rise. As expected, the results indicated a negative temperature coefficient, $-2.5 \frac{\text{C}}{\text{C}}$, in the fuel region and a positive temperature coefficient, estimated to be $+1.3 \frac{\text{C}}{\text{C}}$, in all nine flux traps. These measurements are appropriate for a core containing six neck shim rods per neck and having the control cylinders unbalanced to approximately simulate the reference power split. (auth)

24

9855 (TID-13139) REACTOR TEMPERATURE.
17p. (LADC-4938)
Beryllium oxide and graphite critical experiments operated in the temperature range 90° to 1200°F were analyzed with a simple gas model neutron scattering matrix. A comparison of room temperature and high temperature experiments showed no systematic trends clearly ascribable to the method of calculation. It was concluded that in this temperature range the gas model is sufficiently accurate for the analysis of this reactor type for the epithermal to intermediate flux spectra found here. (auth)

25

30p.
Theoretical values of resonance integrals and Doppler coefficients of U-238, Th, and Au were found using recently improved, yet efficient, calculational procedures. A range of effective scattering cross section per atom from infinity to very small values, and a wide temperature range were investigated. The mathematical representations and sources of physical parameters are reviewed. The Doppler coefficient was found to exhibit a maximum at an intermediate value of self-shielding, which is determined primarily by the properties of the lowest energy resonances. Two parameter fits were made to both resonance integrals and Doppler effect coefficients for practical rod sizes. Comparison with recent experiments showed significant lack of correlation with theoretical calculations for the thorium resonance integral and the U-238 Doppler coefficient. (auth)

1964

26

44930 (ANL-6936(p.17-38)) LIQUID-METAL-COOLED REACTORS. (Argonne National Lab., III.).
Doppler measurements in ZPR-III are reported for $^{235}$U metallic fuel mockups and uranium-plutonium test samples. Axial flux traverses were made by the foil-activation technique in a 600-liter core containing U, Na, and graphite with a length-to-diameter ratio of about 1 to 1. Spatially dependent worth studies were made on Rh reflectors in ZPR-IX, along with studies of B worth enhancement by H and measurements of central worth coefficients of Au, Zr, and ZrH$_2$. Development work on Ti-V cladding materials is described, and preliminary corrosion tests of V-base binary alloys are reported. Developments in fast-reactor fuel processing are also reported. EBR-II measurements made during the period included heat balance and transfer functions in approach-to-power experiments and power and flow coefficients. Analyses of the Ar cover gas system in EBR-II revealed that about 2200 PPM He and 20 PPM H$_2$ were present. Operation of the EBR-II power plant is summarized, Development and operation of the fuel-cycle facility and FARET are reported. (J.R.D.)

27c

Data were presented on spectral and reactivity measurements in an assembly of UF$_6$ and paraffin, the sensitivity of BF$_3$ detector banks, fast neutron penetration in water, the scattering kernel for polyethylene, lattice vibrational spectrum of magnesium, and the specific heat of magnesium. (C.E.S.)

28a

Experimental results of measurements of the Doppler temperature coefficient of reactivity of U$^{233}$ for fast-spectrum assemblies on the ZPR III are reported. (R.E.U.)

29a

Doppler reactivity measurements were made for uranium-238 heating in two-zoned core loadings of the ZPR-III fast critical assembly. The central zone of the first loading, in which the measurements were made, had the composition of a 5000-liter uranium monocarbide fast power breeder. The second central-zone loading differed from the first by replacement of 40% of the sodium with graphite. This change gave an increased Doppler reactivity effect and a second point for theoretical comparison. The design of the Doppler element and of the core loading, both of which were planned to minimize non-Doppler reactivity effects, is described. Supplementary
Doppler 30-38

Experiments which show that non-Doppler reactivity effects must represent at most a small part of the observed reactivity change are discussed. An experiment which tested the importance of resonance depression of the neutron flux incident upon the element, due to adjacent $^{235}$U in the central-zone loading, is described and results are given. Calculational comparisons with these experimental results using recently developed ANL cross sections are reported. They show close agreement with the first experiment and poorer agreement with the softer spectrum case. (auth)

30.


The experiment and theory are compared for some high-temperature clean cores; reactivity and power distributions were studied. Experimental results from both the pressurized Test Reactor and High Temperature Test Facility are discussed. (R.E.U.)

31.


32.


33.


A negative Doppler effect was observed in measurements on a metallic uranium sample (enriched to 9% in $^{235}$U) placed in a typical fast-reactor spectrum which has a median fission energy of 195 kev. The plausibility of the negative sign is supported on theoretical grounds, although with the use of standard analytical techniques and the limited number of resonance parameters at present available, it cannot be calculated for this spectrum. The value for $\rho/\rho_0(dp/dt)$ was found to be $-5.86 \times 10^{-5}$/K. Measurements were made on $^{232}$Th and the data on $^{232}$Th, previously published, were extended to 920°C. The agreement between analytical and experimental values for the latter materials is good. (auth)

34.


Doppler effect measurements for $^{232}$Th, $^{233}$U, and $^{234}$U performed in the AETR fast spectrum are presented and compared with theoretical results. (C.E.S.)

35.


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

Calculations of the temperature coefficients of reactivity in the high-flux charge are compared with measurements of the coefficients in both a subcritical facility and a zero-power critical facility and with measurements in the reactor. (D.C.W.)

36.


37.


The report presents experimental measurements of bucklings, flux fine structure, and fission rate distributions in graphite-modulated lattices fueled with plutonium/uranium metal at temperatures up to 400°C in the subcritical assemblies SCORPIO I and SCORPIO II. The experimental techniques employed are described in some detail. The accuracy of the experimental measurements appears to be adequate for testing methods of calculation being developed for the calculation of reactivity and temperature coefficient of reactivity for power reactors containing plutonium and uranium. (auth)

38.


Assembly 43 is the second two-zone core built in the ZPR-III facility. The central zone and the low-density axial blanket were designed to simulate a large breeder reactor. Criticality was achieved by an annular driver, while an interposed buffer modified the spectrum of the neutrons diffusing from the driver to the central zone. Experiments included sodium void coefficients for central substitutions and for full-length axial measurements. Other reactivity measurements included central reactivity coefficients and core-length column substitutions in the central zone and the annular driver. Spectral indices were measured, and reaction-rate traverses and activation experiments were performed. The reactivity change due to the Doppler effect caused by heating a large, depleted, uranium oxide sample was also determined in the experimental program. The experimental results indicate that the measurements can be used for checking calculations of dilute fast cores, as well as for evaluating the two-zone experimental technique. (auth)


The efficacy of the Doppler effect as a shutdown agency in power excursions is discussed, and recent work on Doppler evaluations are reviewed and some thermal calculations given. The programming of control rod withdrawal is also reviewed. Nuclear data for 25, the age of fission neutrons to 1.46 ev in water, and neutron thermalization are discussed briefly. (D.L.C.)

Doppler 39-48


Four experimental and two theoretical studies of the Doppler coefficient of UO2 fuel were compared. The results appear to divide into two distinct groups with about the same dependence on surface-to-mass ratio, but separated by a difference of ~15%. (auth)


The change in k(D,T) of a heterogeneous lattice caused by a uniform change in the temperature of the fuel was measured, using the Physical Constants Testing Reactor. The test lattice was moderated with graphite and fueled with concentric-tube elements of slightly enriched uranium metal. The temperature of the fuel was varied from 297 to 1241°C. The change in k(D,T) with high temperature was nonlinear and could be represented by the relation: k(D,T) = C(α(T − T0) + βUT90 + γUT120), where T is in degrees Kelvin. The experimentally measured values of the constants were α = (−0.303 ± 0.004), β = (−0.120 ± 0.004), γ = (−0.085 ± 0.004). The unit functions, U, represent the changes in k(D,T) caused by the isothermal volume expansion of the fuel element when the uranium metal undergoes transformations in its crystal structure from alpha to beta and from beta to gamma phases. The term C is a normalization factor related to the lattice under study. The reactivity techniques employed here are shown to be four times more sensitive than activation methods for determining the functional relationship between the effective reactivity integral of a fuel element and the temperature of the element. The constant, α, was experimentally separated into two components: α1 = (−0.240 ± 0.04), which is associated with the average interior temperature of the fuel, and α2 = (−0.068 ± 0.04), which is associated with the temperature of the surface of the fuel. This separation allows treatment of nonuniform temperature distribution in the fuel. (auth)
Doppler 49-55


The Doppler coefficient of 235U was measured in the ZEBRA CORE 45A, a zoned reactor containing a central region whose composition represents a large sodium-cooled fast power-breeder reactor having 235U 239Pu mononuclear fuel. The reactor regional parameters are tabulated. The reactivity coefficient was found to be approximately -3 x 10^-4°C. (L. B. S.)


Measurements using a central hot loop in Zebra Core 5 are described. Results are given for the Doppler coefficient found in a number of assemblies with PuO2 and 16% PuO2/84% depleted UO2 pins, loaded with different combinations of steel, sodium or void pins. The mixed oxide results are in general about 20% more negative than was calculated using the FDO data set, but agreement is good if the plutonium contributions in the calculations are omitted. The small positive Doppler coefficient calculated for 238Pu was not observed, and two measurements indicated instead a small negative effect. The Doppler effect in the mixed oxide systems was found to vary approximately as 1/T. The results from the empty loop and non-fissile assemblies indicate either a small negative Doppler effect in steel or alternatively the presence of an unexplained expansion effect. (auth)


The effect of resonance overlap on multigroup resonance cross sections and the Doppler coefficient is examined. Consistent definitions are given for the group flux and effective cross section used in a homogeneous mixture. The validity of the approximations is discussed for typical fast reactor cores, and an example is presented of a correction for self-overlap. The calculation of reactivity coefficients by means of first-order perturbation theory is interpreted, and it is shown that the same result is obtained with either the TRUE FLUX or the I/E FLUX method of defining the effective cross sections; however, the TRUE FLUX method is the one recommended. Approximate expressions for the change in the effective absorption cross section due to Doppler broadening of resonances are developed, and con-
56.

38427 (ANL-7120, pp. 603-9) MEASUREMENT AND ANALYSIS OF DOPPLER EFFECT IN PLUTONIUM-FUELED FAST REACTOR ASSEMBLIES. Fischer, G. J.; Meneley, D. A.; Hwang, R. N.; Groh, E. F.; TIlI, C. E. (Argonne National Lab., Ill.). Measurements made with the ZPR-3 fast critical assembly are described. Zone-loading procedures were used to produce a mockup of a large fast power breeder using \(^{238}\text{Pu}, \(^{239}\text{Pu}\) monoboride fuel, Na coolant, and stainless steel clad and structure. In another assembly 40% of the Na cans were replaced by graphite. The measurements indicated that the positive Doppler reactivity contribution of \(^{238}\text{Pu}\) in the fuel and in the neutron-energy spectrum of a large power breeder reactor is small in magnitude compared to the strong negative Doppler effect of \(^{239}\text{Pu}\). Experimental results are compared with theoretical calculations; experimental and theoretical uncertainties are discussed. It is concluded that much more work must be done with the fissionable isotopes in varying fast reactor spectra. (A.G.W.)

57.


58.


59.


Spherical samples, 2 cm in diameter, of \(^{235}\text{U}, \(^{233}\text{U}, \) and \(^{239}\text{Pu}\) were irradiated in turn at various temperatures ranging from 170 to 770°C in a central cavity of a spherically symmetric B–Be photo neutron source. The \(^{235}\text{U}(n,\gamma)\) reaction rate was measured by counting the \(^{238}\text{U}\) activity produced, and the (n,f) reactions were monitored by counting the fission neutrons emitted. The results obtained for the dependence of reaction rate on temperature were compared with those from a computer calculation based on a program developed by Brissend and Durston. (ANL-7050, p.31 (1965)). (auth)

60.


Measurements of the Doppler effect in resonant neutron capture were made for samples having a nonuniform temperature distribution. These measurements were made on Th and \(\text{ThO}_2\) rods of approximately \(\frac{1}{2}\)-in. diameter. An activation technique was used, and the samples were exposed in a cadmium thimble at the center of a pool research reactor. The activated 

61.


The equipment and method used for activation measurements of the Doppler effect in the fast-neutron spectrum of the Mixed Spectrum Critical Assembly are described. Measurements were made on natural and enriched U metal foils. Results indicate a significant underprediction of the Doppler effect. This discrepancy is believed to be at least partly due to the calculated neutron spectrum being substantially too hard. Random and systematic errors in the measurements are discussed. The technique requires careful control of foil position and counting procedure, but reproducible measurements can be performed to approximately 10% of the magnitude of the observable Doppler effect. (A.G.W.)

62.

15732 (TID-21654) DOPPLER COEFFICIENT OF THORIUM OXIDE. Quisenberry, K. S.; Knolls Atomic Power Lab., Schenectady, N. Y. (General Electric Co.). Contract W-31-109-eng-52. 16p. Dep. mn. CFSTI $1. 00 cy. 50. 56 mm. The neutron Doppler coefficient is briefly discussed. The method of measurement for ThO\(_2\) in the KAPL Thermal Test Reactor is briefly reviewed. The results are given. (M.O.W.)
Doppler 63-68


From Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, Argonne, III.

A study is made of the effect on the Doppler coefficient from plutonium-239 and of the present uncertainties in the data from 0.1 to 10 keV and, in particular, in the resonance parameter data in the unresolved resonance region. Also some of the approximations involved in the methods for comparing calculations with experimental data on the sodium coefficient are examined. (J.F.P.)


Improved data-gathering techniques, consisting largely of providing a servo-controlled regulating rod for the reactivity measurement fully automated with the oscillator system, improved the precision of Doppler measurements in the ZPR-6 by better than an order of magnitude. The improved techniques were used to study expansion effects and the temperature dependence of the Doppler effect. Data are summarized on the effects of environment of the sample in the reactor on the measurements, the $^{233}\text{U}$ Doppler effect as a function of temperature, reactivity changes for $^{235}\text{U}$ oxide with freely expanding and axially constrained samples as a function of temperature, reactivity changes for various isotopic ratios of $^{233}\text{U}$ to $^{239}\text{U}$ as a function of temperature, and reactivity effect of expansion of U metal. (A.G.W.)


Temperature coefficients of slightly enriched $\text{UO}_2-\text{H}_2\text{O}$ lattices were measured in the Oschni Critical Facility and analyzed in terms of critical parameters. The lattices studied were mainly those of 2.5% enriched $\text{UO}_2$, 10 mm in diameter and clad in 0.8 mm thick Al tube, and with water to fuel volume ratios of 2.5 and 1.5. Analyses were made with Deutsch’s four factor critical equation and also with three group one dimensional diffusion calculations. The result shows the importance of the influence of the reflector, which is the largest positive contribution to the temperature coefficient, being about $1 \times 10^{-4} \Delta k/k/°C$. This was experimentally demonstrated by measuring the coefficient for a core with an inner reflector water gap. Analyses also show that the discrepancy between experiment and calculation, which increases from 0.2 to $0.7 \times 10^{-4} \Delta k/k/°C$ as the water to fuel volume ratio decreases, may be mostly resolved by applying corrections for the space-dependent spectrum within the cell using the THERMOS code and for the effect of Dancoff factor with the TUS-ZUT code. Calculation with these corrections applied reproduces experiment within an accuracy of about $0.2 \times 10^{-4} \Delta k/k/°C$ over the temperature range of 20 to 70°C. (auth)

68. 20198. (AEEW-R-164) FURTHER BUCKLING MEASUREMENTS ON A HEATED GRAPHITE LATTICE FUELLED WITH 0.8 PERCENT $\text{PuO}_2/\text{UO}_2$ CLUSTERS. Wilson, D. J. (Atomic Energy Establishment, Winfrith (England)). Oct. 1965. 26p. Dep. BIS 50.90. HMSO 4s. 6d.

The buckling of a graphite moderated stack fueled with 0.8% $\text{PuO}_2/\text{UO}_2$ 21 rod clusters in a 14" lattice pitch was measured by the exponential method. The stack was built in the SCORPIO I facility which enabled the experiments to be made at 200 and 350°C as well as at ambient temperature. The experimental results are shown to be in very satisfactory agreement with predictions using the ARGOSY IV method. (auth)