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REACTOR PHYSICS DATA
FOR THE UTILIZATION OF PLUTONIUM
IN THERMAL POWER REACTORS

L. C. Schmid
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Special Plutonium

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FOR THE UTILIZATION OF PLUTONIUM IN THERMAL POWER REACTORS

By

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May 1968

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REACTOR PHYSICS DATA
FOR THE UTILIZATION OF PLUTONIUM IN THERMAL POWER REACTORS

ABSTRACT

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

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I. INTRODUCTION

In many ways the problems connected with plutonium utilization in commercial reactors are the extension of those which would occur for highly irradiated uranium fuels since the latter contain appreciable amounts of plutonium. However, the physics characteristics of plutonium enriched reactors are inherently more difficult to calculate than those for uranium enriched reactors. Problems which occur in predicting the behavior of uranium fueled reactors are also present in the plutonium system because the fuel of interest is a mixture of the oxides of plutonium and uranium. Thus, to begin with, the problems involving calculations of reactions for epithermal neutrons with uranium are present. The presence of plutonium adds to these problems because of the complex cross section behavior of the plutonium isotopes in the neutron energy range below 3 eV. Capture of neutrons by ^{239}Pu produces an absorber, ^{240}Pu , which has a larger cross section than the ^{236}U produced by captures in ^{235}U . In addition, the amount of ^{240}Pu produced from the ^{239}Pu is dependent on α^{239} the ratio of neutrons being captured to neutrons causing fission. This ratio is a strong function of the neutron energy in the thermal energy region. By comparison, the similar ratio for ^{235}U is relatively independent of neutron energy. Successive captures after ^{240}Pu produces first a fissionable isotope, ^{241}Pu ,

and then an absorber, ^{242}Pu . The isotopes ^{239}Pu and ^{241}Pu have a large resonance at ~ 0.3 eV and isotopes ^{240}Pu and ^{242}Pu have giant resonances at 1.0 and 2.7 eV respectively. These resonances lie in the region of neutron thermalization in H_2O moderated systems and as a result the problem of accurately computing thermalization events for plutonium fueled reactors is more difficult.

In general, the resonance structure of the plutonium isotopes and the energy dependence of α^{239} makes the problem of calculating plutonium systems more difficult than uranium systems. The complex cross section behavior causes moderator temperature, fuel temperature, and void coefficients of reactivity to be larger for plutonium systems and adds to the difficulty in adequately computing these coefficients. The magnitude of the ^{239}Pu and ^{240}Pu cross sections combined with a non- $1/v$ behavior in the thermal and near thermal energy region increases the difficulty of calculating power distributions relative to that for uranium systems. The plutonium plays the important role in the burnup behavior of plutonium-enriched reactors, and the presence of several fissile and fertile isotopes add to the complexity for computing the transmutations of plutonium isotopes.

Problems connected with the kinetic behavior of plutonium systems are not much different from those of uranium systems. The effective delayed neutron fraction $\hat{\beta}$ and neutron lifetime ℓ for plutonium systems are smaller than those for uranium systems. Thus, a more rapid increase in power would result initially for a given increase in reactivity. However, the variation of $\hat{\beta}/\ell$ with irradiation is smaller for a plutonium system. Also, at prompt critical the rate of power increases, which is typified by β/ℓ , is

approximately the same or less than that for a uranium system.

A major USAEC program for studying the use of plutonium fuel in current United States power reactors is the Plutonium Utilization Program^(1,2) conducted by the Pacific Northwest Laboratory of Battelle Northwest. The general objective of this program is to develop the base technology required for safe and economical recycle of plutonium in thermal power reactors. To accomplish such a general objective, it is necessary to conduct work in many areas. Some of the work involves fuel design, fabrication and irradiation studies, chemical processing studies, studies of the economic utilization of plutonium in current reactors, as well as reactor physics studies.

The main objective of the physics part of the program is to evaluate, and improve where necessary, calculational methods and approximations to a point where reliable engineering calculations can be made for predicting the nuclear characteristics of plutonium-fueled thermal reactors. Extensive cross section and reactor physics data are required to evaluate the calculational methods. These data have come from experiments conducted at the Pacific Northwest Laboratory as well as at other laboratories. Cross section data have been obtained from numerous differential and integral measurements. The precision of these data is continuously being improved as a result of new measurements and evaluations.

The use of plutonium in H_2O reactors has been studied under programs in addition to the USAEC Plutonium Utilization Program. The Saxton Plutonium Program⁽³⁾ is a joint USAEC-EURATOM program with the purpose of developing information about the use of plutonium-enriched fuel in

pressurized water reactors. The work involves zero burnup studies with critical facilities and a burnup experiment in the Saxton reactor in which ~40% of the core is plutonium fuel. Other USAEC-EURATOM programs have been conducted at the General Electric Vallecitos Atomic Laboratory. These studies included integral spectral measurements in a driven sub-critical assembly under various conditions of temperature and pressure,⁽⁴⁾ and a fuel burnup experiment⁽⁵⁾ conducted in the Vallecitos Boiling Water Reactor (VBWR). The Empire State Atomic Development Associates, Inc. and Westinghouse have funded a series of experiments⁽⁶⁾ designed to study the utilization of plutonium in pressurized water reactors. The Edison Electric Institute is funding a program⁽⁷⁾ at General Electric and Westinghouse to study the problems of plutonium recycle in a boiling water and a pressurized water reactor respectively. There is also interest in plutonium utilization in European countries. One of the more significant projects is the Euratom Plutonium Project⁽⁸⁾ being conducted jointly by Belgo Nucléaire and CEN.

The purpose of this report is to summarize experimental data which has been and is currently being obtained in the United States for the purpose of utilizing plutonium in thermal power reactors. The scope of the report has been limited to data required for studying the reactor physics aspects of plutonium recycle in thermal power reactors. Data for lattices moderated with H_2O , D_2O , and graphite are included. However, the discussions are centered around the use of plutonium in H_2O reactors because these reactor types are of most interest in the United States. Plutonium data obtained in other programs not directly connected with present-day commercial reactors has not been summarized. Thus, data such as that obtained with homogeneous solutions⁽⁹⁾ and Phoenix cores⁽¹⁰⁾ are not

included although they add to the understanding of the fundamental reactor physics problems connected with plutonium utilization.

An attempt has been made to provide a ready reference for what data exists and where it can be found. How the data helps to understand problems connected with calculating H_2O -plutonium systems is explained and some needs which still exist are mentioned. Comments about the precision of basic cross section data are included in the summary of experimental data since they are an important aspect of understanding the problems. Criticality, reactivity coefficients, kinetics, and burnup data are referred to and each section is followed with a synopsis. Conclusions are made about the use of the data in evaluating methods and cross sections for H_2O moderated reactors.

II. BACKGROUND

The physics technology of plutonium reactors is an outgrowth of the technology developed for uranium reactors. In this section, background information is provided concerning the relevance of the uranium technology to the development of experimental data for plutonium; in the areas of cross section measurements, reactor physics measurements, and theoretical methods. It might appear that the latter is out of place in this report until it is remembered that most data are obtained primarily to evaluate and improve the methods.

Cross Sections

The development of knowledge of fundamental neutron cross section information on plutonium isotopes has historically lagged behind that of the uranium isotopes. The reasons for this have

primarily been lesser motivation and lack of availability of adequate isotopic samples. No significant extension of experimental technique over that required for uranium isotopes is required to obtain differential nuclear data of comparable precision. The differential data for ^{239}Pu has for many years been of a precision comparable with that for ^{233}U and ^{235}U although there is for ^{239}Pu a unique lack of a reasonable number of confirmatory measurements of most of the data below 1 eV. An important problem with nuclear data for ^{239}Pu and ^{241}Pu is the relatively large uncertainties which must be attached to the absolute value of the partial cross sections and neutron reproduction factors. These uncertainties are due to the resonance structure of the cross sections of these isotopes in the thermal region. Whereas integral measurements in near-thermal neutron spectra can yield 2200 m/s values of alpha or eta with high precision for the uranium isotopes, the comparable precisions for the plutonium isotopes are factors of several worse. The interpretation of such measurements for ^{239}Pu and ^{241}Pu requires not only that the cross-section shapes be more accurately known than for uranium but also requires a precision of knowledge of the shape of the thermal spectrum which is unattainable except for completely thermalized spectra. The knowledge of the fundamental constants for ^{239}Pu and ^{241}Pu has been significantly improved with the recent completion of measurements of eta for mono-energetic neutrons.⁽¹¹⁾ However, consideration of these values along with the other nuclear data yields discrepancies which indicate that the values of the fundamental neutron cross sections of the plutonium

isotopes are still significantly more uncertain than are the values for the uranium isotopes.

Lattice Studies

Historically, the development of experimental techniques for uranium systems have progressed from the very simple to the very complex. Early studies of uranium systems were often limited to exponential assemblies by lack of material, so that only bucklings measurements could be made. As more materials became available, fully critical cores could be assembled. With the resulting increase in available flux level, measurements of cell and core fine structure became practical. Many detailed studies were made of reaction rates in fuel assemblies and in moderator regions, to obtain data for testing computational methods.

The same progression has occurred for plutonium systems with a quicker arrival at studies of specific problems characteristic of the use of plutonium in boiling and pressurized water reactors. There has been one marked difference in the progression however. The detailed fine structure measurements within the fuel have, in general, not been made in the plutonium-enriched systems. One reason for this omission has been the toxic nature of plutonium which dictates handling precautions that increase the cost of such measurements by factors of 5-10 and which requires additional confining materials that in some cases increase the experimental uncertainty. A second reason has been the development of very sophisticated computational methods which are capable of calculating the detailed flux fine structures

with about the same precision as measurements would provide. A third reason has been the very small supply and the high cost of highly purified plutonium isotopes for use in studies of individual reaction rates.

As a result, the measurements have been confined mainly to the important and more easily measured plutonium lattice data of critical mass, bucklings, power distributions obtained by gamma-scanning fuel rods, and reactivity coefficients. The fact that few reaction-rate experiments have been performed to date does not imply that none are needed. On the contrary, some of the lattice correlations have shown discrepancies which can only be resolved by careful study of individual reaction rates in thermal and epi-thermal energy regions. A few measurements of reaction rates are being made now and more effort will be applied to this type of measurement as required for evaluations of calculational methods.

Theoretical Methods

The methods utilized in calculating the reactor physics parameters of plutonium systems are basically those methods developed originally for uranium systems. The plutonium systems have necessitated certain modifications to these methods but, for the most part, the basic techniques remain unchanged. However, most of the empirical methods (e.g., Westcott spectrum notation, etc.) which have been successfully applied to uranium-fueled reactors cannot be used successfully⁽¹²⁾ for plutonium-fueled reactors. Certain of these empirical techniques have been "massaged" to fit observed data, but,

in general, these techniques have not been very successful for pre-experiment predictions.

The basic methods in general use at the present time are based upon Fourier-transform multi-group calculations of the neutron spectrum, followed by few-group calculations of leakage. The spectral calculation for non-thermal energy groups uses either the P1 or the B1 approximation in solving the neutron transport equation, (i.e. MUFT,⁽¹³⁾ or GAM^(14,15) codes). The spectral calculation in the thermal energy region is based upon integral transport theory and uses a bound scattering kernel (i.e., the THERMOS code⁽¹⁶⁾). Certain of these codes have been combined into automated packages, such as the LASER,⁽¹⁷⁾ HAMMER,⁽¹⁸⁾ and the HTH⁽¹⁹⁾ codes. Unfortunately, the cross sections used in these codes differ from laboratory to laboratory, and it has been difficult to make meaningful intercomparisons of the results of theory-experimental correlations performed at different laboratories.

In order to more effectively use the experimental data to evaluate the computational methods in a systematic and consistent manner, a specific calculational approach was chosen at Pacific Northwest Laboratory as a base calculational technique. This base technique uses the HRG⁽¹⁵⁾ and THERMOS⁽¹⁶⁾ codes to compute the slowing-down and thermalization spectra to obtain few-group constants for use in a one-dimensional neutron diffusion calculation⁽²⁰⁾ of the reactor loadings. A comprehensive set of other multi-dimensional diffusion and transport theory codes has been compiled or developed to provide a specific analytical capability. For more detailed reactor

calculations and to obtain results to compare with other calculational techniques when data is difficult to obtain experimentally, a general purpose Monte Carlo program⁽²¹⁾ is used.

In order to assure the use of a consistent cross section library and to have available various cross section values to compare against one another, the BNW Basic Library System⁽²²⁾ has been used at PNL since around 1960. This library has other codes connected with it which allow revision and addition of data and which process data into forms usable by reactor physics analysis codes. The data in the Library is made available to all of the general reactor physics codes in use at Pacific Northwest Laboratory to allow consistent and systematic evaluation of calculational methods to be made as well as evaluations of different cross-section values. More recently, a parallel capability has been provided which makes use of the Evaluated Nuclear Data File, ENDF/B and its processing codes.⁽²³⁾ This capability will allow a systematic intercomparison of theoretical methods (e.g., HAMMER to HTH) and cross section compilations (e.g., BNW Master Library to ENDF/B).

Theory-experiment correlations performed at Pacific Northwest Laboratory include nearly all of the plutonium-fueled lattice experiments performed to date. Since the fuel which is of most importance to a central power station is a mixture of plutonium and uranium, it is important to understand the calculational problems involved with uranium systems as well as systems which contain plutonium alone in

order to obtain a fundamental understanding of the calculational problems for the mixture. For this reason, results from enriched uranium experiments conducted in support of the Yankee reactor have also been included in the correlations. The results of these studies have pointed out major problem areas in these methods and emphasize the need for additional experimental data.

III. EXPERIMENTAL DATA

Experimental data used in the evaluation of computational methods developed for plutonium systems is available from a variety of sources. Sources of data which include basic cross sections, reactor physics parameters, and fuel isotopic compositions, are included in this section. The cross section information is available from differential and integral measurements. Reactor physics information about static and dynamic parameters is obtained from lattice studies. The variations of these parameters with irradiation are obtained from reactor studies along with fuel isotopic compositions.

The sources for lattice information are separated according to whether the data applies to H_2O , D_2O , or graphite moderator. One exception is studies, conducted in the Physical Constants Testing Reactor, which provide data for all three moderators. Since the data from this facility is unique in that it applies to the infinite medium lattice and the results are predominately for graphite moderator, the combined work is summarized under Experiments Using D_2O or Graphite. Some results obtained with H_2O moderator have been summarized where appropriate. The original references should be used to assure the accuracy of the results. In addition, descriptions of fuel elements used in all the experiments are included in the Appendix.

A. Cross Sections

Correlations of reactor physics data for thermal-reactor plutonium-recycle systems is quite sensitive to the spectrum-averaged cross sections in the thermal and near-thermal part of the neutron spectrum. The uncertainties in the shapes and absolute values of the low-energy nuclear data for the plutonium isotopes are comparable with and correlated to the uncertainties which need to be assigned to the calculated neutron spectra. Least-squares adjustments of the 2200 m/s constants of the fissile nuclei⁽²⁴⁻²⁷⁾ have yielded precisions for the plutonium isotope constants which might appear to be nearly adequate for reactor analysis. It is apparent, however, that the quoted precisions do not have the significance which is frequently attached to them if it is recognized that the values and precisions result primarily from input values which have been derived from the analysis of reactor spectra measurements. These measurements have been made in reactor systems which have frequently been as complicated to analyze as the plutonium recycle systems to which the results are to be applied. Thus gross disagreements might be expected to exist in the calculated reaction rates in a system analyzed with a Westcott flux⁽³⁰⁾ with one cross section shape and those resulting from a spectrum calculated by a thermalization code using the same cross section shape.

The existing least-squares evaluations of 2200 m/s constants have not included information about the cross-section shapes which were used in the interpretation of the integral experiments. In the absence of this information, the procedure followed at PNL has been to define the

cross-section shapes of each of the fissile nuclei according to some evaluation. These same shapes have been used for the past several years. For each of these nuclei, several sets of low-energy cross section data are kept in the BNW Master Library with the same cross-section shapes normalized to the output values of different evaluations.⁽²⁴⁻²⁹⁾ These normalizations do not have the significance usually implied and are not necessarily a self-consistent set of values because of the shape dependence. They do, however, allow the effects of cross-section uncertainties to be defined to some extent by using different normalizations in identical calculations.

For the cross sections above the thermal region a different philosophy has been applied. Because of their much lesser importance in thermal systems these cross sections have been updated in the Master Library whenever apparently improved or more complete data have become available. Although different resonance parameter information has sometimes caused apparently significant changes in calculated resonance integrals, the real significance of such changes is difficult to define. This situation arises because experimental and calculated values are quite sensitive to the cross sections and the details of how the slowing down and thermal neutron distributions join, especially for ^{239}Pu and ^{241}Pu .

The following paragraphs define qualitatively the sources of evaluations of the shapes of those cross sections most important to the analysis of plutonium recycle which are in the BNW Master Library and used in the analyses. Also, the uncertainties in the differential

data most important to the analyses are pointed out.

^{235}U Cross Sections. The ^{235}U cross sections are defined at point values up to 0.5 eV. The shapes of the cross sections at low energies were derived by Liikala⁽³¹⁾ based primarily on an evaluation by Leonard⁽²⁸⁾. Unpublished investigations⁽³²⁾ have shown that these energy variations differ significantly from the shapes used at Chalk River^(30,33). For even a quite well-thermalized spectrum ($r \approx 0.019$), we obtain effective values of α which differ by $\sim 2\%$ using the same Westcott conventional flux. Thus the interpretation of Pu recycle systems involving ^{235}U are subject to uncertainties greater than this due to cross-section shape alone. Uncertainties due to uncertainties in calculated thermal spectra have been shown to be much less than this⁽³²⁾. Monoenergetic η and integral α values yield $\bar{\nu}$ values for ^{252}Cf (the standard for $\bar{\nu}$ measurements) which can be used used to check the consistency of the data. The use of recent measurements⁽¹¹⁾ result in a value of 3.80 with a precision approaching ~ 0.6 percent using ^{235}U data.

^{239}Pu Cross Sections. These cross sections are defined as point values for energies up to 3.0 eV. The shapes are those determined by Leonard⁽³⁴⁾ in 1959. This evaluation preceded the results obtained by Safford, et al⁽³⁵⁾ for the total cross section. Incorporation of these results and a smaller scattering cross section based on an improved analysis would yield slightly different shapes. Since no further evaluations based on simultaneous fitting of $\sigma_f(E)$, $\sigma_T(E)$ - $\sigma_s(E)$, $\eta(E)$ have been made, the 1959 shapes have been retained. The

effects of uncertainties in shapes of the ^{239}Pu cross sections have been studied by Jenquin, et al.⁽³⁶⁾ Jenquin found significant differences in effective values of η between the BNW cross sections and the shapes used by Schmidt⁽³⁷⁾. These differences were typically 1 to 2% for the systems considered. It is not clear that these differences are representative of realistic shape uncertainties since Schmidt's cross-section shapes were apparently not subjected to critical evaluation. Unpublished calculations by Leonard, et al.⁽³²⁾ have shown significant uncertainties in effective values of cross sections due to uncertainties in calculated thermal spectra even for rather well-thermalized systems ($r \approx 0.019$). Such uncertainties, due to spectra alone, were typically 1% in α and greater than that for reaction rates.

Analyses of PRTR burnup data⁽³⁸⁾ and an NRU integral measurement^(39,32) have indicated a 2200 m/s value of α of about 0.35. Combining this result with Smith's monoenergetic η value⁽¹¹⁾ (2.114 ± 0.007) and accepted data on relative values of $\bar{\nu}$ gives values of $\bar{\nu}$ for ^{252}Cf which are 1-2% less than the value of about 3.80 obtained by identical procedures for ^{233}U and ^{235}U . The source of this discrepancy is unclear. The 2200 m/s value of η of Smith appears low due to a comparison of his shape from 0.025 eV to 0.06 eV and the shapes of our cross sections. There is also an uncertainty in the experimental $\bar{\nu}$ ratios due to differences in shape of fission neutron spectra which have probably not been adequately evaluated. Nevertheless this discrepancy in 2200 m/s values

has been considerably reduced from the similar discrepancies of 5% which existed previously⁽²⁸⁾.

²⁴¹Pu Cross Sections. These cross sections are defined by point values in the BNW Master Library up to an energy of 2.0 eV. The fission cross section shapes were determined by the data of Leonard, et al.⁽⁴⁰⁾ which have not been improved upon significantly⁽⁴¹⁾ in recent years. The absorption cross section was based⁽³¹⁾ on preliminary data from the MTR, and a new analysis based on the now-existing data would probably yield a significantly different shape⁽⁴¹⁾. The ²⁴¹Pu absorption cross section is probably uncertain by several percent due to 1) uncertain sample assays, 2) isotopic impurity, and 3) possible small-angle scattering effects due to the use of oxide samples in the total cross-section measurements. Nevertheless, an analysis similar to that of ²³⁹Pu using Smith's monoenergetic eta value⁽¹¹⁾, an alpha of about 0.36, and accepted data on nu bar ratios indicates a nu bar value of ²⁵²Cf of about 3.77 which is within 1 percent of the value obtained for ²³³U and ²³⁵U. This agreement may be fortuitous.

²⁴⁰Pu Cross Sections. The ²⁴⁰Pu absorption cross section used on the BNW Master Library is based on calculations using 1 eV resonance parameters and is normalized to a 2200 m/s value of 281 barns. Only two precision measurements of the total cross section of ²⁴⁰Pu exist and these differ by 5.3 percent. Of more importance, a discrepancy in the radiation widths of the ²⁴⁰Pu resonance was recently discovered.

Measurements made for the resonances above 1 eV have been interpreted to yield average radiation widths of 18- to 23-meV^(42,43) whereas the accepted value for the 1 eV resonance is about 31 meV.⁽⁴⁴⁾ This discrepancy is not serious per se in calculating reaction rates in thermal reactors but implies that the value assigned to the 1 eV resonance may be significantly in error. Further measurements in the thermal and 1 eV resonance region with modern techniques are extremely desirable. Calculated reaction rates in thermal systems for ^{240}Pu must be considered uncertain by 10-20% at the present time due only to cross section uncertainties.

^{242}Pu Cross Sections. The absorption cross sections used for ^{242}Pu have been subject to a great deal of misinterpretation. Integral experiments⁽⁴⁵⁾ at Chalk River have been interpreted to give a 2200 m/s value of 19.8 ± 1.1 barn. The normalization used in the past in the BNW Master Library has been 30 barns due to an undue consideration of an experimental value of 30 ± 10 barns⁽⁴⁶⁾. Monoenergetic measurements have recently been made at INC in the thermal and resonance region⁽⁴⁷⁾. These unpublished measurements, made with an oxide sample, have been interpreted to give 2200 m/s absorption cross-section values of 22- to 28-barns depending on the interpretive uncertainty of the possible small-angle scattering effects. In addition, the resonance integral calculated from the measured resonance parameters is 10-20 percent lower than that interpreted from the NRU integral experiments⁽⁴⁶⁾. A rather intermediate description of the ^{242}Pu cross sections has also been used recently, at PNL, which has a normalization of 26.0 barns and the measured resonance parameters.

B. Lattice Studies

Low power, uniform lattice experiments have long been used to provide the data needed to assess the adequacy of the calculational methods employed in the nuclear design of power reactors. It is important to be able to accurately predict the physics characteristics of uniform lattices, since if the calculational methods cannot accurately predict the properties of a uniform lattice, they are unlikely to be able to predict accurately the characteristics of the power reactor. Therefore, the studies with plutonium-enriched fuel have had as their objective the determination of what are the problem areas in the measurement techniques, the theoretical methods, or the cross section values, which prevent close agreement between measurement and calculation.

The uranium-fueled reactor programs have generated a wealth of experimental data with which to test the methods. The data for clean uniform lattices fueled with slightly enriched uranium cover around 250 and 85 lattices for metal and oxide fuel, respectively. These experiments include measurements of the fine structure parameters of the lattice cell (e.g., ρ_c^{28} , δ^{28} , etc.) as well as those parameters which describe the gross reaction behavior such as bucklings and critical masses. These gross reaction data have provided the basis for evaluating the adequacy of theoretical methods and the cross section data. As a result, the methods and cross sections have been improved to a stage where reliable engineering calculations are being made. Consequently, reactor designers do not usually perform

prototypical critical experiments prior to building uranium power reactors.

The experimental information for plutonium-fueled reactors is sparse relative to that available for uranium-fueled systems. However, studies have been conducted for a range of moderator-to-fuel volume ratios, plutonium concentrations, and ^{240}Pu concentrations so that an important set of data is available for evaluation of calculational methods. Since approximations utilized in the calculational methods depend on the moderator-to-fuel ratio, a range of ratios broader than necessary for pressurized or boiling water designs has been studied. The results allow conclusions to be drawn about the adequacies of the approximations.

Some of the major characteristics of the plutonium fuel which have been varied in the reactor studies in H_2O are summarized in Table I. Fuel rods containing plutonium only have been used at two plutonium concentrations, 1.8 and 5.0 wt%, with the same atom percent ^{240}Pu ; and at two different at% ^{240}Pu with approximately the same plutonium concentration. New measurements with a high buckling lattice for the Al-5.0 wt% Pu fuel have recently been completed at five concentrations of boron in the moderator. Studies involving the mixed oxides of plutonium and uranium have been accomplished at different concentrations of plutonium and essentially three ^{240}Pu concentrations, 8, 17, and 24 at%. Since physics characteristics of plutonium- H_2O reactors such as temperature, void and boron coefficients of reactivity and power distributions are

TABLE I
STUDIES WITH PLUTONIUM-ENRICHED FUEL

Light Water Moderator

<u>Fuel Type</u>	<u>wt% Pu</u>	<u>at% ²⁴⁰Pu</u>	<u>Configurations</u>	<u>References</u>
Al-Pu	1.8	6.0	Uniform	(52)(113)
	2.0	16.5	Uniform	(53)(114)
	5.0	6.0	Uniform, Boron	(115)
(Depl.) UO ₂ -PuO ₂	0.9	8.0	Uniform	(89)
(Depl.) UO ₂ -PuO ₂	1.5	8.0	Uniform	(49)(55)
(Nat.) UO ₂ -PuO ₂	1.5	16.0	Uniform, Cold and Hot	(4)
(Nat.) UO ₂ -PuO ₂	2.0	8, 17, 24	Uniform and Zoned	(48)(57)
(Nat.) UO ₂ -PuO ₂ and UO ₂ -2.7 at% ²³⁵ U	2.0	8, 24	Uniform, Zoned, Mixed, Nonuniform, Boron	(62)
(Nat.) UO ₂ -PuO ₂ and UO ₂ -2.3 at% ²³⁵ U	2.0	8, 24	Uniform, Zoned, Nonuniform	(59)
(Nat.) UO ₂ -PuO ₂	4.0	18.0	Uniform	(93)
(Nat.) UO ₂ -PuO ₂ and UO ₂ -5.74 wt% ²³⁵ U	6.6	8.0	Uniform, Zoned, Nonuniform, Boron	(63)

Heavy Water Moderator

Al-Pu Clusters	1.8	6.0	Uniform, Zoned	(69)(70)(67)
(Nat.) and UO ₂ Clusters --		---		
(Nat.) UO ₂ -PuO ₂ Clusters	2.0	8.0	Uniform, Boron	(77)(84)
(Depl.) UO ₂ -PuO ₂ Clusters	0.31 0.27 0.38	20.0 6.0 6.0	Zoned Zoned Zoned	(79)

Graphite Moderator

Al Pu Clusters	1.8	6.0	Clusters, Uniform	(82)
	2.2	20.0	Clusters, Uniform	(84)
(Depl.) UO ₂ -PuO ₂ Clusters	0.9	8.0	Uniform	(86)
(Depl.) UO ₂ -PuO ₂ Clusters	2.6	29.0	Uniform	(87)
(Nat.) UO ₂ -PuO ₂	2.0	8.0	Uniform	(90)

strong functions of the plutonium concentration, experiments⁽⁹³⁾ are being conducted using 4 wt% PuO₂ in UO₂ fuel. This concentration is of interest for the use of plutonium in power reactors moderated with H₂O and the plutonium isotopic composition is typical of that obtained from current uranium boiling water reactors.

1. Criticality in H₂O

The experiments conducted with the plutonium fuels in H₂O are summarized in Table II. The data generated in these experiments are mainly critical mass and/or bucklings. There is a limited amount of information available concerning the moderator and void coefficients of reactivity, absorber worths, kinetics parameters and power distributions. Meager information is currently available on the lattice cell parameters such as disadvantage factors and resonance reaction rates. Thus, the analytical correlations have mainly been made on the basis of multiplications, k_{eff} or bucklings, B^2 . Additional cell parameter data would help determine the cause of discrepancies which appear between experiment and calculation. The possibility exists that the discrepancies may be associated with errors in the experiment, errors in the theoretical methods, errors in the cross-section data, or possibly errors in all three. Experiments performed at different laboratories with the same fuels help to locate systematic experimental errors. This was done, with the UO₂-2 wt% PuO₂ fuels containing 8 and 24 at% ²⁴⁰Pu, at the Plutonium Recycle Critical Facility⁽⁵⁷⁻⁵⁹⁾ and the Westinghouse Reactor Evaluation Center.⁽⁶²⁾

TABLE II. SUMMARY OF EXPERIMENTS USING PLUTONIUM FUEL IN H₂O MODERATED LATTICES

<u>Zircaloy-Clad Al-Pu Fuel</u>								
<u>Laboratory</u> *	<u>Assembly</u> *	<u>Weight % of Pu in Fuel</u>	<u>Content of Pu²⁴⁰ in Pu (wt%)</u>	<u>Fuel Diameters (in.)</u>	<u>Range of Water/Fuel Volume Ratios</u>	<u>Range of Values of Bm² (M⁻²)</u>	<u>Whether Poison Used in Water</u>	<u>No. of Lattices Investi- gated</u>
PNL	E, AC and C	1.8	5.58	0.500	1.20 to 2.70	47.3 to 61.3	No	5
PNL	E, AC and C	2.0	16.46	0.500	1.20 to 2.70	36.3 to 48.5	No	6
PNL	E and AC	5.0	5.33	0.500	1.20 to 6.17	-1.97 to 112.61	Yes	1 Borated 9 Unborated
<u>Zircaloy Clad UO₂-PuO₂ Fuel</u>								
PNL	E and AC	1.5	7.85	0.372	1.10 to 5.58	48.0 to 78.5	No	6
WAPD	C	6.6	8.6	0.3374	1.68 to 10.8	108.8 to 159.6	No	5
PNL	E, AC and C	2.0	7.65	0.508	1.49 to 9.95	32.7 to 103.4	No	9
PNL	E, AC and C	2.0	16.54	"	2.45 to 6.97	55.5 to 89.1	No	6
PNL	E, AC and C	2.0	23.50	"	1.49 to 6.97	44.1 to 78.9	No	8
WAPD	C	2.0	7.65	"	1.21 to 7.84	69.1 to 105.9	Yes	2 Borated 5 Unborated
WAPD	C	2.0	23.50	"	3.42 to 4.23	73.3 to 79.5	No	2
PNL	E and AC	4.0	18.10	0.4975	1.56 to 10.9	61.6 to 87.6	No	3

* PNL = Pacific Northwest Laboratory, WAPD = Westinghouse Atomic Power Division, E = Exponential,
AC = Approach to Critical, C = Critical.

Studies at Pacific Northwest Laboratory. Critical mass and/or bucklings were determined by subcritical experiments in the Critical Approach Facility over a broad range of volume ratios of moderator-to-fuel. The basic quantities determined in the subcritical experiments are the number of rods for a critical loading and the relaxation length for thermal neutrons in the vertical direction. The critical number of rods were deduced by extrapolation⁽⁴⁸⁾ of multiplication data obtained when fuel rods were loaded up to 96% of the critical mass. The relaxation length was obtained from vertical traverses of the neutron density in an exponential loading of approximately 50% of the critical number of rods. The number of rods for critical, the relaxation length, the physical length of the rods and the number of rods in the exponential loading are combined to yield values for the buckling at critical and an average reflector savings. The reflector savings is an average which is assumed to have no geometrical dependence and is equal in both the axial and radial direction.

The critical number of rods was obtained by fitting a straight line by the method of least squares to plots of the inverse of the multiplication versus the number of rods which were loaded. Various techniques were used to obtain the multiplications. The most satisfactory technique involved placing three counters symmetrically around the core at a radius beyond where the critical core radius was expected to be. The count

rates obtained at these positions were corrected for the attenuation of neutrons through the thickness of water between the edge of the core and the counters. Data corrected in this manner resulted in straight lines after approximately 65% of the critical mass had been loaded.

The number of rods for a critical loading was obtained by fitting a straight line to the linear part of the data. The number of rods, N_c , for a critical loading was used in the equation for the critical buckling (i.e., geometrical buckling, B_g^2).

$$B_g^2 = \left(\frac{2.4048}{R_c + \lambda} \right)^2 + \left(\frac{3.1416}{H + 2\lambda} \right)^2 \quad (1)$$

where the critical radius, R_c is

$$R_c = \ell \sqrt{N_c \frac{\sqrt{3}}{2\pi}} .$$

The H , λ , and ℓ are the length of the fuel, the reflector savings and the triangular lattice spacing respectively.

Vertical traverses were obtained along the central axis of the loading with either a BF_3 counter or indium foils when approximately 50% of the critical mass was loaded. These data were fitted by the method of least squares to a hyperbolic sine function which corrected for the end effects and determined the inverse relaxation length, γ . The value of γ was used in the equation for the material buckling.

$$B_m^2 = \left(\frac{2.4048}{R_e + \lambda} \right)^2 - \gamma^2 . \quad (2)$$

The R_e is the radius of the equivalent bare cylindrical loading (i.e.,

$$R_e = \ell \sqrt{\frac{N_e \sqrt{3}}{2\pi}}$$

where N_e is the number of rods in the exponential loading). The reflector savings, λ , is assumed to be independent of the number of rods loaded and is the same quantity as that in Eq. 1.

Values of the reflector savings λ and the buckling $B^2 = B_m^2 = B_c^2$ were calculated for each lattice by equating the material and critical bucklings and solving for λ . The error in the value for the critical number of rods is the standard deviation of measurements from three detectors. The error on the value of λ is caused by the error on the critical number of rods and by the error on the relaxation length. The latter error is calculated by least squares analysis from the goodness of fit to the data from the vertical traverse. Special techniques were required to obtain data for the Al-Pu studies with an equivalent precision to that obtained for the UO_2 -Pu O_2 studies because of a neutron source contribution from (α, n) reactions in the aluminum.

Results for the measurements in the Critical Approach Facility are listed in Tables III-V. These critical mass and buckling measurements span a broad range of moderator-to-fuel volume ratios and have become a standard set of data for use in comparison of calculational methods for plutonium systems at many laboratories. For this reason, several new measurements (Table III)

TABLE III

RESULTS OF SUBCRITICAL MEASUREMENTS WITH Al-Pu RODS IN H₂O

Lattice Spacing (in.)	H/Pu (Atom Ratio)	H ₂ O/Rod (Volume Ratio)	Critical		Exponential			
			No. Rods	Buckling (10 ⁻⁶ m ⁻²)	No. Rods	Vertical Relaxation Length(cm)	Reflector Savings and Extrapolation Length, λ(cm)	
Al-1.82 wt.% Pu Rods (average temperature, 17°C)								
New→	0.75	630	0.9361	563.4 + 0.9	60.2 + 0.2	348	-25.03 + 0.09	8.89 + 0.05
	0.80*	810	1.203	510.3 + 0.9	61.3 + 0.3	312	-23.93 + 0.13	8.26 + 0.07
				484.5 + 0.3	60.0 + 0.1	241	-19.19 + 0.06	9.19 + 0.03
	0.85	1001	1.487	493.9 + 1.0	60.1 + 0.3	312	-25.05 + 0.12	7.56 + 0.06
	0.90	1204	1.788	515.8 + 1.0	52.7 + 0.3	300	-24.07 + 0.18	7.96 + 0.12
	0.95	1418	2.106	577.7 + 1.2	47.3 + 0.3	366	-29.30 + 0.16	7.07 + 0.06
Al-2.0 wt.% Pu Rods (average Temperature 23°C)								
New→	0.66	308	0.4993	1383.0 + 12	42.0 + 2	702	-26.21 + 0.59	8.59 + 0.38
	0.75	578	0.9361	848.4 + 0.8	48.5 + 0.3	432	-23.84 + 0.34	8.78 + 0.13
	0.80	743	1.203	795.6 + 0.7	47.4 + 0.2	408	-24.28 + 0.25	8.44 + 0.09
	0.85*	918	1.487	785.6 + 0.4	45.4 + 0.2	408	-25.22 + 0.10	7.86 + 0.08
				802.0 + 8				
New (20°C)	0.90	1104	1.788	869.8 + 0.3	41.2 + 0.1	474	-29.21 + 0.22	6.90 + 0.07
	0.93	1221	1.976	1041.0 + 2	33.2 + 0.1	517	-31.32 + 0.08	8.28 + 0.05
	0.95	1301	2.106	1099.2 + 0.4	36.3 + 0.1	558	-26.70 + 0.17	4.65 + 0.04
Al-5 wt.% Pu Rods								
New→	0.75	218	0.9361	356.1	96.1			8.10
	0.85	355	1.487	230.2	108.4			7.92
				220.7 + 0.2				
	0.90	427	1.788	192.0	112.6			7.93
	1.00	583	2.442	170.1	108.5			7.73
New→	1.05	665	167.3 + 0.1	(Data has recently been obtained also for 47,100,164,229,& 285 wppm boron)				
	1.10	755	3.165	166.5	101.1			7.34
	1.20	944	3.956	181.1	88.4			7.03
	1.30	1149	4.812	215.5	76.8			6.04
New→				214.9 + 0.4				
	1.50	1609	6.748	307.8	53.3			5.80

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BNWL-80

* Additional measurements made in a critical loading in the PRCF.

TABLE IV

RESULTS OF SUBCRITICAL MEASUREMENTS WITH UO_2 -1.5 wt.% PuO_2 RODS IN H_2O
Average Temperature, 25°C

Lattice Spacing(in.)	H/Pu (Atom Ratio)	$\text{H}_2\text{O}/\text{Rod}$ (Volume Ratio)	Critical		Exponential		
			No. Rods	Buckling (M^{-2})	No. Rods	Vertical Relaxation Length(cm)	Reflector Savings and Extrapolation Length, λ (cm)
0.55	230	0.8382	1487 $\pm$ 3	48.0 $\pm$ 1.2	714	-19.7 $\pm$ 0.1	8.40 $\pm$ 0.06
0.60	326	1.187	829 $\pm$ 7	65.1 $\pm$ 1.8	420	-17.66 $\pm$ 0.04	7.99 $\pm$ 0.10
0.71	567	2.063	484 $\pm$ 1	78.5 $\pm$ 0.3	252	-16.28 $\pm$ 0.02	7.27 $\pm$ 0.03
0.80	794	2.889	420 $\pm$ 1	74.9 $\pm$ 0.3	210	-15.72 $\pm$ 0.02	6.93 $\pm$ 0.03
0.90	1077	3.922	452 $\pm$ 2	60.9 $\pm$ 0.2	240	-18.54 $\pm$ 0.03	6.75 $\pm$ 0.03
0.93	1169	4.255	488 $\pm$ 2	55.2 $\pm$ 0.2	270	-20.40 $\pm$ 0.04	6.64 $\pm$ 0.05

TABLE V
RESULTS OF SUBCRITICAL MEASUREMENTS
WITH 2 wt% PuO₂-UO₂ RODS IN H₂O *

Temperature (°C)	Lattice Spacing (in.)	H/Pu (Atom Ratio)	H ₂ O/Rod (Volume Ratio)	Critical		Exponential		
				No. Rods	Buckling (m ⁻²)	No. Rods	Vertical Relaxation Length (cm)	Reflector Savings and Extrapolation Length λ (cm)
8% Pu ²⁴⁰ RODS								
23.9	0.80	238	1.187	319.0±0.3	88.0±0.3	163	-15.46±0.09	7.93±0.04
20.2	0.93	391	1.956	193.0±0.1	102.9±0.3	97	-14.88±0.07	7.55±0.03
19.8	1.05	554	2.768	149.8±0.4	103.4±0.6	73	-14.41±0.11	7.57±0.06
23.4	1.143	693	3.464	146.6±0.3	97.8±0.4	73	-14.78±0.05	7.05±0.03
22.6	1.32	991	4.956	163.0±0.2	76.8±0.3	85	-17.70±0.14	6.73±0.05
26.2	1.386	1113	5.564	180.9±0.4	70.0±0.4	91	-17.75±0.11	6.00±0.06
19.4	1.6107	1550	7.750	472 ±3	32.7±0.4	241	-33.15±0.50	3.85±0.21
16% Pu ²⁴⁰ RODS								
20.0	0.93	391	1.956	241.2±0.3	88.4±0.3	121	-16.03±0.07	7.67±0.04
22.6	1.05	554	2.768	188.1±0.3	89.0±0.5	91	-15.36±0.12	7.62±0.06
20.5	1.143	693	3.464	186.8±0.4	81.9±0.3	91	-15.92±0.07	7.29±0.04
25.5	1.32	991	4.956	221.3±0.3	61.8±0.2	109	-18.72±0.12	6.88±0.06
24.4	1.386	1113	5.564	254.9±0.6	55.5±0.3	127	-19.74±0.09	5.87±0.06
24% Pu ²⁴⁰ RODS								
24.4	0.80	238	1.187	524.2±1.3	62.4±0.3	253	-18.88±0.08	8.32±0.05
24.7	0.93	391	1.956	287.5±0.4	78.9±0.3	139	-16.33±0.07	7.67±0.04
23.5	1.05	554	2.768	234.2±0.3	77.1±0.3	121	-17.72±0.11	7.63±0.05
22.0	1.143	693	3.464	232.1±0.2	72.2±0.2	109	-16.23±0.06	7.00±0.04
25.3	1.32	991	4.956	296.7±0.4	53.6±0.2	151	-20.76±0.07	5.81±0.04
23.6	1.386	1113	5.564	366.7±0.4	44.0±0.1	181	-23.07±0.08	5.47±0.4

* Critical measurement made in the PRCF at an 0.85 lattice spacing.

have been made in order to assure that the data is accurate. The new measurements have centered mainly around the Al-Pu fuel. Experiments with (113-115) this fuel were some of the first conducted with plutonium and the measurements were made with Lucite tubes around the fuel rods because of the concern for plutonium leakage from the rods. In conducting the experiments precautions were taken to assure that the Lucite did not change the critical mass appreciably and that all voids between the Lucite and the fuel were removed. Now, with the greater handling experience there is no such concern and the new measurements are conducted without any Lucite being present. The new measurements have not resulted in differences which improve the correlation between the experiment and the calculated value appreciably. However, the differences do tend to remove some of the trends which appear in previous correlations. Because of this, it is expected that further remeasurements will help understand better where some of the problem areas are in the calculational methods.

For interpolation purposes the function⁽⁴⁹⁾

$$y = \frac{A}{x - B} + \frac{C}{D - x} \quad (3)$$

is fit to the data by the proper choice of the four parameters, A, B, C, and D. The number of rods for critical or the inverse buckling is represented by y and the lattice spacing or the atom ratio H/Pu is represented by x. This equation has been shown to fit all of the data relatively well.

The worth of centrally located, cylindrical absorber rods containing hafnium oxide and boron were measured⁽⁵⁰⁾ bare and cadmium covered relative to the worth of peripheral fuel rods in three different lattices. Data were obtained for rods containing five different hafnium oxide densities and four different boron densities. Reactivity values were obtained using computed worths of peripheral rods. Subsequent measurements⁽⁵¹⁾ in a critical facility has corroborated the validity of this measurement technique. All three types of fuel which contained 2.0 wt% $\text{PuO}_2\text{-UO}_2$ were used for these measurements.

2. Reactivity Coefficients and Kinetics for H_2O

Studies at Pacific Northwest Laboratory. A summary of critical experiments conducted in the Plutonium Recycle Critical Facility with Al-1.8 wt% Pu⁽⁵²⁾ and Al-2.0 wt% Pu⁽⁵³⁾ fuel at lattice pitches of 0.80 and 0.85 in. respectively is presented in Table VI. Critical masses, flux shapes, kinetics parameters, moderator level, temperature void, and material coefficients of reactivity have been determined. The ratio of effective delayed neutron fraction to neutron lifetime has been measured for the Al-1.8 wt% Pu core using reactor noise techniques. The response of the loading to step changes of negative reactivity was also studied.

The moderator level coefficient of reactivity was measured⁽⁵⁴⁾ for levels from ~8 in. below to ~9 in. above the top of the Al-1.8 wt% Pu cores and a value for the migration area was

TABLE VI

SUMMARY OF H₂O-MODERATED CRITICAL EXPERIMENTS IN THE PRCF

²⁴⁰ Pu Fuel Type			Measured Quantities																							
8%	16%	24%	Critical Mass	Radial B ²	Axial B ²	Radial λ	Axial λ	Kinetics	Fuel Worth	Moderator Level Coefficients	Void Worth	Temperature Coefficients	Reactivity Worths	Fuel Rods	Copper Rods	HfO ₂ Rods	EuO ₂ Rods	Aluminum Rods	Boron-Al Rods	Cadmium Rods	Zircaloy, et al	Flux Shapes By:	Activation	Fission Product Scan		
Al-1.8 wt% Pu (0.80 in. triangular pitch)																										
A11			✓				✓	✓	✓	✓	✓	✓		✓	✓	✓		✓						✓		
Al-2.0 wt% Pu (0.85 in. triangular pitch)																										
A11			✓							✓																
UO ₂ -1.5 wt% PuO ₂ (0.71 in. triangular pitch)																										
A11			✓					✓		✓	✓	✓		✓	✓		✓		✓	✓				✓		
UO ₂ -2.0 wt% PuO ₂ (0.85 in. triangular pitch)																										
A11			✓	✓	✓	✓	✓	✓	✓	✓		✓		✓		✓		✓	✓					✓	✓	
Inner Cylinder		Outer Cylinder	✓																							
		All Inner Cylinder	✓	✓	✓	✓	✓	✓	✓	✓		✓		✓	✓									✓	✓	
	Outer Cylinder	Cylinder	✓				✓			✓														✓	✓	
	Inner Cylinder	Outer Cylinder	✓				✓			✓														✓	✓	
Outer Cylinder		Inner Cylinder	✓				✓			✓														✓	✓	

deduced from the data. The level coefficient was determined for the Al-2.0 wt% Pu core for levels above the top of the fuel. Temperature coefficients of reactivity were measured using the Al-1.8 wt% Pu fuel for temperatures between 19 and 35°C. Coefficients of reactivity for film voids around fuel rods and tubular voids in fuel rod positions were measured as a function of radius from the center of the Al-1.8 wt% Pu core to beyond the core edge. The reactivity effect of a 0.585 in. O.D. by 8 in. long void was also measured as a function of position. Pieces of Teflon and various densities of styrofoam 3 in. long and surrounding seven fuel rods were used to measure the effect of interstitial voids at the center of the core. In addition, the teflon worth was measured as a function of radius.

Coefficients of reactivity for polyethylene, lucite, teflon, polycarbonate, copper, aluminum, aluminum oxide, europium oxide, and hafnium oxide were determined at the center of the Al-1.8 wt% Pu core. The copper, aluminum, europium and hafnium were in the form of rods. A rod containing 0.0369 gm/cm³ of europium oxide was used. Concentrations of 0.125, 0.54, and 1.08 gm/cm³ were used for the hafnium oxide rods. In addition to the measurements at the center of the core, the copper rod was measured as a function of radius. The worth of fuel placed in a tubular void and in moderator was measured. The reactivity uniformity of both types of fuels were also measured.

Radial, axial, and cell traverses were obtained using bare and cadmium covered lutetium. Radial traverses were obtained using copper. Radial and axial traverses were made using bare and cadmium covered ^{233}U , ^{235}U , and ^{239}Pu fission chambers. The deviation of the slowing down neutron spectrum from a $1/E$ shape was deduced at a few points using palladium and gold foils.

A summary of critical experiments⁽⁵⁵⁾ conducted with UO_2 -1.5 wt% PuO_2 fuel and H_2O moderator in the Plutonium Recycle Critical Facility is presented in Table VI. The experiments were conducted in preparation for an irradiation experiment in the Experimental Boiling Water Reactor (EBWR) using the same fuel. A lattice pitch of 0.71 in. which corresponds to an average moderator-to-oxide volume ratio for the EBWR was chosen for the measurements.

Critical mass, bucklings, reflector savings, kinetics parameters, coefficients of reactivity for moderator level, temperature, void, and various material changes and foil activation data were obtained using the 0.71 in. lattice. The ratio of the effective delayed neutron fraction to the neutron lifetime was measured using reactor noise techniques. In addition, the flux transient caused by negative insertion of reactivity was studied. Reactivity coefficients of moderator level changes were measured levels from approximately 6 in. above the top of the fuel to 15 in. below the top of the fuel.⁽⁵⁶⁾

The worth of H_2O and voids in positions of fuel rods were measured as a function of radius. The worth of a film void around a fuel rod was also measured as a function of core radius. A film and rod void was also measured as a function of height in the core. Temperature coefficients of reactivity were measured between 20 and 72°C. Aluminum-plutonium, UO_2 - PuO_2 , copper, europium oxide, hafnium oxide, Zircaloy-2, and cadmium reactivity measurements were made relative to H_2O . Foil activation data and axial, radial, and cell traverses were made with bare and cadmium covered, gold, palladium, copper and lutetium foils.

Experiments using UO_2 -2 wt% PuO_2 fuel⁽⁵⁷⁾ were conducted at a lattice spacing of 0.85 in. in the Plutonium Recycle Critical Facility and were the beginning of a series of reactor applications experiments. Single-fuel region experiments were conducted with fuel which contained plutonium containing 8, 16 and 24% ^{240}Pu . Two-fuel region experiments were conducted using various combinations of the three different ^{240}Pu concentrations. Measured parameters included critical mass, kinetics data, spatial flux distributions, power peaking and power sharing distributions and reactivity coefficients. Radial and axial bucklings and reflector savings were deduced from spatial distributions of thermal activation of copper pins and the spatial distribution of gamma-ray activities in the fuel rods. The ratio of the effective delayed neutron fraction

to the mean neutron lifetime, was obtained in the single-fuel region cores. Moderator level reactivity coefficients at the top of the core and above, and temperature coefficients of reactivity over the temperature range between 20 and 50°C were determined for the two single-fuel region cores. Worths of fuel rods, cylindrical voids, and annular voids and other materials were measured as a function of radius in the single-fuel region cores also.

Current emphasis is being placed on reactor applications experiments involving non-uniform loadings of fuel and arrangements which simulate the heterogeneous geometry of present-day H₂O power reactors. Power distribution and power sharing measurements between uranium and plutonium fuels are being made in the Plutonium Recycle Critical Facility using techniques which measure the fission product gamma-rays emitted from the fuel rods. A series of these measurements⁽⁵⁹⁾ which study the reactor physics problems connected with plutonium recycle in a boiling water reactor has just been completed under a joint USAEC and National Committee for Nuclear Energy (CNEN) of Italy.

The experiments have centered around measurements of power distributions. Rod-to-rod distributions have been measured in cores that contained water gaps, partial moderator voids, plutonium-enriched bundles surrounded by enriched uranium bundles and vice versa. Similar studies were completed near control blades and in composite bundles containing rods of several levels of plutonium enrichment. The rods were arranged in the bundles

to minimize power peaking at the bundle edges and corners. Another part of the program has been in the measurement of the relative worths of control blades which contain different absorbing materials, and the changes in these relative worths between an enriched uranium core and a plutonium-enriched core.

Five different fuel compositions were used in the study: $2.3\% \text{ }^{235}\text{UO}_2$, $\text{UO}_2\text{-}0.9 \text{ wt\% PuO}_2$, $\text{UO}_2\text{-}2 \text{ wt\% PuO}_2$ ($8\% \text{ }^{240}\text{Pu}$), $\text{UO}_2\text{-}2 \text{ wt\% PuO}_2$ ($24\% \text{ }^{240}\text{Pu}$) and $\text{UO}_2\text{-}4 \text{ wt\% PuO}_2$. The diameters of the enriched UO_2 rods were smaller (0.44 in.) than the diameters of the mixed oxide rods (0.508 in.). This resulted in a moderator/oxide volume ratio of 2.41 and 1.55 respectively for the different rods, when arranged in the 0.75 in. square pitch lattice.

A large assortment of lattice configurations was measured during the power distribution measurements. Uniform lattices, both rectangular and nearly cylindrical, were assembled to compare with the critical size measurements made with the $\text{UO}_2\text{-PuO}_2$ fuels in the same lattice pitch at WREC⁽⁶²⁾ under the ESADA program. Critical size determinations were made for each of the core configurations studied. The radial power distributions were measured in the many lattice configurations shown in Table VII. In the cases of the water hole, water slot and water cross, fuel rods were removed from the central lattice position, from a row extending across the core, and from two rows which extended across the core and intersected at the central position. The 7 x 7 and 9 x 9 bundle configurations

TABLE VIIPOWER DISTRIBUTION MEASUREMENTS IN THE REACTOR APPLICATIONS EXPERIMENT

<u>Lattice Configuration</u>	<u>Shape</u>	<u>2.35% $^{235}\text{UO}_2$</u>		<u>UO_2-2 wt% PuO_2</u>	
		<u>Uniform</u>	<u>Zoned (with)</u>	<u>Uniform</u>	<u>Zoned (with)</u>
Regular		✓		✓	
Water Hole		✓		✓	
Water Slot		✓		✓	
Water Cross		✓		✓	
7 x 7 Bundle		✓	✓ (8%)	✓	✓ (UO_2) (24%)
9 x 9 Bundle		✓	✓ (8%)	✓	✓ (UO_2)
9 x 9 Bundle (voids)		✓		✓	
9 x 9 Control Blades		✓		✓	
9 x 9 Composite			✓ (comp)		✓ (comp)
11 x 11 No Gap			✓ (8%)		✓ (UO_2)

were formed by removing rods from four rows which extended across the core, to form bundles of rods, one of which was centered in the core. Moderator voids were simulated in the central bundle by inserting air-filled aluminum tubes into the spaces between rods. Zoned loadings were made in which the central bundle contained a fuel type different from the surrounding rods. Composite bundles were assembled in which the corner rods contained UO_2 -0.9 wt% PuO_2 , the edge rods contained UO_2 -2 wt% PuO_2 and the remaining inside rods contained UO_2 -4 wt% PuO_2 .

The effect of

- the excess moderator in the water gaps and at the corners and edges of bundles,
- the lack of moderator in the partially voided regions,
- the interactions between fuels of different compositions, and
- the local depression adjacent to control blades

upon the power distributions in the cores were measured by scanning the fission product activities in selected rods following low power irradiations. Normalization experiments were necessary to relate the observed fission product activities to the power production in each type of rod, since the fission yields are different for plutonium fission and uranium fission, the mixed oxide fuels had differing

enrichments and isotopic compositions and since the rod volumes, UO_2 to mixed oxide, were different. The inter-normalizations were determined calorimetrically by measuring the rate of temperature rise at the surface of the cladding of the rods, while being irradiated at a steady power level in the same neutron flux, and by counting the activity of ^{140}La present in each of the rod types about 15 days after the calorimetric measurements.

The structure of the control blade permitted the absorbing materials to be changed, thus allowing measurements to be made in which only the contained absorbers were different. The absorbers studied included: Boral, B_4C , HfO_2 , Hf metal, and a mixture of rare earth oxides (Hf, Dy, Ho, Er). The control-blade worth measurements were performed with the reactor subcritical, using the endogenous pulsed source method.⁽⁶⁰⁾ This method determines the prompt-neutron-density decay eigenvalue, α , from a time analysis of the die-away of spontaneous bursts of neutrons in the assembly. The blade worths, which were derivable from the measured values of α , ranged in the vicinity of 10\$ and were determined with a precision of a few percent.

A determination of the value of α for the just critical system where α is the ratio of the effective neutron fraction to the neutron lifetime was included with the control blade measurements in the uniformly enriched UO_2 cores. The large spontaneous source inherent in the plutonium-enriched fuels

prevented a direct measurement of the critical α for those cores. However, an extrapolation was made from measurements performed in cores which were subcritical by known small amounts to obtain a value of the ratio of the effective delayed neutron fraction to the neutron lifetime for the plutonium-enriched system.

An extensive theory-experiment correlation program is being conducted for these measurements, aimed initially at testing the ability of the RIBOT⁽⁶¹⁾ code to adequately predict the behavior of plutonium-enriched fuels inserted into present generation boiling water reactors. Subsequent correlations will be made using other calculational methods such as THERMOS, HRG, HAMMER, HFN, EXTERMINATOR, etc., to test the capabilities of these codes.

The values of critical mass, buckling, and reflector savings measured in the Plutonium Recycle Critical Facility are summarized in Table VIII. Comparisons are made with values obtained for the Critical Approach Facility. The latter values are obtained directly from measurements or indirectly from interpolations or corrections. When direct comparisons or comparisons involving corrections based on measurements are made, the results from both facilities are in reasonable agreement.

TABLE VIII

COMPARISON OF CRITICAL MASS AND BUCKLING MEASUREMENTS

FROM CAF AND PRCF MEASUREMENTS

Facility	Fuel	Lattice Pitch(in.)	Critical Mass (Rods)	Buckling (M^2)			Reflector Savings (cm)		
	Al-1.8 wt% Pu			B^2_{radial}	B^2_{Total}	B^2_{axial}	λ_{radial}	$\bar{\lambda}_{\text{ave.}}$	λ_{axial}
PRCF	8% ^{240}Pu	0.80	483 $\pm$ 1						
CAF			484 $\pm$ 1						
	Al-2.0 wt% Pu								
PRCF	16% ^{240}Pu	0.85	802 $\pm$ 8						
CAF			797 $\pm$ 6 ^a						
	UO_2 -1.5 wt% PuO_2								
PRCF		0.71	482 $\pm$ 1	74.5 $\pm$ 0.3	5.3 $\pm$ 0.1		7.0 $\pm$ 0.4	7.1 $\pm$ 0.2	
CAF			484 $\pm$ 1	78.5 $\pm$ 0.3			7.27 $\pm$ 0.03		
	UO_2 -2.0 wt% PuO_2								
PRCF	8% ^{240}Pu	0.85	252 $\pm$ 1	82.2 $\pm$ 3.3	8.89 $\pm$ 0.09		8.6 $\pm$ 0.5	7.0 $\pm$ 0.3	
CAF			246 ^b	96.6 ^b					
PRCF	24% ^{240}Pu	0.85	394 $\pm$ 1	63.3 $\pm$ 1.4	9.11 $\pm$ 0.07		7.8 $\pm$ 0.3	6.3 $\pm$ 0.2	
CAF			382 ^b	71.6 ^b					
PRCF	8% ^{240}Pu	0.75 ^c	321 $\pm$ 2						

a Corrected for Lucite tubes using estimate based on measurement of worth in the PRCF.

b Interpolated result.

c Square pitch and identical to result obtained at WREC; all other pitches are triangular.

Studies at The Westinghouse Reactor Evaluation Center.

Critical experiments were conducted at the Westinghouse Reactor Evaluation Center⁽⁶²⁾ using the fuel rods which had been used at Pacific Northwest Laboratory^(48,57,59) and which contained UO_2 -2 wt% PuO_2 . Templates with two different pitches were used for these experiments. The holes in the templates were arranged in the form of a square lattice and, by loading only particular holes, experiments were conducted at five different square pitches. The periphery of most cores was approximately cylindrical except for two lattice spacings for which results using both a square and a cylindrical periphery were compared. The poisoning effect of boron in the moderator was measured for two lattice spacings and for two different boron concentrations.

The different configurations which were studied are listed in Tables IX-XI. Critical masses, bucklings, power distributions, and reactivity coefficients were studied in single region experiments using each type of plutonium fuel and fuel enriched to 2.72 wt% $^{235}\text{UO}_2$. Radial and axial buckling values were determined by counting the fission product activity emitted by the fuel rods. In addition, power distribution measurements were made in square, cylindrical, and interspersed mixtures of salt and pepper type of rod arrays using two types of fuel in various combinations of the three different arrays. Finally, power distributions were measured in three region cores with the fuels arranged in slab geometry.

TABLE IX

COMPARISON OF SINGLE REGION RESULTS FROM UO_2 -2.0 wt% PuO_2 EXPERIMENTS AT BATTELLE NORTHWEST AND WESTINGHOUSE

The moderator temperature was $\sim 22^\circ\text{C}$ and $\sim 23^\circ\text{C}$, respectively, for the WREC and CAF experiments.

Fuel Type (UO_2 -2.0 wt% PuO_2)	Square Lattice Pitch (in.)	H/Pu Atom Ratio	Boron wppm	Critical No. of Rods		Buckling (M^{-2})	
				WREC ^(a)	CAF	WREC	CAF
8% ^{240}Pu	0.69	176	0	514	536 ^(b)	69.6 ± 1.0	$70.9^{(b)}$
8% ^{240}Pu	0.69	176	261 ± 1	631		62.6 ± 0.6	
8% ^{240}Pu	0.69	176	526 ± 1	749		58.3 ± 0.6	
8% ^{240}Pu	0.69	176	0	552 (sq)		68.7 ± 0.8	
8% ^{240}Pu	0.75	244	0	321	308 ^(b)	90.0 ± 0.9	$89.2^{(b)}$
8% ^{240}Pu	0.9758	552	0	160	321 ^(c) 155	104.7 ± 0.9	103.0
8% ^{240}Pu	0.9758	552	261 ± 1	243		83.7 ± 0.8	
8% ^{240}Pu	0.9758	552	526 ± 1	379		63.1 ± 0.6	
8% ^{240}Pu	0.9758	552	0	169 (sq)		107.1 ± 1.2	
8% ^{240}Pu	1.0607	688	0	152	147	98.4 ± 1.2	97.6
8% ^{240}Pu	1.380	1303	0	249	235 ^(b)	50.3 ± 0.3	$53.8^{(b)}$
24% ^{240}Pu	0.9758	552	0	247	237	79.5 ± 1.1	78.0
24% ^{240}Pu	1.0607	688	0	243	232	73.5 ± 0.7	72.5
<hr/> (UO_2) <hr/>							
2.72% ^{235}U	0.69		0	317			
2.72% ^{235}U	0.9758		0	293			

(a) The water level for the critical loading did not provide an infinite reflector for all cases.

(b) These numbers required extrapolation rather than interpolation of the curve which fits the CAF data.

(c) From a square-pitch measurement with a critical loading in the PRCF.

TABLE X

SUMMARY OF REACTIVITY EXPERIMENTS USING UO_2 -2.0 wt% PuO_2 FUEL AT
THE WESTINGHOUSE REACTOR EVALUATION CENTER

Fuel Type		Loading	Boron (wppm)	Moderator Worth	1 Water Hole*	5 Water Holes*	9 Water Holes	1 Ag-In-Cd rod	9 Ag-In-Cd rods	Co Rod	Zr Tube (Void)	Al Tube Voids (4 x 4 and 16 x 16)
8% ^{240}Pu	24% ^{240}Pu											
Square Lattice Pitch 0.69 in.												
A11		23 x 25 rods	0	✓	✓	✓		✓		✓	✓	
A11		27 x 27 rods	0	✓					✓			
			116	✓			✓		✓			
			261	✓								
			315	✓	✓	✓		✓		✓	✓	
Square Lattice Pitch 0.75 in												
A11		21 x 21 rods	0	✓	✓		✓	✓	✓			✓
Square Lattice Pitch 1.0607 in.												
	A11	Cylinder	0		✓			✓				

* Relative power measured in neighboring rods.

TABLE XI

POWER DISTRIBUTION MEASUREMENTS
 USING UO_2 -2.0 wt% PuO_2 AND 2.72 wt% UO_2 FUEL IN MULTI-REGION CORES
 AT THE WESTINGHOUSE REACTOR EVALUATION CENTER

<u>Fuel Type and Loading</u>		Axial B^2 , Inner	Axial B^2 , Outer	B^2	9 Water Holes*	9 Ag-In-Cd Rods*
<u>^{240}Pu</u>	<u>^{235}U</u>					
8%	24%					
<u>Square Lattice Pitch 0.69 in.</u>						
Outer Square	Inner Square	✓		✓		
Inner Square	Outer Square	✓	✓	✓		
	Inner Square	✓	✓	✓		
	Outer Square	✓	✓	✓		
	Inner Square	✓	✓			
Outer Cylinder	Inner Cylinder	✓	✓	✓		
Salt Square	Pepper Square	✓	✓	✓		
Salt Square	Pepper (2 mod. hts.) Square	✓	✓	✓	✓	✓
	Salt Square	✓	✓	✓	✓	✓
	Salt Cylinder	✓	✓			
<u>Square Lattice Pitch 0.75 in.</u>						
Outer Cylinder	Inner Cylinder	✓	✓	✓		
Outer Cylinder	Inner Cylinder Pitch 1.0607 in.	✓	✓	✓		

*Reactivity measurements, no power distributions

The subcritical measurements at Battelle Northwest, which were conducted in the Critical Approach Facility (CAF) using hexagonal lattices are compared in Table IX to the critical mass and buckling results obtained at the Westinghouse Reactor Evaluation Center (WREC) for square lattice pitches at equivalent hydrogen-to-plutonium ratios. The agreement between the measurements is good for bucklings but less good for the critical mass. Part of the disagreement occurs because the experiments in the WREC were not reflected with an effectively infinite top reflector. However, it is expected that this fact would account for only 1-2 rods. The difference could be caused by different absorbing materials in the cores. The latter, however, would make the buckling agreement worse. Comparisons of critical masses and bucklings for experiments in the CAF and the Plutonium Recycle Critical Facility (Table VIII) corroborate the subcritical technique used in the CAF. Assuming that the WREC results are also correct, the difference could be caused by the square versus the triangular pitch. If this is true, the square lattice cell would be expected to have a value for the infinite medium multiplication factor different from that for the triangular lattice cell. Experiments are being conducted in the Physical Constants Testing Reactor which will determine the infinite multiplication factor for a triangular and a square lattice pitch with equivalent H/Pu ratio.

The configurations and materials used in the reactivity experiments conducted with the UO_2 -2 wt% PuO_2 rods in H_2O at the Westinghouse Reactor Evaluation Center⁽⁶²⁾ are summarized in Table X. Reactivity effects of water height and various materials were measured using the UO_2 -2.0 wt% PuO_2 fuel which contained 8% ^{240}Pu and at two different lattice pitches. The core periphery was arranged either in the form of a square or rectangle and at one lattice spacing three boron concentrations were used. The reactivity effect of having a water gap in the center of the core, caused by removing either 1, 5, or 9 fuel rods was determined. In addition, the reactivity effect of either 1 or 9 silver-indium-cadmium rods in place of fuel rods was measured for various geometrical arrays of slots and clusters. Other materials that were measured were an empty zirconium tube and a cobalt rod. In one case, the effect of voids was measured by inserting aluminum tubes between the fuel rods in the central position. Relative powers in the fuel rods near the 1 and 5 rod water hole were determined by counting the fission products emitted from the rods. The reactivity worth for a central water hole or Ag-In-Cd rod was measured with fuel loaded at a third lattice pitch.

Configurations used for power distribution measurements across cores containing two types of fuel in square, cylindrical, and salt and pepper types of loadings are summarized in

Table XI. The $\text{UO}_2\text{-PuO}_2$ and UO_2 rods have different fuel diameters. Therefore, with the same lattice pitches different moderator-to-fuel volume ratios occur in the multi-region cores. One experiment was conducted with the same diameter fuel but in two regions with different lattice pitches to determine the effect of varying the moderator-to-fuel volume ratio in a multi-region core. Gamma-ray activities from fission products were related to specific power rates using calorimetric techniques for two different types of fuel. The relation was determined as a function of cooling time. In all cases listed in Table XI distributions were measured perpendicular to the periphery of the core. In many configurations power distributions were also measured across diagonals. A reanalysis⁽⁹²⁾ of the power normalization data has resulted in rod power values for the $^{235}\text{UO}_2$ fuel which are 6-7% higher relative to the $\text{UO}_2\text{-2.0 wt% PuO}_2$ fuel than reported previously,⁽⁶²⁾ (see the Saxton discussions which appear later).

Configurations used for measuring power distributions across three different slab regions of $\text{UO}_2\text{-2 wt% PuO}_2$ and 2.72 wt% $^{235}\text{UO}_2$ fuel are summarized in Table XII. Three regions were arranged with two external UO_2 regions arranged symmetrically on opposite sides of an internal $\text{UO}_2\text{-PuO}_2$ slab region. A 0.633 in. water gap occurred between the slab regions. In the UO_2 regions a smaller pitch was used than in the PuO_2 region in order to have the same volume ratio of hydrogen-to-fuel throughout the core. Power distributions were measured

TABLE XII

Reactivity and Power Distribution Experiments for Multi-Region Slab Loadings
at the Westinghouse Reactor Evaluation Center

(A water gap of 0.633 in. occurred between the slabs.)

Rod Array and Fuel ^a for Inner Slab (0.69" pitch)	Rod Array and Fuel for Outer Slabs (0.575" pitch)	Boron (wppm)	Voids:					Power Distributions	Critical Height
			4x4 = 12 rods	10x10 = 36	10x10 = 100 rods	16x22 = 352	12x12 = 144		
19 x 19 8% ²⁴⁰ Pu	23 x 10 U 2.72%	0	✓	✓	✓	✓		✓	✓
23 x 19 8% ²⁴⁰ Pu	27 x 15 U 2.72%	526	✓		✓		✓	✓	✓
(240 8% + 189 24%) ^b	25 x 10 U 2.72%	0						✓	✓
(228 8% + 171 24%) ^c	25 x 10 U 2.72%	0						✓	✓

(a) UO₂-2 wt% PuO₂

(b) Alternate strips of 8% and 24% fuel parallel to the slabs.

(c) Alternate strips of 8% and 24% fuel perpendicular to the slabs.

(d) Power distribution measurements were made for these arrays.

for two of the configurations with and without a central zone containing aluminum tubes to simulate voids between the fuel rods. Arrays of 4 x 4, 10 x 10, 12 x 12, and 16 x 22 tubes were used in the central zone.

As part of the Saxton Plutonium Program, a series of critical experiments⁽⁶³⁾ were conducted at the Westinghouse Reactor Evaluation Center using UO_2 -6.6 wt% PuO_2 and 5.74 wt% $^{235}\text{UO}_2$ fuel which was subsequently irradiated in the Saxton Reactor. The purpose of the experiments was to verify the nuclear design of the Saxton partial plutonium core. Because of the small size of the Saxton reactor, the plutonium concentration of the fuel is much larger than the 3-4 wt% expected for a large pressurized water reactor.

The critical experiments, conducted in single region and multi-region cores using fuel which now is loaded into the Saxton Reactor, are summarized in Table XIII. Experiments were conducted at five different square lattice pitches and critical masses, relative power distributions, reactivity coefficients, flux shapes, and kinetics parameters were determined. The spacing of 0.56 in. has the same H/Pu ratio as in the plutonium fueled region of the Saxton Reactor operating at power.

Power normalizations for the different fuels used in the multi-region experiments were obtained using two methods. In one method the normalization was obtained using calorimetric techniques with temperatures determined by thermocouples placed

TABLE XIII

Summary of H_2O Moderated Critical Experiments Using UO_2 -6.6 wt% PuO_2 at the
Westinghouse Reactor Evaluation Center

Square Lattice Pitch (in.)	Critical Mass	Power Distributions: Core	Water slot	Aluminum plate	5 Ag-In-Cd rods	Fuel	Reactivity Worths Water slot	Aluminum plate	5 Ag-In-Cd rods	Borated moderator	Fuel	Temperature Coeff.	Moderator level coefficients	Activation: Core	Rod circumference	Bucklings: Radial	Axial	Interstitial voids	Kinetics: Pulse neutron	Noise analysis
(UO ₂ -6.6 wt% PuO ₂)																				
Single Fuel Region																				
0.52	✓*												✓							
0.56	✓*		✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓		
0.735	✓*												✓			✓	✓			
0.792	✓*	✓											✓			✓	✓			
1.04	✓*												✓			✓	✓			
Single Fuel Region (6.74 wt% ²³⁵ UO ₂)																				
0.52	✓*												✓			✓*	✓*			
0.56	✓*	✓		✓	✓	✓	✓	✓	✓		✓	✓	✓	✓	✓	✓	✓		✓	
0.792	✓*	✓											✓			✓	✓			
Two-Fuel Region (UO ₂ and UO ₂ -PuO ₂) (square pitch, 0.56 in.)																				
UO ₂	UO ₂ -PuO ₂																			
Outer	Inner	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓				✓		✓
Inner	Outer	✓	✓	✓	✓					✓	✓									

* For cores with a cylindrical boundary. All other measurements made with a square or rectangular boundary.

on the cladding of an insulated rod. In another technique, the number of fissions were deduced from the ^{140}La activity produced in foils of the fuel material. The normalizations reported differ by 6-7%. In the analysis, the calorimetric measurements were corrected to each other allowing for the difference in fuel weight but not for the difference in specific heat of the two types of fuel rods. As a result the power reported for an UO_2 rod was underestimated⁽⁹¹⁾ by a factor⁽⁹²⁾ of 1.053 using the calorimetric method. After this correction is made the normalizations are in much better agreement. The power normalizations measured for these fuels and for the UO_2 -2.0 wt% PuO_2 and 2.72 wt% $^{235}\text{UO}_2$ fuel agree within 5%.

Relative power distributions were measured near a water slot, an aluminum plate, and a control rod. The water slot was obtained by removing five fuel rods from the center of the core. The aluminum plate was 0.125 in. thick by 26.0 in. square and placed between the templates in a slot formed by removal of fuel rods. The control rod was 0.4 in. in diameter and consisted of an unclad alloy of 80 percent cadmium, 15 percent indium, and 5 percent silver. The power generated in fuel rods which were located from between five to six positions from the slot, plate, or control rod were measured for uranium and plutonium cores and in the presence and absence of the perturbation.

Reactivity worths of the water slot, the aluminum plate, and the control rods were measured along with the reactivity

coefficient for boron. Fuel element worths were also obtained when an array of 19 x 19 fuel rods was changed to a 21 x 21 rod array. Worths of arrays of 3 x 3 and 7 x 7 rods of one type of fuel were measured in the 0.56 in. lattice of the other type of fuel. In addition, L shaped arrays of UO_2 rods were measured in a multi-region core to simulate the Saxton design. Reactivity changes caused by the peripheral substitution of rows of UO_2 , and by placing an aluminum plate or control rods at the interface of the two regions were measured. Temperature coefficients of reactivity were also determined at a few values between 17 and 73°C for the various loadings of the 0.56 in lattice.

Moderator level coefficients were measured for approximately six points, ranging down to five inches below the top of the fuel. In one case, data was obtained as far as 16 in. below the core top. This data was used in reactivity worth measurements and in determinations of the axial reflector savings.

Relative fluxes were measured as a function of position in the core using uranium, gold, and dysprosium foils. Fast flux was determined by measuring the fission products which occur because of fission in ^{238}U . The resonance flux was determined by measuring the ^{239}Pu x-ray which occurs in the decay of ^{237}Np which is present because of resonance capture in ^{238}U . The thermal flux is determined from the gold and dysprosium foils. Only bare foils were used to measure the

distributions and the traverses were made from one core edge through the core and on into the reflector on the opposite side of the core.

The circumferential thermal flux at a rod surface was obtained using a dysprosium wire. Measurements were made around a UO_2 rod placed next to the reflector, next to the aluminum plate, and next to another fuel rod. Measurements were made around a UO_2 - PuO_2 rod placed next to the reflector, next to a single water hole, and next to another fuel rod. Summaries are presented for ratios of the peak to minimum flux.⁽⁶²⁾

Bucklings for the loadings were determined by measuring the fission products emitted from the fuel rods. All gamma rays with energies greater than 0.5 MeV were used in the measurements. Decay and wait times were automatically corrected for by using a monitor which was counted at the same time as the rods were counted. Radial bucklings were obtained at 4-6 different vertical positions. Data was obtained every 2 cm along the length of the rod in order to determine axial buckling.

Thin-walled aluminum tubes were placed at interstitial positions to simulate voids in multi-region and single-region cores containing fuel at a lattice pitch of 0.42 in. The reactivity effects and power distribution changes were measured for uniform arrays of voids and as a function of position of a void region.

Both pulse neutron and noise analysis techniques were used to obtain the ratio of the effective delayed neutron fraction

to the neutron lifetime. The Rossi-alpha values were determined using the pulsed neutron technique as a function of axial buckling (H^{-2}) for a configuration of 5.74 wt% $^{235}\text{UO}_2$ fuel. These values were extrapolated to the critical axial buckling. Reactor noise data was obtained for a critical configuration of a 27 x 27 rod array with an inner 19 x 19 rod region of $\text{UO}_2\text{-PuO}_2$.

Studies at the Vallecitos Atomic Laboratory. A series of measurements⁽⁴⁾ of fission and absorption rates in $\text{UO}_2\text{-PuO}_2\text{-H}_2\text{O}$ lattices has been performed at the Vallecitos Atomic Laboratory of General Electric Company under the joint USAEC-EURATOM Research and Development Program. Subcritical lattices were assembled inside a 304-SS pressure vessel, which could be heated and was capable of achieving water temperatures up to 540°F at pressures up to 1050 psig. The lattices were driven with fission neutrons originating in a fission plate, located on top of the Nuclear Test Reactor (NTR) and directly beneath the pressure vessel. The lattices were square-pitch and had water-to-fuel ratios of 3:1 and 2:1, with multiplications of 0.965 and 0.969 respectively. The fuel was natural $\text{UO}_2\text{-1.5 wt% PuO}_2$ (16.3% ^{240}Pu), in the form of sintered pellets 0.49 in. dia. assembled in rods with a fuel length of 24 inches, clad with 0.032 inches of Zircaloy. A total of 525 standard rods and 20 special rods were fabricated.

Irradiations were performed at five temperatures with each lattice. Fission rates and absorption rates were measured inside the fuel rod, at the rod surface and in the moderator

between rods. Information about the neutron spectrum was obtained from the activation of selected absorbers. The absorbers, Lu, ^{168}Yb , ^{139}La , ^{197}Au , ^{191}Ir , ^{193}Ir , and ^{63}Cu were prepared in dilute form to reduce resonance self shielding. Fission rates were measured using very thin foils of highly enriched ^{235}U (97.4%) and ^{239}Pu (99.7%).

Relative activation rates for copper and the various resonance absorbers have been calculated using the THERMOS code and a modified version of THERMOS called EPITHERMOS for comparison with the experimental results. Agreement was quite good at 70°F but the calculations overpredicted the change in the resonance-to-1/v ratios at higher temperatures. The measured ratios of ^{239}Pu fissions to ^{235}U fissions were larger than the calculated ratios in all cases, but differences were within the assigned uncertainties.

3. Experiments Using D₂O or Graphite

Studies at Pacific Northwest Laboratory. Measurements made with plutonium fuels and D₂O moderator in the Plutonium Recycle Critical Facility (PRCF) are summarized in Table XIV. The measurements have been made in support of the operations of the Plutonium Recycle Test Reactor (PRTR). Therefore, the experiments have used UO₂, Al-Pu, and UO₂-PuO₂ fuel elements which ultimately have been irradiated in the PRTR. Studies conducted in the PRCF and PRTR previous to 1964 have been summarized.^(65,66) All of the experiments have been conducted with 19-rod clusters of the fuel arranged on a 8 in. lattice

TABLE XIV

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spacing. All fuel elements hung directly into the moderator of the Plutonium Recycle Critical Facility except for the experiments which use the UO_2 -2.0 wt% PuO_2 fuel elements. The critical loading results include perturbations from safety rods and control rods. The safety rods leave three void regions in the cores when the rods are withdrawn. The control rods leave a void and cadmium in the core in their withdrawn position.

Experiments^(67,68) have been conducted in D_2O to obtain critical loadings, kinetics parameters, reactivity coefficients and activation data for a loading which contains only Al-Pu fuel. The ratio of effective delayed neutron fraction to the neutron lifetime was deduced from reactor noise data. The transient occurring after a negative insertion of reactivity was studied also. Moderator level coefficients of reactivity were measured for levels between the top of the fuel and 8 in. above it. The coefficient was also measured at one point 8 in. below the top of the fuel for comparison with similar measurements with other cores.

Spatial neutron fluxes in the axial direction were obtained with bare and cadmium covered BF_3 chambers. A radial traverse was obtained using gold foils and a BF_3 chamber. The flux distribution in a fuel cluster and in the moderator adjacent to the cluster was determined using lutetium foils. Spectral indices which characterize the ratio of epithermal to thermal neutrons and the thermal-neutron distribution have been

deduced from the data. The parameters were obtained using corrections which allow for the depletion of neutrons in the energy region of the 0.3 eV resonance.

The interaction of safety rod worths was measured by determining the worth of the rods separately and as a unit. The worths were obtained by comparing flux transients calculated for step changes in reactivity to that which occurs after insertion of the safety rods. The relative worth of a cadmium covered BF_3 chamber as a function of vertical position was obtained by measuring changes in subcritical multiplication as determined from another BF_3 chamber.

Measurements of kinetics parameters, reactivity worths, and flux distributions were made⁽⁶⁹⁻⁷¹⁾ in zoned loadings of Al-Pu and natural UO_2 clusters. The ratio of effective delayed-neutron fraction to neutron lifetime was determined from reactor noise data. The flux transient from the insertion of negative reactivity was also studied.

Moderator level coefficients of reactivity were determined⁽⁷²⁾ for levels from 8 in. above to 28 in. below the top of the fuel. A value for the migration area of the core was deduced from the data. Temperature coefficients of reactivity were determined⁽⁷³⁾ from reactivity measurements as the temperature was raised from 25-41°C in five steps.

Spatial flux distributions were measured⁽⁷⁹⁾ in the vertical and radial directions using bare and cadmium covered BF_3 chambers. Gold foils were used to obtain activation data in

the vertical and radial directions. Spectral indices which describe the epithermal and thermal neutrons were obtained⁽⁷⁴⁾ at the center of the core in a single Al-Pu rod with an H₂O environment and in a central rod of a UO₂ cluster in an air environment.

The reactivity worth of an Al-Pu and UO₂ cluster irradiated in the Plutonium Recycle Test Reactor was measured as a function of irradiation. The measurements were made relative to an unirradiated standard. In addition, the reactivity worth of several unirradiated UO₂-1.0 wt% PuO₂ clusters and one unirradiated UO₂-0.9 wt% PuO₂ clusters were measured.⁽⁷¹⁾

A strong anti-shadowing effect was determined⁽⁷⁵⁾ for the safety rods by measuring the worth of the rods separately and as a unit. The worths were obtained by comparing flux transients calculated for step changes in reactivity to that which occurred after insertion of the safety rods.

The configuration for a critical loading and the spatial flux distribution using gold foils were determined⁽⁷⁶⁾ for a loading that did not contain the cadmium from the control rods.

A series of measurements has been conducted⁽⁷⁷⁾ in the Plutonium Recycle Critical Facility (PRCF) to provide information for verifying calculational methods used in the pre-experiment analysis of the Batch Core Experiment in the Plutonium Recycle Test Reactor (PRTR). The fuel elements contained UO₂-2.0 wt% PuO₂ and are currently being irradiated in the PRTR

in the Batch Core Experiment. The following parameters have been investigated in the PRCF experiments:

- moderator level coefficient of reactivity
- boron worth
- fuel element worth
- flux distributions
- coolant void and H_2O coefficient of reactivity
- temperature coefficient of reactivity and
- kinetic parameters.

The quantities measured in the experiments have been compared with values calculated⁽⁷⁷⁾ during the pre-experiment analysis.

These comparisons have increased confidence in the values obtained from the pre-experiment analysis of the Batch Core Experiment in the PRTR.

The moderator was initially clean D_2O (99.75%) to which fuel clusters-thimble- D_2O coolant assemblies were added, one at a time, on an 8-in. triangular lattice pitch, to reach a critical loading in the PRCF. Fuel-thimble-coolant assemblies were added to reduce the critical moderator height to levels below the top of the fuel. Then boric acid, enriched to 92% in the ^{10}B isotope was added to the moderator in small increments to raise the critical moderator height to the top of the fuel again but now at the increased core size. Critical core loadings were determined for concentrations of ^{10}B in the D_2O moderator ranging from 0 to ~11 wppm. During each sequence of fuel and boron addition, measurements were made of the reactivity worth of the fuel, boron and moderator level changes.

The reactivity worth of an incremental increase in the height of the moderator was determined as a function of the critical moderator height, over a wide range of critical heights, boron concentration and core sizes, and formed the basis for other reactivity worth measurements. Measurements were made of radial and axial flux profiles for boron concentration of 0.93 and 4.64 wppm ^{10}B , by activating bare and cadmium-covered gold foils. Cell traverses were obtained with bare and cadmium-covered lutetium foils in cores containing 4.64 and 11.25 wppm ^{10}B in the moderator. The reactivity worth of the removal of D_2O coolant and H_2O coolant from inside the thimble containing the central fuel cluster was measured. Temperature coefficients between 20 and 70°C were determined for the cores containing 4.64 wppm and 11.25 wppm ^{10}B in the moderator. The ratio of delayed-neutron fraction to neutron lifetime was measured with 0.93, 4.64 and 11.25 wppm ^{10}B in the moderator using reactor noise techniques. The ratio also was measured using reactor oscillation techniques after eleven Al-1.8 wt% Pu fuel elements were added to the periphery of the core containing 11.25 wppm ^{10}B .

Measurements^(80,88) using the Physical Constants Testing Reactor are summarized in Table XV and have involved the use of similar types of fuel with various plutonium concentrations and at% ^{240}Pu as was the case for the Critical Approach Facility and the Plutonium Recycle Critical Facility (PRCF). Measurements are made of the neutron reaction rates in a test sample

TABLE XV

REACTIVITY MEASUREMENTS WITH PLUTONIUM-ENRICHED FUELS IN THE PCTR

<u>Fuel Type</u>	<u>wt% Pu</u>	<u>at% ^{240}Pu</u>	<u>Type of Measurement</u>	<u>Ref.</u>
Al-Pu	1.8	6.0	k_{∞} -1 and microscopic parameters for:	
			(1) 3 Al-Pu rods in a 19-rod UO_2 cluster, in 3 D_2O lattices	(80)
			(2) 19-rod cluster of Al-Pu in a "Supercell" with six UO_2 clusters, in one D_2O lattice.	(81)
			(3) 19-rod clusters in 3 graphite lattices.	(82)
	1.9&2.2	6&16	Fuel temperature coefficient of k_{∞} for 19-rod clusters in graphite, $T \leq 400^\circ\text{C}$.	(83)
	2.2	20	k_{∞} -1 and microscopic parameters for 19-rod clusters of Al-Pu, in three graphite lattices.	(84)
	Variable	6&16	Effective Resonance Integral of ^{240}Pu in graphite.	(85)
(Depl) $\text{UO}_2\text{-PuO}_2$	0.9	8	k_{∞} -1 and microscopic parameters for:	
			(1) 19-rod clusters in one graphite lattice	(86)
			(2) Single rods, in two graphite lattices	(87)
			(3) Single rods, in one H_2O lattice	(88)(89)
(Depl) $\text{UO}_2\text{-PuO}_2$	2.6	29	k_{∞} -1 and microscopic parameters for single rods in two graphite lattices	(87)
(Nat.) $\text{UO}_2\text{-PuO}_2$	2.0	8	k_{∞} -1 and microscopic parameters for:	(88)
			(1) Single rods in two graphite lattice.	
			(2) Single rods in one H_2O lattice	
			(3) Particle size studies, Pu particles ranging from 0 to 350 microns.	

by using foil activation ratio techniques and the neutron multiplication factor of the test materials is obtained by a null-reactivity technique. A typical test volume is small compared to the test volume required for an exponential assembly and allows the characteristics of a reactor cell to be studied in the zero leakage situation which is close to that of the actual power reactor. This approach gives important experimentally verifiable reactor physics parameters at a very low investment in reactor test materials.

The majority of the studies have been accomplished using graphite as the moderator. However, the capability of the Physical Constants Testing Reactor is not limited to graphite moderated systems. The reactor has been operated with a D₂O calandria in the core in design studies for the Plutonium Recycle Test Reactor, and with an H₂O calandria to obtain data which supplement that obtained from the Plutonium Recycle Critical Facility.

The H₂O experiments included^(88,89) measurements of $k_{\infty}-1$ for UO₂-0.9 wt% PuO₂ rods in a square lattice with a center-to-center spacing of 1 inch; and UO₂-2 wt% PuO₂ rods in a hexagonal lattice with a center-to-center spacing of 1.0746 inches. The moderator-to-oxide volume ratios of the square lattice and the hexagonal lattice are equivalent and equals 3.69. The former experiment used natural boron dissolved in the H₂O moderator to reduce the multiplication of the test cell and its

neighboring cells to unity. Subsequent experiments will provide correlations between the zero leakage test cell results and critical assembly results from the PRCF.

Other results from experiments using the PCTR which are of significance for H_2O reactor studies are the effective resonance integral⁽⁸⁵⁾ of ^{240}Pu and, the fuel temperature coefficient⁽⁸³⁾ of k_∞ for an Al-Pu-graphite lattice, with two concentrations of ^{240}Pu . These two measurements, which are relatively independent of moderator characteristics, provide valuable insight into the problems of resonance absorption in ^{240}Pu and provide data with which to test the computational methods.

Another important measurement in the series involves the reactivity effect due to a change of PuO_2 -particle size in the uranium oxide. The fabrication of fuel rods has a cost incentive for non-homogeneity and the fuel fabricator would like to make particle sizes as large as compatible with the heat transfer and physics characteristics. The changes in reactivity due to the differences in the size of the PuO_2 -particles in the mixed oxide reactor fuels are difficult to predict and can be important in the overall evaluation of a reactor loading. For this reason a series of fuel rods containing UO_2 -2.0 wt% PuO_2 has been prepared which vary in PuO_2 particle size. Rods with four different particle sizes between 0 and 350 microns have been prepared using plutonium containing 8 wt% ^{240}Pu . Two of the sizes (0 and 100 microns) have also been prepared using plutonium which contains 24 wt% ^{240}Pu . Results in a graphite lattice show

the mass of copper required for $k_{\infty} = 1.0$ decreases as the particle size increases. Theory-experiment correlations are being made by using the computer code GRAINS.⁽⁹⁰⁾

A valuable by-product of the k_{∞} -1 measurements are activation profiles in the test lattices for various resonance and thermal absorbers and for several fission foil materials. These microscopic parameters are very useful for testing the adequacy of the calculational methods as applied to these zero leakage systems.

Studies at Savannah River Laboratory. Experiments were conducted⁽⁷⁹⁾ in D_2O at Savannah River Laboratory using fuel rods containing mixtures of PuO_2 and UO_2 which simulate natural UO_2 fuel irradiated to 5000 MWd/ton. The fuel contained 0.312, 0.269, and 0.376 wt% PuO_2 in depleted UO_2 , and the ^{240}Pu isotopic concentrations were 20, 6, and 6 wt% of the plutonium, respectively. The changes in lattice parameters were studied by measuring bucklings and neutron reaction rates. Both 19- and 31-rod clusters with D_2O , gas, and organic coolants were studied at triangular lattice pitches of 9.33 and 12.12 in. Uniform lattice bucklings were determined by substitution measurements in natural UO_2 lattices and changes in buckling induced by uniform lattice heating were determined in exponential measurements.

Thermal-neutron distributions were measured by irradiation of Mn foils. Spectral indices were deduced from activation of

Lu-Mn and ^{239}Pu - ^{235}U foils, resonance neutron capture and fast fissions using ^{238}U foils, all fissions using ^{235}U foils and fissile conversion ratios using ^{238}U and ^{235}U foils.

4. Synopsis

The experimental information available on the reactor physics behavior of plutonium-fueled uniform lattices covers a wide range of moderator-to-fuel ratios and fuel concentrations but a limited range of parameters (e.g., critical masses). The data on the critical masses of plutonium fuel covers 75 lattices and the majority of these are H_2O moderated. The critical mass data for plutonium-only fuel cover essentially two concentrations and compositions of plutonium. The critical mass data for UO_2 - PuO_2 fuels covers four concentrations of plutonium and three plutonium compositions.

The available experimental information on the reactivity coefficients and kinetics parameters is quite sparse. The information on moderator temperature and void coefficient of reactivity cover three plutonium compositions and the lattice spacing is different for each experiment. Experimental values of moderator temperature coefficients are presented in Table XVI to illustrate the information presently available. The data is limited to temperatures less than 70°C and has been measured at a single lattice pitch for a given fuel. Results of analytical correlations are also given ⁽¹⁰²⁾ in the table. Because of the data limitations, the only conclusion that can be drawn from this

study (Figure 1) is that a discrepancy between calculational and experimental results exists which is a function of the fissile atom content in the fuel, and that this discrepancy is independent of the fissile material. In order to determine some of the inadequacies of the methods in this area, a systematic set of data is required in which one parameter is varied and the others are held constant. The experimental information on the kinetics parameters also is quite meager. The results of measurements for the ratio of effective delayed neutron fraction to neutron lifetime $\hat{\beta}/\lambda$ available for plutonium fueled lattices are summarized in Table XVII. It is of interest to note the variation of $\hat{\beta}/\lambda$ with fuel type and moderator type. The ratio is smaller for fuel containing only plutonium because of the lower value of $\hat{\beta}$. Values of the ratio are lower for D_2O moderated reactors relative to H_2O moderated reactors because of the larger neutron lifetime, λ , in the D_2O loadings. Further measurements in which only one parameter (e.g., lattice pitch) is varied and others are held fixed are desirable.

TABLE XVI
Summary of Moderator Temperature Coefficients

Reactor System (All Clean-Lattice H ₂ O- Moderated Criticals)	$\frac{1}{k_{\text{eff}}} \frac{\Delta k_{\text{eff}}}{\Delta T}$, ($\Delta k/k/^\circ\text{C}$) at $\sim 40^\circ\text{C}$	
	Experiment	Calculation
2.73% Enriched UO ₂ (Yankee)	-6.9×10^{-5}	-8.0×10^{-5}
5.74 wt% Enriched UO ₂ (Saxton)	-5.5×10^{-5}	-9.0×10^{-5}
2 wt% PuO ₂ -UO ₂ (24% ²⁴⁰ Pu)	-2.7×10^{-5}	-1.6×10^{-5}
2 wt% PuO ₂ -UO ₂ (8% ²⁴⁰ Pu)	-2.4×10^{-5}	-4.6×10^{-5}
6.6 wt% PuO ₂ -UO ₂ (Saxton)	-7.8×10^{-5}	-13.1×10^{-5}
1.5 wt% PuO ₂ -UO ₂ (EBWR)	-2.2×10^{-5}	$+0.8 \times 10^{-5}$

FIGURE 1
 Calculational-Experimental Discrepancy in
 Moderator Temperature Coefficient

TABLE XVII

Values of the Ratio $\hat{\beta}/\lambda$ of Effective Delayed
Neutron Fraction to the Neutron Lifetime

<u>Fuel</u>	<u>H/Pu</u>	¹⁰ B	$\hat{\beta}/\ell(\text{sec}^{-1})$	
		<u>wppm</u>	<u>Experimental</u>	<u>Analytical</u>
<u>H₂O Moderator:</u>				
Al-1.8 wt% Pu	810	0	33 $\pm$ 3	29
0.16 wt% ²³⁵ UO ₂ -1.5 wt% PuO ₂	567	0	71 $\pm$ 6	71
0.71 wt% ²³⁵ UO ₂ -2.0 wt% PuO ₂				
8% ²⁴⁰ Pu	294	0	127 $\pm$ 11	113
24% ²⁴⁰ Pu	294	0	114 $\pm$ 15	127
0.71 wt% ²³⁵ UO ₂ -6.6 wt% PuO ₂				
and 5.74 wt% ²³⁵ UO ₂		0	292	
5.4 wt% ²³⁵ UO ₂		0	240	
<u>D₂O Moderator:</u>				
Al-1.8 wt% Pu		0	3.6 $\pm$ 0.3	3.1
Al-1.8 wt% Pu				
and 0.71 wt% ²³⁵ UO ₂		0	6.3 $\pm$ 6	5.8
0.71 wt% ²³⁵ UO ₂ -2.0 wt% PuO ₂		0.93	5.0 $\pm$ 0.3	5.0
		4.64	9.4 $\pm$ 0.5	9.7
		11.25	12.9 $\pm$ 0.7	13.1
0.71 wt% UO ₂ -2.0 wt% PuO ₂				
and Al-1.8 wt% Pu		11.25	10.7 $\pm$ 0.4	

C. Reactor Studies

The accurate prediction of fuel burnup and reactor core lifetime is important to the production of low-cost power with nuclear power systems. Reasonably accurate predictions of initial loading, depletion rate and final fuel exposure can result in savings of fuel cycle and plant depreciation costs. Thus it is important to conduct studies to determine reactor burnup behavior and to obtain experimental data for testing the accuracy of burnup calculational methods. For these reasons experiments are being conducted to measure the burnup behavior of loadings of plutonium fuels in the Experimental Boiling Water Reactor (EBWR), the Saxton Reactor, and in the Plutonium Recycle Test Reactor (PRTR). Also, fuel burnup studies utilizing clusters of specific fuel composition have been made and are continuing in the Plutonium Recycle Test Reactor. The information obtained from special fuel rods irradiated in the Vallecitos Boiling Water Reactor and the EBWR also provided useful information.

1. EBWR Plutonium Experiment

An irradiation experiment⁽⁹⁴⁻⁹⁶⁾ was conducted in the Experimental Boiling Water Reactor⁽⁹⁷⁾ at Argonne National Laboratory. This experiment was part of a joint Argonne National Laboratory and Pacific Northwest Laboratory program to demonstrate the utilization of plutonium in a boiling H₂O power reactor and to obtain useful physics information on the behavior of a plutonium fuel in such a reactor system. The characteristics of the reactor loading is summarized in Table XVIII. The experiment was

TABLE XVIII

CHARACTERISTICS OF THE LOADING IN THE EBWR PLUTONIUM EXPERIMENT

<u>Zone</u>	<u>Fuel Type</u>	<u>Enrichment</u>	<u>Number of Pins per assembly</u>	<u>Number of Assemblies per Zone</u>
Central	UO ₂ -PuO ₂ *	1.5 wt% PuO ₂ in depleted (~0.165 at%) UO ₂	36 (6x6 Array)	36
Driver Zone	ZrO ₂ -UO ₂	Fully Enriched UO ₂	49 (7x7 Array)	8
Shim Zone	UO ₂ Containing Burnable Poisons	6 wt% ²³⁵ U (0.158 wt% Eu ₂ O ₃ 0.0288 wt% Sm ₂ O ₃)	36 (6x6 Array)	52
Outer Zone	UO ₂	Natural U	36 (6x6 Array)	52

*Certain assemblies contained special pins which were, natural UO₂,
Al-3.4 wt% Pu with 8 and 26% ²⁴⁰Pu, and UO₂-1.5 wt% PuO₂ with 20 and
26% ²⁴⁰Pu.

originally intended to be similar to the Plutonium Recycle Test Reactor Batch Core Experiment, but for a boiling water reactor. However, because of the priority of other work it was not possible to carry out this experiment to the extent that was originally planned.

The experiments which were conducted are summarized in Table XIX. Zero power startup experiments⁽⁹⁸⁾ were conducted with the plutonium zone and as a whole core was loaded. At three different increments of core exposure, standard and special rods were removed for nondestructive and destructive burnup analyses. These rod types are listed at the bottom of Table XVIII. When the experiment was terminated the critical mass determination was made for the plutonium zone and a full 36-rod element of fuel was removed for nondestructive gamma-scanning and destructive burnup analysis.

Gamma-scanning measurements were made on an intact 36-rod bundle⁽⁹⁹⁾ as well as on each of the individual rods.⁽¹⁰⁰⁾ The scanning results include axial power profiles on each rod, rod-to-rod power inter-normalizations, and fission product gamma-ray spectra taken with NaI(Tl) and high resolution Ge(Li) detectors. The first set of rods have been sampled destructively to obtain concentrations of key fission products and fuel isotopes. Sampling and analysis of the final set of rods is in progress.

TABLE XIX

MEASUREMENT SUMMARY FOR EBWR AND SAXTON PLUTONIUM IRRADIATION EXPERIMENTS

Core Loading	Average Exposure Plutonium Zone (MWd/MTU)	Measurement													
		Critical Mass	Boron Worth	Fuel Worth	Control Rod Worth	Kinetics	Activation Distributions	Core Depletion	Temperature Coefficients	Power Coefficient	Pressure Coefficients	Short-Term Fission Products	Fuel Sampling	Gamma Scanning	Destructive Analysis
EBWR EXPERIMENT															
Pu Zone	0	✓	✓	✓	✓		✓		✓						
Pu Zone	~2950	✓													
Full Core	0		✓		✓	✓	✓		✓						
Full Core	~1500 at 42 MW						✓						/	/	*
Full Core	~1500 at 70 MW						✓								
Full Core	~2650												✓	✓	*
Full Core	~2950												✓	*	*
SAXTON EXPERIMENT															
Full Core	0		✓	✓	✓	✓	/		✓	✓	✓	✓			
Full Core	~4500		✓		✓			✓	✓		✓	✓	✓	✓	✓
Full Core	6112		✓		✓				✓						
Full Core	9364		✓		✓				✓						

*Measurements not completed.

2. Saxton Plutonium Experiment

An irradiation experiment is being conducted in the Saxton Reactor under the sponsorship of USAEC-EURATOM, the Westinghouse Electric Corporation, and the Saxton Nuclear Experimental Corporation. The basic objective of this experiment, The Saxton Plutonium Program, is to develop information concerning the utilization of plutonium-enriched fuel in a typical pressurized water reactor system through design, fabrication and operation of a partial core of $\text{UO}_2\text{-PuO}_2$ fuel in the Saxton reactor.⁽³⁾ The program includes; post-irradiation fuel examination, post-operation evaluation of the performance of core materials, and an evaluation of the capability of core design methods.

The characteristics of the reactor loading are summarized in Table XX. To be able to compare fuel performance, the loading includes fuels made by two different fabrication processes and which are clad in Zircaloy-4 and 304 stainless steel. Two assemblies contain vibrationally-compacted $\text{UO}_2\text{-PuO}_2$ and seven assemblies contain pelletized $\text{UO}_2\text{-PuO}_2$. Core nuclear parameters were determined at startup⁽¹⁰¹⁾ and at various increments of core exposure. A summary of these measurements is given in Table XIV. The average fuel exposure for the plutonium zone as of September 1967 was $\sim 12,200$ MWd/MTM.* The power level was raised from 23.5 MWt to 35 MWt in the fall of 1967 and the reactor will operate at this level until the end of operation of the Saxton partial plutonium core.

*Megawatt days per metric ton of metal (U+Pu).

TABLE XX
CHARACTERISTICS OF THE LOADING
IN THE SAXTON PLUTONIUM EXPERIMENT

<u>Zone</u>	<u>Fuel Type</u>	<u>Enrichment</u>	<u>Number of Pins Per Assembly</u>	<u>Number of Assemblies Per Zone</u>
Central	UO ₂ -PuO ₂	6.6 wt% PuO ₂ in natural UO ₂	81 (9x9 Array)*	9
Outer	UO ₂	5.74 wt% ²³⁵ U	81 (9x9 Array)*	12

*9 rods are eliminated from one outer corner of each assembly to allow room for cruciform control rods positioned between assemblies.

3. Batch Core Experiment

One of the main experiments which is currently being conducted in the Plutonium Recycle Test Reactor is called the Batch Core Experiment.^(78,103,104) This experiment involves the study of the characteristics of a plutonium batch core as a function of irradiation. The analytical correlations and experimental studies conducted for the Batch Core Experiment will determine the limitations in the theoretical methods for cross-sections used to predict core lifetime, isotopic inventory, transient behavior, and reactivity coefficients of plutonium-fueled thermal reactors.

The Plutonium Recycle Test Reactor⁽¹⁰⁵⁾ is a D₂O-moderated and cooled pressure tube reactor, with positions for 85 fuel

elements. The central zone of the current loading is called the Batch Core and contains UO_2 -2 wt% PuO_2 19-rod clusters. Surrounding this central zone is a buffer zone, an outer region which contains predominately the same type of elements. The central zone remains unperturbed and the reactor physics characteristics of this zone are studied as a function of irradiation. This experiment utilizes a simple reactor geometry and is a controlled environment with high specific powers for the mixed oxide fuel. There are no control rods or shim rods perturbing the core since the reactor operates on the moderator level principle and has boron dissolved in the moderator to compensate for high excess reactivity. Reactivity coefficients, the boron concentration for the just critical situation, and the isotopic concentration as a function of exposure are being measured.

As of March 1968, two burnup steps have been completed;⁽¹⁰⁶⁾ one at an average exposure of ~ 1700 MWd/MTU for the 55 elements, and another at ~ 3600 MWd/MTU as an average exposure for the 55-element core. At each burnup step fuel rods are removed from clusters in the core for nondestructive gamma ray scanning and destructive burnup analysis.

The experiments conducted at various burnup steps are summarized in Table XXI. In general, the experiments comprise a measurement of the reactivity worth of an irradiated cluster and unirradiated cluster at a radius of 40.6 centimeters, the moderator boron worth, and the boron concentration for the cold, xenon-free,

MEASUREMENT SUMMARY FOR PTR BATCH CORE IRRADIATION EXPERIMENT

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critical measurement of the Batch Core. Kinetics measurements are obtained using pile oscillator techniques. The reactivity worth of bare and cadmium-covered absorber rods are also measured at positions intermediate between fuel element positions as a function of irradiation. There are two rods containing boron in aluminum, two rods containing hafnium oxide in aluminum, and one containing only aluminum. All rods were measured bare and three of them were measured while covered with cadmium sleeves, and in addition, the worth of fission products are determined for the full core along with other power operation information.

The 55-element core is expected to run out of reactivity at an average exposure of ~ 12000 MWd/MTU. This corresponds to a maximum element exposure of ~ 15000 MWd/MTU averaged over the whole element or approximately 22000 MWd/MTU for maximum exposure at a point. When that occurs the reactor loading will be adjusted in order to irradiate the elements until an exposure more typical of present-day power reactors is reached (e.g., 30-35000 MWd/MTU for the element average).

4. Special Sample Irradiation Studies

A variety of special sample irradiations have been and are being carried out to study the changes in isotopic composition of plutonium-enriched fuels as functions of fuel exposure. Experiments using single rods in the UO_2 - PuO_2 zone in EBWR, a single bundle of UO_2 -1.5 wt% PuO_2 in VBWR and the 19-rod clusters of Al-Pu natural UO_2 , and several enrichments of UO_2 - PuO_2 in the PRTR are described in the following sections.

Single Rods in EBWR. A set of special rods was included in each quadrant of the plutonium zone in the EBWR Experiment. These rods contained natural UO_2 , Al-3.4 wt% Pu with 8 and 26% ^{240}Pu , and UO_2 -1.5 wt% PuO_2 with 20 and 26% ^{240}Pu . These rods were included to provide additional information on the burnup characteristics of the UO_2 - PuO_2 fuel.

The isotopic changes which occur in the UO_2 and Al-Pu fuels provide useful information concerning the buildup and burnout of plutonium in the mixed oxide fuel. The plutonium rods with varying ^{240}Pu content were included to provide information on the burnup characteristics of these types of fuels and to attempt to determine if burnup information for even more highly irradiated fuels could be deduced from these fuels.

VBWR Plutonium Experiment. An irradiation experiment⁽⁵⁾ was conducted in the Vallecitos Boiling Water Reactor (VBWR) under the joint USAEC-EURATOM Research and Development Program at the Vallecitos Atomic Laboratory of General Electric Company. Special fuel sections were installed in the lower section of a 16-rod square bundle in the VBWR. The fuel was in the form of sintered pellets of UO_2 - PuO_2 , clad in Zircaloy-2. About 10% of the PuO_2 was segregated in particles ranging in size from 0.005 to 0.020 inches in radius, with most of the particles near the 0.010 inch size. The specific fuel description is contained in Table XXII.

TABLE XXIIUO₂-PuO₂ BURNUP EXPERIMENT IN VBWR

Pellet Diameter: 0.373-0.376 in. dia.

Fuel Length: 9.390 ± 0.060 in.

Average Composition: 1.47 wt% PuO₂ in UO₂

Average Weight: 173.2275 g/rod

Average % TD: 93.27%

Initial Isotopic Composition (atom percent)

Uranium: 235 (0.765), 236 (0.0008), 238 (99.234)

Plutonium: 239 (79.468), 240 (17.014), 241 (2.968),
242 (0.550)

Fuel Cladding: Zircaloy-2, 0.022 in. wall thickness.

Flux wires containing dilute concentrations of selected resonance absorbers were included in the test assembly, to obtain activation data from ¹⁷⁶Lu, ¹⁹¹Ir, ¹⁹³Ir, ¹⁹⁷Au and ¹⁸⁶W, relative to ⁶³Cu, for characterization of the neutron spectrum. A relatively constant spectrum throughout the UO₂-PuO₂ assembly was inferred from these data.

One rod was removed from the assembly, and replaced with an unirradiated rod at each of three exposure levels, 1860, 3000, and 5300 MWd/t. The program was terminated at the 5300 MWd/t step, about halfway to the original goal exposure of 10,000 MWd/t.

Three pellets from each rod were destructively analyzed to obtain the new isotopic compositions of uranium and plutonium. The

pellet volumes were divided into inner and outer halves, by drilling out the center half of the pellet, to obtain some indication of the spatial variation of burnup within the pellets.

Concurrent with the burnup experiment, a program was carried on to develop computational methods capable of accurately predicting the spatial variation of burnup in $\text{UO}_2\text{-PuO}_2\text{-H}_2\text{O}$ lattices. Cross sections for the burnup calculation were generated using the THERMOS code for thermal energies, a modified version of THERMOS called EPITHERMOS to treat the ^{240}Pu resonance at 1 eV, and a version of the MUFT code for energies above 0.625 eV. Corrections were computed for self-shielding in Pu particles.

Over the rather narrow exposure range of the experiment, the results of the computations agreed fairly well with the experimental values. The concentrations of higher Pu isotopes tended to be overestimated by a few percent, while the uranium isotopic concentrations tended to be underestimated. It was concluded that the computational methods were fairly adequate and that the amount of plutonium segregation in the $\text{UO}_2\text{-PuO}_2$ fuel did not compromise the analysis of the experiment.

Single Clusters in PRTR. Concurrent with the past and current experiments, which have been conducted in the Plutonium Recycle Test Reactor, have been the controlled irradiation of special clusters. The type of clusters which have been irradiated, and in some cases have already been nondestructively and destructively

analyzed for reactor burnup data, are summarized in Table XXIII. A series of these elements has been irradiated to various exposures in the Plutonium Recycle Test Reactor and then nondestructively analyzed by gamma scanning techniques and destructively analyzed for reactor burnup data. Some clusters are still undergoing further irradiation in the Batch Core Experiment.

The plutonium-aluminum fuel which contains either 7 or 17 at% ^{240}Pu has been irradiated to approximately 52% depletion of the original plutonium. The data collection for these elements is completed⁽¹⁰⁷⁾ except for some nondestructive gamma scanning. In addition to the isotopic data, effective cross section ratios^(108,110) have been deduced for the fuel and used in the correlation of the calculational methods. The effective cross section ratios and the isotopic data can be used to evaluate independently the methods and the cross sections used to calculate the reactor burnup. In addition, effective cross section ratios are a more sensitive test of the methods than the multiplication for the isotopic data.

5. Special Analysis Methods

The determination of power distribution information and isotopic compositions as functions of exposure are important results obtained from irradiation experiments. Correlation of these data with predictions based on calculation is an important use of the results. The following sections present some discussion about special kinds of information that are obtained by non-

TABLE XXIII

PRTR SINGLE ELEMENT IRRADIATIONS

<u>Fuel Type</u>	<u>Cluster</u> <u>Average Exposure Range</u>	<u>Goal Exposure</u>	<u>Experiment Status (5/1/68)</u>
Al-1.8 wt% Pu 6.25 at% ^{240}Pu	13.8-109.5 MWd (~125 MWd peak) (~50% Depletion)		Irradiation Completed. Analysis Completed and Reported.
Al-2.6 wt% Pu 16.45 at% ^{240}Pu	35.0-160.3 MWd (~200 MWd peak) (~52% depletion)		Irradiation Completed. Analysis Completed and Reported.
Natural UO_2	850-7000 MWd/MTU (~8850 MWd/MTU peak) 11,100 MWd/MTU (~14,100 MWd/MTU peak)	15,000 MWd/MTU	Irradiation Completed. Analysis in Progress. Irradiation Continuing in PRTR Fringe.
UO_2 -0.48 wt% PuO_2 16 at% ^{240}Pu	2000-5000 MWd/MTU (~6600 MWd/MTU peak) 8600-12,700 MWd/MTU (~16,700 MWd/MTU Peak)	15,000 MWd/MTU	Irradiation Completed. Analysis in Progress. Irradiation Continuing in PRTR Fringe.
UO_2 -1 wt% PuO_2 8 at% ^{240}Pu	1100-4300 MWd/MTU (~6000 MWd/MTU Peak) 7700-9400 MWd/MTU (~13,300 MWd/MTU peak)	15,000 MWd/MTU	Irradiation Completed. Analysis in Progress. Irradiation Continuing in PRTR Fringe.
UO_2 -2 wt% PuO_2 8 at% ^{240}Pu	~5000 MWd/MTU (~9000 MWd/MTU peak)	>15,000 MWd/MTU	

destructively scanning the gamma-ray energies and spatial distributions of fission products and by destructive burnup analysis of the irradiated fuel.

Gamma Scanning of Irradiated Fuel. Nondestructive analysis of irradiated fuel is accomplished with a gamma-ray scanner located in the fuel storage basin of the Plutonium Recycle Test Reactor. This instrument is used for scanning fuel rods, clusters, and bundles of rods which have been irradiated in power reactors. The scanning results include axial power profiles on each rod, rod-to-rod power inter-normalizations, and fission-product gamma-ray spectra. The data is obtained with NaI(Tl) and high resolution Ge(Li) detectors. The potential and possible application of this information can be described with typical results from measurements using the fuel irradiated in the EBWR.^(99,100)

An axial scan of gamma rays emitted from several rods irradiated in the EBWR is shown in Figure 2. These rods were irradiated in three 36-rod bundles and arranged diagonally through 1/4 of the plutonium zone. Information about the void content along the length of the rod as well as depressions due to control blades which were inserted in the core can be obtained from the scan. In addition, plots of data from the same height on each of the rods allow radial core traverses to be developed. By properly choosing the fission products which are used in the scan, it is possible to draw conclusions about the effects of the irradiations which have been integrated over the total irradiation as opposed to those which have just occurred in the near-term.

A scan of the ^{95}Zr relative activity around one side of the bundle of rods which has been irradiated in the Experimental Boiling Water Reactor, is shown in Figure 3. Three things of importance can be observed in this scan. First, the lower power generation which has occurred in a natural UO_2 rod relative to the neighboring mixed oxide rods is determined by the scan. Secondly, a higher power has been generated in the two corner $\text{UO}_2\text{-PuO}_2$ rods relative to that generated in other mixed-oxide rods because of their positions near the corners of the bundle where there is greater flux peaking due to water gaps. Finally, an increase in power generation in one rod is observed because of the power shape of the reactor core.

Another potential use of the gamma-scanning technique is energy analysis using high resolution Ge(Li) detectors. It has been suspected for some time that the products produced in fission contain a lot of information about the number of fissions that have occurred, in what fissile material they occurred, and even the average flux that the fuel was subjected to. It was not until the advent of Ge(Li) detectors that the potential for obtaining some of this information nondestructively could be investigated. Part of the gamma-ray energy spectra emitted by an enriched UO_2 rod irradiated in the EBWR is shown in Figure 4. Things of importance to note are the peaks due to:

- $\text{Zr}^{95}\text{-Nb}^{95}$ and Cs^{137} which have approximately the same yield for uranium and plutonium fissions,

FIGURE 2

Axial Gamma Scans of Radial Rods

FIGURE 5
ENERGY SPECTRA FROM Al-Pu ROD

- Rh^{106} which has a higher yield for plutonium fissions, and
- Ce^{144} - Pr^{144} which have a higher yield for uranium fissions.

Similar results for a rod which contains plutonium only is shown in Figure 5. The effect of the different fission yields can be seen by comparing this spectrum with that of Figure 4. These are

- the higher peak of Pr^{144} relative to Cs^{137} in the uranium rod and
- the higher Rh^{106} peak in the plutonium fuel.

In addition, there are unlabelled peaks which appear only in the spectrum from the uranium fuel. These peaks are present because of the burnable poison that was put in the enriched uranium fuel. Peaks such as these show the potential for using burnable poisons as exposure monitors.

Both the spatial and energy scanning using the non-destructive techniques have great potential for use in evaluating methods which will have to be used in determining how to place plutonium in the present-day uranium-fueled power reactors. In addition to developing the potential applications, the results are useful for optimizing the destructive sampling and maximizing the reactor power for fuel irradiations.

Deductions of Effective Cross Section Ratios. A very powerful least squares fitting tool⁽¹⁰⁸⁾ which uses a multi-parameter, non-linear regression analysis has been developed⁽¹⁰⁹⁾ to deduce effective cross section ratios⁽¹¹⁰⁾ from isotopic burnup data. This analysis is accomplished simultaneously in a multi-variable

space and allows for the correlation of errors between variables.

The hypothetical solution of a burnup problem in three-dimensions is shown in Figure 6. Each of the axes correspond to one of the isotopic concentrations which describe a data point; either ^{239}Pu or ^{240}Pu or ^{241}Pu . The ^{242}Pu axis is a fourth dimension and does not appear in the figure but is present in both measurement and theory. If a hypothetical sample starts out containing initially only ^{239}Pu , it might as a function of irradiation describe the helix shown in Figure 6 until at some long time later it would end up at the origin or at least in a dimension not included on the figure. If at various times the sample is destructively analyzed for its isotopic content, each data point would describe a point in this space since it would have a ^{239}Pu , ^{240}Pu , ^{241}Pu , etc. concentration. This data from a single sample defines a measurement vector, $\underline{y}$. The error on the data point is described by the errors on each of the isotopic concentrations so each data point is surrounded by an ellipsoid of error. There is a correlation between the errors, which is analytically described by a variance-covariance matrix. The ellipsoid represents the one standard deviation error region in the same sense as does the error flag for the more familiar and simple least-squares problem which has an error assigned to only one of the variables.

By properly choosing the functional shapes which are allowed for the effective cross-section ratios, the solutions to the

FIGURE 6

Three Dimensional Representation of a Statistical Analysis
for Plutonium Burnup

burnup equations can be written in closed form and a least squares fit made to the data points. The results of such a fit would be typified by the helix $g = 0$ and the standard deviations resulting from the goodness of fit to the experimental data would be represented by the surface of the helical tube. The deviations of the data point from the helix are described by vectors dy . The projection of the helical tube onto a plane described by only two of the axes would show the variation of one of the concentrations, e.g., the ^{240}Pu plotted against ^{239}Pu concentration in a two-dimensional representation.

This technique has been applied⁽¹⁰⁸⁾ to the data from the Al-2.6 wt% Pu (HxMC fuel) and the Al-1.8 wt% Pu clusters irradiated in the Plutonium Recycle Test Reactor. The HxMC fuel initially contained 17 at% ^{240}Pu and some of the samples were irradiated to approximately 52% depletion of the original plutonium. A two-dimensional representation of the least squares result is shown in Fig. 7.

In this case, the data point is described by two parts of the total information which makes up the data point in the multi-dimensional space. The dark line is the two-dimensional projection of the helical curve through the data points and the width of the line is the standard deviation obtained from the goodness of fit. The parameters which result from such a multi-variable fit are the effective-cross-section ratios. These ratios are a more fundamental quantity than isotopic concentrations and can be compared directly to the corresponding calculated ratios. For the analysis of the

FIGURE 7

VARIATION OF ISOTOPIC CONCENTRATIONS
WITH EXPOSURE FOR Al-2 wt% Pu FUEL

plutonium-aluminum data it was possible to assume that all of the effective cross-section ratios were constant as a function of time except the ratio of the ^{240}Pu cross section to that of ^{239}Pu . As shown in the figure this cross section ratio was assumed to have the functional form which varied inversely as the atom concentration of ^{240}Pu .

The ratio of the effective cross section of ^{240}Pu to that of ^{239}Pu is shown in Figure 8 as a function of ^{240}Pu atom concentration while the ^{239}Pu was depleted to approximately 52% of its original value. Results from analyses of other Al-Pu fuels are shown in the figure along with that from the HxMC fuel. The solid curves are obtained from the simultaneous least squares fit and the bounds of the curves are the standard deviations determined from the fits. The crosses plotted in the curve were obtained by another analysis method⁽¹⁰⁸⁾ which allows the variation of the effective cross-section ratio to be obtained from the data. This is accomplished by deducing ratios over small increments of plutonium depletion. The agreement between the crosses and the curves lend credence to the shape assumed for the simultaneous fit.

In the past, the primary objective of obtaining isotopic concentrations from irradiated fuel has been to use the concentrations of the heavy isotopes to test the accuracy of burnup calculations. Once the comparison between calculated and measured isotopic concentrations is made there is no basic criteria that can be used to state how well the calculated and measured results

FIGURE 8

VARIATION OF $\hat{\sigma}_a^{40}/\hat{\sigma}_a^{49}$ WITH EXPOSURE FOR Al-Pu FUELS

The Lx and Hx notations refer to Al-1.8 wt% Pu (7 at% ^{240}Pu) or Al-2.6 wt% Pu (17 at% ^{240}Pu) respectively. The 0 and MC notations refer to outer or middle and center rods of a cluster respectively.

agree since the concentrations for one isotope may agree better than another. In contrast, values for effective cross section ratios which best describe the data are a criteria. The ratios provide a compact set of experimental numbers which can be compared with those obtained from calculations and provide a more sensitive test of the adequacy of the cross sections and theoretical methods. In addition, correlations of the ratios combined with the usual correlations of concentrations allow the possibility of unraveling problems in the areas which cause uncertainties in the calculated result.

The success of the first attempt to extract cross section information by least squares analysis of burnup data from Al-Pu fuel elements leads to speculation about its application and extension to related problems. One possibility involves the mixture of plutonium and uranium. However, the complexity of the added burnup equations and the requirement of certain functional forms, for the variation of the cross-section ratios, may be too stringent a requirement on the method. In anticipation of the advantages to be gained by succeeding, these problems are worth pursuing.

6. Synopsis

Data which are extremely important to evaluating the calculational methods employed in reactor design have been and are being obtained from reactor studies. Information concerning the core reactivity characteristics initially and as a function of irradiation has been obtained from the plutonium experiments conducted in the Experimental Boiling Water Reactor (EBWR) and

SAXTON Reactor. The non-destructive and destructive analyses of fuels irradiated in the EBWR provide detailed data on the radial and axial power profile as a function of burnup. These data cover a limited range of exposure.

The plutonium Batch Core Experiment being conducted in the Plutonium Recycle Test Reactor (PRTR) is providing significant data on the changes in reactivity and isotopic composition as a function of exposure. This experiment is controlled to yield maximum information on the physics characteristics of plutonium fuel as a function of irradiation.

Data on the change in isotopic composition with exposure have been and are being obtained from special sample irradiations in the EBWR, Vallecitos Boiling Water Reactor, and the PRTR. These data cover a range of plutonium concentrations and exposures. A special method was developed to deduce ratios of effective cross sections from measured isotopic concentrations. This method has been applied successfully in deducing ratios of effective cross sections for Al-Pu fuels irradiated in the PRTR. Work is in progress to determine whether these ratios can be obtained from other fuels irradiated in the PRTR and from fuels irradiated in the EBWR.

IV. CONCLUSIONS

Cross Sections. The precision of the knowledge of the thermal cross-section values of plutonium historically has been distinctly inferior to that of the uranium isotopes. In the past, plutonium has been considered primarily as a minor constituent of thermal-reactor systems and thus there has not been a strong motivation for improved precision. In plutonium-enriched systems, however, the analyst is faced with precisions in the fundamental values which are factors of several worse than he is accustomed to considering in ^{235}U systems. In addition, he is faced with an additional and related uncertainty in calculating reaction rates in reactor spectra because of significantly non- $1/v$ behaviorisms of the thermal cross sections of the principal fissile isotopes, ^{239}Pu and ^{241}Pu , and the complication of the resonance structure of ^{240}Pu and ^{242}Pu .

In the calculation of plutonium-enriched systems the cross sections of the fertile isotope ^{240}Pu are of importance. However, it is probably not generally appreciated that the 2200 m/s absorption cross section of ^{240}Pu is inadequately known. Only two precision measurements of this quantity exist and their values differ by 5.3 percent. In addition, the uncertainties in the parameters of the important 1 eV resonance contribute an uncertainty of about 5% to the infinitely dilute resonance integral. When problems of calculating resonance integrals for reactor systems are considered the calculated ^{240}Pu reaction rates must be uncertain by more than 10 percent. Clearly more monoenergetic cross-section measurements are desirable for this isotope.

In the derivation of best estimates of the values of the 2200 m/s constants of ^{239}Pu and ^{241}Pu the evaluator is faced not only with a paucity of measurements but the fact that a majority of the values must be interpreted from measurements made in reactor spectra. In order to improve the knowledge of the cross-section values of ^{239}Pu and ^{241}Pu one of the more promising approaches would be a comprehensive study of the interpretation of spectrum-averaged measurements with modern calculational methods. In the past such measurements have been almost universally interpreted in terms of the Westcott formalism, i.e., a Maxwellian $1/E$ spectrum, and a universal joining function.

At the present time the International Atomic Energy Agency is sponsoring a new least-squares adjustment which will incorporate the results of a dozen or so measurements completed since 1965. In this evaluation the uncertainty in interpreting a 2200 m/s value due to averaging a Maxwellian spectrum over each cross section will be more carefully evaluated. It should be clearly recognized, however, that this procedure will not assess uncertainties due to deviations from a Maxwellian shape. In addition, there exists the related uncertainty of the possibility of compounding the error by using 2200 m/s values derived on the assumption of a Maxwellian shape and then calculating reaction rates using spectrum calculational programs such as THERMOS. While it can be safely assumed that the forthcoming adjusted constants will be the best estimate of the values at this time, the precisions which will be assigned should be carefully considered.

Lattice Studies. Analytical correlations of the experiments using fuels containing plutonium have been performed by various laboratories with varying degrees of success. Some of these analyses were made using the fundamental mode solution (i.e., using the experimental values of buckling to describe leakage) in computing values of k_{eff} whereas others utilized one-dimensional multigroup diffusion theory solutions. Differences which tend to question the validity of the buckling concept as accurately describing the leakage from these types of assemblies are noted in calculated results. It is difficult to make a direct comparison of the results obtained from the various installations because of the differences in basic cross sections and methods. However, no one particular analysis method gives results which are in systematic agreement with measurement for either a single set (i.e., one particular fuel) of these experiments or the various sets considered together.

The accuracy for predicting criticality of plutonium fueled systems is of the order of $\pm 3\%$ whereas it is perhaps $\pm 1\%$ for uranium reactors. In individual circumstances, such as for a given set of UO_2 - PuO_2 lattice experiments, the agreement between calculated and measured values of k_{eff} are better than the 3% value. However, when all of the lattice experiments performed using plutonium fuel are considered, the largest discrepancies between calculation and experiment occur for those plutonium systems which should be the least difficult to compute accurately, namely the Al-Pu fueled lattices. Thus, the somewhat better agreement obtained for the UO_2 - PuO_2 lattices may be the result of errors in the theoretical methods and/or cross sections which tend to compensate.

A theoretical correlation of most of the above described experiments, using the base calculational approach referred to earlier, shows a markedly good agreement for the multiplication of the mixed oxide systems but a poor agreement for the plutonium-only or uranium-only systems.⁽¹¹⁶⁾

An error analysis of the calculated k_{eff} using the various errors which are assigned to input parameters in the calculation result in the fact that one should still expect only approximately a 2% accuracy on the value of k_{eff} calculated for the mixed-oxide system. The accuracy which results is much better than this and is suspected to be due to compensating errors in the methods used for the uranium and plutonium calculations. A trend exists in the correlation results for the mixed oxide and uranium oxide lattices as a function of the volume ratio of moderator-to-fuel. Such a trend could be removed by a different treatment of the ^{238}U resonance calculation.

The discrepancy is largest for the 2 wt% plutonium-aluminum measurements which have the highest ^{240}Pu concentration in weight percent. However, the 5 wt% rods, because of their higher plutonium concentration, have about the same ^{240}Pu concentration present in the core. If the results from additional remeasurements of the Al-Pu fuel follow the same pattern of removing the trends in the correlations of the data as do the present remeasurements, it could very well indicate that the basic cross sections or the thermalization calculations are causing a discrepancy in these results. These remeasurements should be continued for the Al-Pu fuel and destructive reanalyses should be made of the contents of the fuel. It is very important that the cause of the discrepancy between

the calculational and experimental results be understood for the plutonium-only fuels. A fundamental understanding of the inherent accuracy of the methods and cross sections as they are applied to drawing conclusions about the adequacy of plutonium systems is a prerequisite to the calculational methods in their application to mixed oxide fueled systems.

The limits on the accuracy of predicting criticality must be reduced to the point where reliable engineering calculations can be performed. This accuracy is of the order of that for uranium systems, namely, $\pm 1\%$ in k_{eff} . Uncertainties in predicting k_{eff} which are larger than this represent too severe a limitation on the nuclear design of plutonium systems (i.e., core lifetime, transient behavior, etc.). By proper normalization techniques for a given type of reactor, such an accuracy is possible. However, normalization in this way does not increase the basic understanding of the physics of these systems and does not lend confidence to extrapolating or interpolating in the design of other systems for which there is no experimental information.

Measurements and analyses of uniform lattices provide a basis for assessment of methods and data but are not inclusive of all of the problems inherent in nuclear design. A good correlation of results for uniform lattice measurements in a critical facility does not guarantee a good agreement for critical loadings at the startup of a power reactor. The uniform lattice measurement and the EBWR experiment are a good example. Certain methods and cross sections gave good results for the uniform lattices using the EBWR fuel.^(49,117) However, use of the same methods and cross sections for the EBWR reactor itself grossly underpredicted the critical mass required.^(94,95)

The most rigorous methods of calculation do not predict "a priori" the criticality of plutonium fueled systems to desired accuracies. Normalizations of methods or cross sections or calculated results which are based upon clean uniform lattice analyses must be used in nuclear design. Studies should be continued to determine the limitations inherent in the theory, the cross-section data, and the experimental techniques such that the prediction of criticality in plutonium systems is placed upon a sound theoretical footing. To implement these studies, additional experimental data are required. These measurements should be designed to provide data which will test the accuracy of specific portions of the calculational methods which are suspected to be causing discrepancies (e.g., thermalization and leakage). For those systems in which critical mass and buckling measurements have been made, additional information (such as boron poisoned measurements and intercell activation measurements) is required. Additional critical experiments are required to cover the range of possible plutonium fuel compositions and concentrations to be utilized in plutonium recycle in thermal reactors. This additional data should also provide data specific to testing certain portions of the theory and/or cross-section data which are suspected to be causing discrepancies. Particular emphasis should be given to detailed measurements of the characteristics of non-uniform lattices, e.g., the effects due to control rods, control rod-gaps, power distributions, etc. Only with this additional information is it possible to definitely determine the limitations inherent in the theory, the cross section data, and the experimental techniques such that the uncertainty in predicting criticality of plutonium systems is reduced to that desired.

Plutonium-enriched systems exhibit negative temperature and void coefficients of reactivity which are generally larger than for uranium-enriched systems. The moderator temperature and void coefficients for uranium systems are in general characterized by the change in k_{eff} due to water density changes (i.e., H/U atom ratio) because the cross sections are all nearly $1/v$ and η^{235} is relatively flat in the thermal region. Thus, the comparison of calculated and measured values of k_{eff} for lattices which cover a range of H/U ratios give an indication of the ability of the methods to predict these coefficients. For plutonium systems, the ^{240}Pu resonance at 1.056 eV, the resonances at ~ 0.3 eV in ^{239}Pu and ^{241}Pu , and the shape of η^{239} in the thermal-energy range all influence the value for moderator temperature and void coefficients. Because of these influences, the coefficients cannot be expressed merely as a change resulting from a decrease in the moderator density (i.e., H/Pu change). Hence, the room temperature clean lattice criticals at various H/Pu ratios do not suffice as an adequate test of the methods for calculating these coefficients. In addition, the adequacy of the scattering kernel for H_2O plays a significant role in calculating these quantities. Valid checks with thermal-hydraulic measurements in a power environment are required before confidence in calculating the coefficients is possible.

A correlation of some of the temperature coefficient results obtained for clean lattice, H_2O -moderated criticals, using the same base calculational techniques, shows that a definite trend exists with fissile content independent of the enriching material. (104) Such a trend could very well

be indicating that the discrepancies are not a characteristic of plutonium alone. The value for the leakage of neutrons from these lattices is high and temperature dependent. Thus the analytical correlation of the reactivity coefficients measurements is perhaps mainly a test of calculating the leakage effect on the coefficients rather than the coefficients themselves.

The reactivity coefficients are important in reactor design because of their implication on the core lifetime, control requirements, and transient behavior of the power reactor. Thus, more experimental data are required, preferably in lattices which have less leakage than those studied so far. Likewise, measurements in an environment more typical of the power reactor or the power reactor itself are preferred. In fact, valid checks of thermal-hydraulic and reactivity coefficient measurements in a power environment are required before confidence can be obtained in knowing the actual coefficients and their implications.

The power distribution in reactor cores containing plutonium fuel is different from those fueled with uranium. Core non-uniformities more severely affect the local power distributions and the dependence of these distributions with exposure for the plutonium-fueled core. These effects must be considered in the fuel design of cores containing plutonium. However, there are few data available for checking the methods used for calculating power distributions in plutonium-fueled cores. Some data were obtained by Battelle-Northwest in the lattice studies and by Westinghouse in the Saxton and Empire State Atomic Development Associates, Inc. programs. Information which will also be useful for determining the relative changes

in local power distributions has been obtained from the Experimental Boiling Water Reactor as a function of burnup. The data obtained during the conduct of the Saxton irradiation experiment are also important. Additional power peaking and power measurements in non-uniform lattices such as obtained under the USAEC-CNEN program are required to put this portion of the technology on a firm basis.

Usually power distributions cannot be predicted to better than 20% without post-experiment normalizations. Similar to the situation for the void and temperature coefficients, valid checks of thermal-hydraulic measurements in a power environment are required before confidence in knowing the actual power distribution is possible.

Reactor Studies. Lattice measurements can and do supply much information necessary for assessing the accuracy of nuclear design calculational methods. However, irradiation experiments must be performed to obtain information on physics, metallurgical, and engineering characteristics as a function of exposure. Data obtained on the change in reactivity and isotopic composition of the fuel with exposure are required so that meaningful tests can be made of the theoretical methods and cross section data. All of the problem areas involved in predicting criticality, reactivity coefficients, power distributions, etc. are encompassed in the integral experiment labelled burnup. In reality, they are all portions of the core burnup problem.

Safe operation of a reactor system is required. Thus, it is necessary to adequately predict the dynamic behavior of the core so that any transient (long or short term) can be detected and controlled. The problem

is to determine the degree of accuracy required in the kinetics information so that the design is adequate from a safety standpoint. Experimental information is needed for the kinetic, transient and control characteristics for fuels which differ in plutonium composition and concentration. Detailed studies are required on xenon oscillations and on the coupling between thermal hydraulics and neutronics for plutonium-fueled systems to assess what limitations, if any, exist for large plutonium-fueled reactor designs.

It is established that the kinetic behavior of plutonium-fueled reactors is different from that of uranium-fueled reactors. The delayed neutron fraction for ^{239}Pu is less than that for ^{235}U . Calculated values of ratios of β/λ are within about 15% of experimental results and can usually be calculated with a fair degree of confidence. Reactors fueled with plutonium will probably have larger temperature and void coefficients of reactivity. These differences will affect the coupled nuclear and thermal hydraulic characteristics. The degree to which concentration and composition of plutonium affect these characteristics is presently not accurately known. The yield of iodine, the precursor of xenon, is less for ^{239}Pu fissions than for ^{235}U fissions and so less xenon will be present in a plutonium core. The spectrum is also harder in a plutonium-fueled system so the reactivity effect of xenon is expected to be less in a plutonium core. Consequently, the plutonium-fueled reactor may be more stable against xenon oscillations than the uranium-fueled system. The decrease in the worth of xenon between the UO_2 and $\text{UO}_2\text{-PuO}_2$ cores of the Saxton reactor ⁽¹¹¹⁾ corroborate this fact.

The differences in spectra between plutonium- and uranium-fueled reactors affect the control system since both the strength and the shutdown margin of a control system are reduced when plutonium is substituted in a reactor designed for uranium. As an example, the control-rod worth decreased in Saxton⁽¹¹¹⁾ for the plutonium loading relative to the uranium loading. To what extent the strength and shutdown margin are reduced is difficult to assess because of the lack of sufficient experimental information.

Measurements have been made of the worths of hafnium and boron rods in a few plutonium fueled lattices under the Plutonium Utilization Program. Measurements using Ag-In-Cd rods and Co rods were made as part of the Westinghouse program with the Empire State Atomic Development Associates, Inc. and the USAEC and as part of the Westinghouse Saxton program with the USAEC and Euratom. These data are useful in determining the accuracy in calculating the worth of these materials although no method has been demonstrated to reliably calculate rods in the arrangements which were measured. Further measurements in typical reactor geometries of actual control rod designs such as are done in the reactor application studies or in the power reactor are important.

The burnup characteristics of reactors fueled with plutonium are much more complex than for uranium reactors. Uranium-fueled reactors contain significant amounts of plutonium upon reaching goal exposure. The prediction of plutonium buildup and its composition as a function of exposure represents the biggest uncertainty in the design of these systems. By comparison, the irradiation behavior of a plutonium-fueled core is much

more difficult to calculate because plutonium is present initially, it is building in from capture in ^{238}U , and higher mass isotopes are being formed. These effects are all dependent on the amount of plutonium in the fuel initially, the composition of the plutonium, and the lattice, and/or position in the lattice in which the fuel resides.

Irradiations of plutonium fuel need to be continued to obtain data necessary for evaluating the nuclear design methods. Irradiations of partial cores of plutonium have been completed in the Experimental Boiling Water Reactor and are being made in the Saxton core. These data are important to the utilization of plutonium in power reactors in that they represent the only data available on the irradiation characteristics of plutonium-fueled H_2O reactors. The EBWR experiment suffers from the limited range of exposure and the small amount of data available. Both experiments are partial plutonium loadings; hence the observed behavior is the combined result of the plutonium zone behavior and the uranium zone behavior. Sufficient experimental data are not available to extract the behavior of the plutonium zone alone. Thus the data collected do not provide a stringent test of the adequacy of the methods as they apply to plutonium recycle. Nevertheless, the data are still very useful.

The plutonium batch core experiment being conducted in the PRTR is providing very important data on the changes in reactivity and isotopic composition as a function of exposure. These data are significant in that the experiment is controlled to yield the maximum amount of reactor physics information on the irradiation characteristics of plutonium fuel and that the observed reactor behavior is basically due to the central 55 elements.

Though the neutron spectrum in the PRTR is different from that of H_2O reactors, there are some correspondences in that the fuel temperatures are characteristic of those required for an H_2O power reactor. More importantly from a physics standpoint, the reactor system is relatively simple to analyze, and because of this the data are utilized in a more fundamental analysis of the effects of cross section and method uncertainties on prediction of core lifetime and isotopic transmutations.

Further data are needed on the operational characteristics of cores loaded with large fractions of plutonium fuel for purposes of methods evaluation. These data will allow comparisons to be made between cores loaded with uranium and plutonium fuels and as a result extrapolations can then be made, based upon uranium reactor technology, to other plutonium-fueled reactor situations.

Special sample irradiations have been performed in the EBWR, the VBWR and the PRTR to obtain data on the changes in fuel isotopic composition for plutonium-enriched fuels. These data are being used as bases for comparison with calculation in assessing the validity of current computational methods and cross section data. The use of the isotopic concentrations to deduce effective-cross-section ratios provides another key for understanding some of the inadequacies of the methods. Analytical correlations^(108,112) of ratios deduced for Al-Pu fuel irradiated in the PRTR have led to important conclusions about the methods and the basic cross sections. In these correlations, the ratios of effective cross sections are found to be very sensitive to variations in the magnitude and the shape of the thermal cross sections. Since the magnitude

of the thermal cross sections at 2200 m/sec depends on the shape of the cross section in the thermal energy region, care must be exercised in selecting a set of cross sections for reactor design calculations. It has been recommended that future evaluations of 2200 m/sec constants give consideration to cross section shapes and that cross section data include information throughout the thermal-energy range and not just at a single point.

More data are required on the irradiation characteristics of plutonium fuels at high exposures. Current uranium fuel designs have goal exposures of around 30,000 to 35,000 MWd/MTM and none of the data currently available for plutonium fuels are near the exposure level.

Plutonium fuels with varying spatial enrichment can be designed with a smaller cost penalty than can uranium fuels. Because of this, it appears fruitful to optimize the power distribution in cores fueled with plutonium. Studies are required to exploit the advantage to be gained by this situation and experiments will be required to verify the concept.

REFERENCES

1. F. G. Dawson, "Program Analysis and Plans, Plutonium Utilization Program," BNWL-298, Pacific Northwest Laboratory, July 1966.
2. F. G. Dawson, "Plutonium Utilization Program Annual Report, Fiscal Year 1967," BNWL-624, Pacific Northwest Laboratory, January 1968.
3. W. L. Orr, et al., "Saxton Plutonium Program, Nuclear Design of the Saxton Partial Plutonium Core," WCAP-3385-51, Westinghouse Electric Corporation, December 1965.
4. J. G. Carver, "Experimental Plan for the Plutonium Subcritical Program," GEAP-4690, General Electric Company, August 1965.
5. M. A. Robkin, "Program for Development of Plutonium Recycle for Use in Light Water Moderated Reactors - Final Report," GEAP-4907, General Electric Company, August 1965.
6. W. L. Orr, F. L. Langford and R. D. Leamer, "Physics Development for Plutonium Utilization in Pressurized Water Reactors," Joint International Conference on the Physics Problems in Thermal Reactor Design, 27-29 June 1967 at the Institution of Civil Engineers, London, SW1.
7. M. C. Beekman and H. A. Wagner, "EEI Program on Plutonium Fuels," Trans. Am. Nucl. Soc. 9, 539, (1966).
8. M. D'hont, E. Vanden Bemden, J. Van der Spek, "Description of the Belgo-nucleaire-CEN Plutonium Project," Proceedings: Plutonium as a Power Reactor Fuel, ANS Topical Meeting, December 1962, HW-75007.
9. C. R. Richey, et al., "Criticality of Homogeneous Plutonium Oxide - Plastic Compacts at H:Pu = 15," Nucl. Sci. and Eng., 23, 150 (1965) and R. C. Lloyd, et al., "Criticality Studies with Plutonium Solutions," Nucl. Sci. and Eng., 25, 165 (1966), and C. R. Richey, "Theoretical Analysis of Homogeneous Plutonium Critical Experiments," Nucl. Sci. and Eng., 31, 32 (1968).
10. D. D. Lanning and G. J. Busselman, "Phoenix Fuel Program Progress Report," BNWL-635, Pacific Northwest Laboratory, November 1967.
11. J. R. Smith and S. D. Reeder, Second Conference on Neutron Cross Sections and Technology, Paper D-14, Washington, D. C. (1968) and J. R. Smith, S. D. Reeder and R. G. Fluharty, IDO-17083 (February 1966).
12. F. G. Dawson and L. H. Steves, "Comparison of Westcott Spectrum with the Gas and Water Kernel Models: Determination of the Neutron Temperature and Epi-Thermal Index," HW-75278, October 1962.
13. H. Bohl, Jr., E. M. Gelbard and G. M. Ryan, "MUFT-4 Fast Neutron Spectrum Code for the IBM-704," WAPD-TM-72 (1957) and H. Bohl, Jr., A. P. Hemphill, MUFT-5: A Fast Neutron Spectrum Program for the Philco-2000," WAPD-TM-218, (February 1961).
14. G. D. Joanou and J. S. Dudek, "GAM-I: A consistent P2 Multigroup Code for the Evaluation of Fast Neutron Spectra and Multigroup Constants," GA-1850, June 28, 1961 and G. D. Joanou and J. S. Dudek, "GAM-II: A B₃ Code for the Calculation of Fast Neutron Spectra and Associated Multigroup Constants," GA-4265, (Sept. 1963).

15. J. L. Carter, Jr., "Computer Code Abstracts, Computer Code-HRG," Reactor Physics Department Technical Activities Quarterly Report - July, August, September, 1966, BNWL-340, Pacific Northwest Laboratory, October 15, 1966.
16. H. C. Honeck, "THERMOS, A Thermalization Transport Theory Code for Reactor Lattice Calculations," BNL-5826, Brookhaven National Laboratory, September 1961.
17. G. C. Poncelet, "LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS," WCAP-6073, Westinghouse Electric Corporation, April 1966.
18. J. E. Suich and H. C. Honeck, "The HAMMER System," DP-1064, Savannah River Laboratory, January 1967.
19. D. R. Skeen and W. L. Purcell, HTH - Computer Overlay of Programs HRG-THERMOS/BATTELLE-HFN," Reactor Physics Department Technical Activities Quarterly Report, July, August, September, 1967, BNWL-634, Pacific Northwest Laboratory, February 1968.
20. J. R. Lilley, "Computer Code HFN-Multigroup, Multiregion Neutron Diffusion Theory in One Space Dimension," HW-71545, Hanford Laboratories, November 1961.
21. D. H. Thomsen, unpublished, Pacific Northwest Laboratory, 1968.
22. K. B. Stewart, "BNW Master Library," BNWL-CC-325, Pacific Northwest Laboratory, September, 1965.
23. H. C. Honeck, "ENDF/B Specifications for an Evaluated Nuclear Data File for Reactor Applications," BNL-50066(T-467), Brookhaven National Laboratory, May 1966, Revised July 1967.
24. R. Sher and J. Felberbaum, "Least Squares Analysis of the 2200 m/sec Parameters of U^{233} , U^{235} , and Pu^{239} ," "USAEC Report, BNL-722, Brookhaven National Laboratory, June 1962.
25. R. Sher and J. Felberbaum, "Least Squares Analysis of the 2200 m/sec Parameters of U^{233} , U^{235} , and Pu^{239} ," "USAEC Report, BNL-918, Brookhaven National Laboratory, March 1965.
26. C. H. Westcott, et al., "A Survey of Values of the 2200 m/s constants for Four Fissile Nuclides," Atomic Energy Review, 3, p. 2, (1965).
27. C. H. Westcott, "Nuclear Data for Reactors," Vol. II, p. 44, Paris (1966).
28. B. R. Leonard, Jr., "Survey of the Status of Low Energy Cross Sections of Fissile Nuclides," Neutron Physics, Proceedings of Symposium held at Rensselaer Polytechnic Institute, 1961, pp. 3-33, edited by M. L. Yeater, Academic Press, New York, N. Y. 1962, also HW-69342, April 22, 1961.
29. B. R. Leonard, Jr., "Plutonium Handbook, A Guide to the Technology," Vol. 1, 1967.
30. C. H. Westcott, "Effective Cross Section Values for Well-Moderated Thermal Reactor Spectra," AECL-1101, Nov. 1, 1960.
31. R. C. Liikala, unpublished, Pacific Northwest Laboratory.

32. B. R. Leonard, Jr., R. C. Liikala, and U. P. Jenquin, unpublished, 1968, Pacific Northwest Laboratory.
33. E. Critoph, "Effective Cross Sections for U-235 and Pu-239," CRRP-1191, Chalk River Nuclear Laboratories, 1964.
34. B. R. Leonard, Jr., "A Review of the Neutron Cross Sections of Pu²³⁹ Below 1 eV," TNCC(US)-58, August 1959.
35. G. J. Safford and W. W. Havens, Jr., "A Precision Measurement of the Total Cross Section of Pu²³⁹ Between 0.00291 and 0.1 eV," Nucl. Sci. & Eng., 11, pp. 65-68 (1961).
36. U. P. Jenquin, V. O. Uotinen and C. M. Heeb, "Sensitivity of Reactivity Characteristics of Cross-Section Uncertainties for Plutonium-Fueled Thermal Systems," To be published in the Proceedings of the Second Conference on Neutron Cross Section Technology, March 4-7, 1968.
37. J. J. Schmidt, "Neutron Cross Sections for Fast Reactor Materials, Part 1: Evaluation," KFK-120 (EANDC-E-35U), February 1966.
38. W. A. Reardon and D. E. Christensen, "Atom Ratios and Effective Cross-Section Ratios in Highly Depleted Plutonium-Aluminum Alloy Fuel," Nucl. Sci. and Eng., 30, 222-232 (1967).
39. R. W. Durham, G. C. Hanna, M. Lounsbury, C. B. Bigham, R. G. Hart and R. W. Jones, Nuclear Data for Reactors, II, 17, (1966).
40. B. R. Leonard, Jr. and S. J. Friesenhahn, "Pu²⁴¹ Fission Cross Section," Nuclear Physics Research Quarterly Report July, August, September 1959, HW-62727, p. 19, Hanford Laboratories, October 1959.
41. H. D. Lemmel and C. H. Westcott, "Fission and Absorption g-Factors of ²⁴¹Pu," Journ. of Nucl. Energy, 21, 417-424 (1967).
42. M. Asghar, M. C. Moxon and N. J. Pattenden, "Neutron Resonance Parameters of ²⁴⁰Pu," Nuclear Data for Reactors, Vol. II, 145 (1966).
43. H. Weigman, J. Winter and H. Schmid, "Neutron Capture Measurements in the Resonance Region," to be published in the Proceedings of the Second Conference on Neutron Cross Section Technology, paper D-9, Washington, D. C. (Mar. 1968).
44. J. R. Stehn, M. D. Goldberg, R. Wiener - Chasman, S. F. Mughabghab, B. A. Magurno, and V. M. May, Neutron Cross Section, Vol. III, BNL-325 (1965).
45. Butler, Lounsbury and Merritt, Can. J. Phys., 35, 147 (1957) re-evaluated by J. S. Story, private communication (1966).
46. P. R. Fields, et al., "Pile Neutron Cross Sections of the Heavier Plutonium Isotopes," Nucl. Sci. Eng., 1, 62 (1956).
47. T. E. Young, unpublished (1967).
48. W. P. Stinson, "Subcritical Measurements with UO₂-2 wt% PuO₂ Fuel," Plutonium Utilization Program Annual Report Fiscal Year 1967, BNWL-624, Pacific Northwest Laboratory, January 1968 and R. C. Liikala and W. P. Stinson, "Experimental and Analytical Results for PuO₂-UO₂-H₂O Lattices," Trans. Am. Nucl. Soc., 9, 127 (1966).

49. W. P. Stinson, "A Fitting Formula for Critical Mass Data: H_2O Moderated Cylindrical Arrays of Rods," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, Pacific Northwest Laboratory, October 1967 and L. C. Schmid, et al., "Critical Masses and Bucklings of $PuO_2-UO_2-H_2O$ Systems," Trans. Am. Nucl. Soc., 7, 216 (1964).
50. V. O. Uotinen and R. C. Liikala, "Reactivity Worth of Neutron Absorbing Rods in Plutonium-Fueled, H_2O Moderated Systems," Trans. Am. Nucl. Soc., 9, 180 (1966).
51. V. O. Uotinen, "Measurement of the Reactivity Worth of Hafnium Oxide Rods in a $PuO_2-UO_2-H_2O$ Lattice," Trans. Am. Nucl. Soc., 9, 520 (1966).
52. J. H. Lauby, "Summary of Water Moderated Experiments with 1.8 wt% Pu-Al Fuel in the PRCF," Physics Research Quarterly Report, April, May, June, 1965, BNWL-149, Pacific Northwest Laboratory, July 1965.
53. V. O. Uotinen, "Clean Critical Loading of Al-2 wt% Pu Fuel in an H_2O Moderated Core," Plutonium Utilization Program Annual Report, Fiscal Year 1967, BNWL-624, Pacific Northwest Laboratory, January 1968.
54. V. O. Uotinen, A. L. Pitner and W. P. Stinson, "Migration Area and Moderator Reactivity Coefficients for the H_2O Moderated 1.8 wt% Pu-Al Loading in the PRCF," USAEC Report BNWL-160, October, 1965.
55. L. C. Schmid, et al., "Summary of Results of EBWR Critical Experiments," Physics Research Quarterly Report, October, November, December, 1964, HW-84608, Hanford Laboratories, January 1965.
56. V. O. Uotinen, "Migration Area and Moderator Reactivity Coefficients Measured During the EBWR Critical Experiments," Physics Research Quarterly Report, October-December, 1964, HW-84608, January 15, 1965.
57. V. O. Uotinen and L. D. Williams, "Experiments and Calculations for H_2O -Moderated Assemblies Containing UO_2 -2 wt% PuO_2 Fuel Rods," Trans. Am. Nucl. Soc., 10, 186 (1967) and BNWL-SA-1107 (1967).
58. V. O. Uotinen and L. D. Williams, "Critical Experiments in $PuO_2-UO_2-H_2O$ Lattices," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1966, BNWL-400, April 1967.
59. P. Loizzo, R. Martinelli, and L. D. Williams, "USAEC-CNEN Joint Program in the PRCF," Plutonium Utilization Program Technical Activities Quarterly Report, December 1967, January, February 1968, BNWL-739, Pacific Northwest Laboratory, April 1968.
60. N. Pacilio, "Statistical and Pulsed Experiments in Organic Moderated and Reflected Assemblies," Energia Nucleare, 15, 129 (February 1968).
61. P. Loizzo, "RIBOT: A Physical Model for Light-Water Lattice Calculations," BNWL-735, Pacific Northwest Laboratory, February 1968.
62. R. D. Leamer, et al., " PuO_2-UO_2 Fueled Critical Experiments," WCAP-3726-1, Westinghouse Electric Corporation, July 1967.

63. E. G. Taylor, "Saxton Plutonium Program, Critical Experiments for the Saxton Partial Plutonium Core," WCAP-3385-54, Westinghouse Electric Corporation, December 1965.
64. J. G. Carver, "Plutonium Subcritical Experiment Program - Final Report," GEAP-5357, General Electric Company, December 1967.
65. L. C. Schmid, "Survey of Work at Hanford Laboratories in D₂O Moderated Systems," HW-SA-3456, Hanford Laboratories, April 1964.
66. J. L. Crandall, et al., "Lattice Studies and Critical Experiments in D₂O Moderated Systems," 3rd U.N. Conference on the Peaceful Uses of Atomic Energy, Vol. 3, p. 126 (1964).
67. U. P. Jenquin, unpublished, Battelle-Northwest and EACRP Compilation of D₂O Lattice Data, EACRP-L-42.
68. V. O. Uotinen, "Lutetium Activations and Spectral Indices in a D₂O Moderated Reactor with Pu-Al 19-Rod Clusters," Physics Research Quarterly Report, January, February, March, 1966, BNWL-284, Pacific Northwest Laboratory, April 1966.
69. R. A. Bennett and L. C. Schmid, "Approach to Critical and Calibration Experiments in the PRCF," HW-80206, Hanford Laboratories (1963).
70. V. O. Uotinen and L. C. Schmid, "Reactivity Measurements of UO₂ and Pu-Al Fuel Elements," HW-81435, Hanford Laboratories, (1964).
71. V. O. Uotinen and L. C. Schmid, "Reactivity Measurements of Mixed Oxide Fuel Elements," HW-79960, Hanford Laboratories (1963).
72. V. O. Uotinen and L. C. Schmid, "The D₂O Moderator Level Coefficient of the PRCF," HW-81397, Hanford Laboratories (1964).
73. W. P. Stinson, unpublished, Pacific Northwest Laboratory.
74. B. Arcipiani and L. C. Schmid, "Neutron Spectrum Measurements in the Plutonium Recycle Critical Facility," Physics Research Quarterly Report, July, August, September, 1963, HW-79054, Hanford Laboratories, October 1963.
75. H. Sakata and L. C. Schmid, "Safety Rod Interaction in the PRCF," HW-81378, Hanford Laboratories (1964).
76. V. O. Uotinen, "A Poison-Free Loading of UO₂ and Pu-Al in the PRCF with D₂O Moderator," BNWL-372, Pacific Northwest Laboratory, (1967).
77. J. W. Kutcher, et al., "Critical Experiments with PuO₂-UO₂ fuel and D₂O Moderator," Trans. Am. Nucl. Soc., 9, 448 (1967).
78. J. R. Worden, W. L. Purcell, and L. C. Schmid, "Physics Experiment - High Power Density Core of the PRTR," BNWL-221, Pacific Northwest Laboratory, (1966).
79. N. P. Baumann, et al., "Lattice Experiments with Simulated Burned-Up Fuel for D₂O Power Reactors," DP-1122, Savannah River Laboratory (February 1968).

80. R. E. Heineman, F. C. Engesser, J. R. Lilley, and D. J. Donahue, "Measurements of k_{∞} for D_2O Moderated Clusters of UO_2 and Pu-Al Rods," Nuclear Physics Research Quarterly Report, July, August, September, 1957, HW-53492, Hanford Laboratories, (1957).
81. F. C. Engesser and J. R. Lilley, "Plutonium Spike Enrichment," Nuclear Physics Research Quarterly Report, January, February, March, 1958, HW-44879, Hanford Laboratories, (1958).
82. R. I. Smith, "Experimental Studies in Low-Exposure Pu-Al Lattices," Physics Research Quarterly Report, January, February, March, 1962, HW-73116, Hanford Laboratories, (1957).
83. R. I. Smith, "The Fuel Temperature Coefficient of k_{∞} for Plutonium-Aluminum Fuel," Physics Research Quarterly Report, January, February, March, 1962, HW-73116, Hanford Laboratories, (1962).
84. T. B. Thornbury, and L. C. Davenport, "Experimental Studies in High-Exposure Pu-Al Lattices," Physics Research Quarterly Report, April, May, June, 1963, HW-77871, Hanford Laboratories (1963).
85. P. F. Nichols, "Effective Resonance Integral for ^{240}Pu in Plutonium-Aluminum Alloy Rods," Physics Research Quarterly Report, July, August, September, 1962, HW-75228, Hanford Laboratories (1962).
86. T. B. Thornbury and N. A. Hill, "Determination of k_{∞} for a PuO_2 - UO_2 -Graphite Lattices," Trans. Am. Nucl. Soc., 8, pp. 268 (June 1965).
87. N. A. Hill, et al., "Comparison of PCTR Measurements Using Unpoisoned and Poisoned Techniques on PuO_2 - UO_2 Graphite Lattices," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, Pacific Northwest Laboratory (1967).
88. D. F. Newman, D. R. Oden, W. P. Walsh and A. D. Vaughn, "Water Moderated Test Lattices in PCTR," Plutonium Utilization Program Technical Activities Quarterly Report, September, October, November, 1967, BNWL-654, Pacific Northwest Laboratory (1968).
89. D. F. Newman and D. R. Oden, "PCTR Measurement of k_{∞} for UO_2 -0.86 wt% PuO_2 Rods in 1.0 in. Square Pitch, Water Moderated Lattice," Plutonium Utilization Program Technical Activities Quarterly Report, December 1967, January, February, 1968, BNWL-739, Pacific Northwest Laboratory (April 1963).
90. F. R. Czerniejewski, "PuO₂ Grain Reactivity Effects," BNWL-427, Pacific Northwest Laboratory, June 1967.
91. P. Loizzo, BNW Memo to L. C. Schmid, May 26, 1967.
92. W. L. Orr, private communication, June 25, 1968.
93. W. P. Stinson and J. H. Lauby, "Approach-to-Critical Experiments with UO_2 -4 wt% PuO_2 -H₂O Lattices," Plutonium Utilization Program Technical Activities Quarterly Report, December 1967, January, February 1968, BNWL-739, Pacific Northwest Laboratory, April 1968.

94. R. C. Liikala, U. P. Jenquin and W. A. Reardon, "The $\text{PuO}_2\text{-UO}_2$ Experiment in the EBWR," BNWL-126, Pacific Northwest Laboratory, June 1965.
95. B. J. Toppel and R. Avery, "Some Physics Aspects of the Plutonium Experiment in EBWR," ANL-7019, Argonne National Laboratory, April 1965.
96. F. G. Dawson, et al., "The EBWR Plutonium Demonstration Experiment," Trans. Am. Nucl. Soc., 10, 500 (1967).
97. B. J. Toppel, et al., "Safety Analysis Associated with the Plutonium Recycle Experiment in EBWR," ANL-6841, Argonne National Laboratory, November, 1964.
98. P. H. Kier, "Analysis of the Initial Critical Experiments of the EBWR Plutonium Recycle Program," ANL-7368, Argonne National Laboratory, August 1967.
99. D. E. Christensen and G. Manca, "Nondestructive Analysis of Irradiated Fuel," Plutonium Utilization Program Technical Activities Quarterly Report, December, 1967, January, February 1968, BNWL-739, Pacific Northwest Laboratory, April 1968.
100. D. E. Christensen and E. S. Murphy, "Determination of Relative Burnup by Gamma Scanning EBWR Fuel Rods," BNWL-653, Pacific Northwest Laboratory, April 1968.
101. R. H. Chastian and H. I. Sternberg, "Preheated and Observed Performance of the Saxton Partial Plutonium Core During Startup Tests," Trans. Am. Nucl. Soc., 9, 280 (1966).
102. V. O. Uotinen, Unpublished, Pacific Northwest Laboratory.
103. R. I. Smith, J. W. Kutcher and J. H. Lauby, "Critical Experiments with the $\text{UO}_2\text{-2 wt% PuO}_2$ Core in the PRTR," Trans. Am. Nucl. Soc., 10, 185 (1967).
104. J. W. Kutcher, R. E. Harris and R. I. Smith, "Power Tests with the $\text{UO}_2\text{-2 wt% PuO}_2$ Core in the PRTR," Trans. Am. Nucl. Soc., 10, 517 (1967).
105. N. G. Wittenbrock, P. C. Walkup, and J. K. Anderson, "Plutonium Recycle Test Reactor, Final Safeguards Analysis," HW-61236, Hanford Laboratories (1959).
106. J. W. Kutcher and R. I. Smith, "PRTR Batch Core Experiment," Plutonium Utilization Program Technical Activities Quarterly Report, December, 1967, January, February 1968, BNWL-739, Pacific Northwest Laboratory, April 1968.
107. D. E. Christensen, E. B. Reppond and W. A. Reardon, "Burnup Data from Low-Exposure Plutonium-Aluminum Fuel," BNWL-CC-904, November, 1966; and "Burnup Data from High-Exposure Plutonium-Aluminum Fuel," BNWL-CC-905, December, 1966.
108. L. C. Schmid, et al., "Experimental and Theoretical Methods used for Burnup Analysis at Battelle-Northwest," Transactions British Nuclear Energy Society International Conference on Physics Problems of Thermal Reactor Design, London England, June 27-29, 1968.

109. B. H. Duane, "Maximum Likelihood Nonlinear Correlated Fields (BNW Program LIKELY)," BNWL-390, Pacific Northwest Laboratory, September, 1967.
110. R. P. Matsen, "A Technique for the Determination of Ratios of Effective Cross Sections from Reactor Fuel Burnup Data," Nucl. Sci. and Eng., 32, 267 (1968).
111. W. H. Layman, C. R. Montgomery and H. J. Williams, "Utilization of a Zoned Loading of Plutonium Fuel in the Saxton Reactor," Trans. Am. Nucl. Soc., 10, 498 (1967), and W. H. Layman and C. R. Montgomery, unpublished data, Saxton Nuclear Experimental Corporation, November 1967.
112. D. E. Christensen, R. C. Liikala, R. P. Matsen and D. L. Prezbindowski, "Effects of Uncertainties in Nuclear Data on Experimental and Calculated Reactor Burnup," Proceedings Second Conference on Neutron Cross Section Technology, Washington, D.C., March 4-7, 1968.
113. W. P. Stinson and L. C. Schmid, "Subcritical Measurements with 1.8 wt% Pu-Al Rods in Light Water," Physics Research Quarterly Report, April-June, 1962, HW-74190, pp. 17-24, July 16, 1962.
114. W. P. Stinson, J. A. Marks and L. C. Schmid, "Subcritical Measurements with 2.0 wt% Pu-Al Rods in Light Water," Physics Research Quarterly Report, July-September, 1963, HW-79054, pp. 43-46, Hanford Laboratories, October 15, 1963.
115. V. I. Neeley, R. C. Lloyd and E. D. Clayton, "Neutron Multiplication Measurements with Pu-Al Alloy Rods in Light Water," HW-70944, Hanford Laboratories, August 29, 1961.
116. F. G. Dawson, et al., "Analysis of Plutonium Fueled Light Water Reactors," Proceedings Plutonium as a Reactor Fuel, IAEA Symposium, Brussels, Belgium, March 13-17, 1967, and R. C. Liikala and W. L. Purcell, "Comparison of Calculated and Measured Multiplications, k_{eff} , for Some Uranium and/or Plutonium Fueled H_2O Lattices," Plutonium Utilization Program Annual Report, Fiscal Year 1967, BNWL-624, Pacific Northwest Laboratory, January 1968.
117. D. P. Moon, "Consistent Heavy-Atom Cross-Section Evaluation for Plutonium Recycle Calculations," ANL-6972, Argonne National Laboratory, December 1964.

BIBLIOGRAPHYPACIFIC NORTHWEST LABORATORY

F. C. Engesser, "The Metal Temperature Coefficient 19-Rod Clusters of Uranium Oxide with Air Coolant," Nuclear Physics Quarterly Report, April-June, 1958, HW-56919.

The Staff of Hanford Laboratories Operation, Plutonium Recycle Test Reactor Critical Tests, HW-61900-B. July 1, 1960.

L. C. Schmid and W. P. Stinson, Lutetium as a Spectral Index Detector, HW-66319 (1960).

L. C. Schmid and W. P. Stinson, "Lutetium as a Spectral Index Detector, Part II, Resonance Integrals," Physics Research Quarterly Report, January-March, 1961, HW-69475 (1961).

J. R. Triplett, R. E. Dunn, V. W. Gustafson, R. E. Peterson, J. J. Regimbal, J. T. Russell, and L. C. Schmid, Plutonium Recycle Test Reactor Critical Test Results, HW-61900 BA (1961).

The Staff of Hanford Laboratories Operation, Plutonium Recycle Test Reactor Power Tests, HW-61900-C. April 7, 1961

J. R. Triplett, L. C. Schmid, R. E. Peterson, R. E. Dunn, and J. J. Regimbal, "Plutonium Recycle Reactor Critical Tests," Trans. Am. Nuc. Soc., 4, No. 1, (1961).

M. D. Freshley and L. C. Schmid, The Reactivity of High Exposure Plutonium, HW-69169 (1961).

J. R. Triplett and R. E. Peterson, "Measured Physics Parameters, Design Features and Operating Characteristics of the PRTR," IAEA Symposium Proceedings Power Reactor Experiments, Vol. II, p. 213 (1962).

R. E. Peterson and V. W. Gustafson, "Nuclear Control Characteristics of the Plutonium Recycle Test Reactor," Trans. Am. Nuc. Soc., 5, No. 1, p. 173, June 1962.

R. I. Smith and R. E. Peterson, "Experience with the Plutonium Recycle Test Reactor," Proceedings Plutonium as a Power Reactor Fuel, HW-75007, (1962).

J. J. Regimbal, "Plutonium Aluminum Alloy Rods in H₂O: Comparison of Eigenvalue and Perturbation Analysis and Experiment" Trans. Am. Nucl. Soc., 5, 2, 369 (1962).

R. I. Smith, "An Investigation of the Effective Neutron Temperatures in Pu-Al Lattices," Physics Research Quarterly Report, Jan-March, 1962, HW-73116, April 16, 1962.

T. B. Thornbury, "Measurements of Boron and Plutonium Masses Using the PCTR," Physics Research Quarterly Report, July-September, 1962, HW-75228, October 15, 1962.

C. W. Lindenmeier, "Comparison of Nelkin's Water Kernel with Experiment," Physics Research Quarterly Report, July-September, 1962, HW-75228, pp. 4-9, October 15, 1962.

W. B. Henderson and F. G. Dawson, "Variation of Plutonium Isotopic Composition with Burnup in a 19-Rod Aluminum-Plutonium Fuel Element," Trans. Am. Nucl. Soc., 6, No. 1, pp. 163-164 (1963).

D. E. Christensen, et al., "Burnup Data from Three Graded Exposure PRTR Pu-Al Fuel Elements," Trans. Am. Nucl. Soc., 6, No. 2, pp. 268-269 (1963).

R. I. Smith, Transient Fission Product Poisoning in PRTR, HW-77638 (1963).

P. L. Hofmann, L. C. Schmid, and F. G. Dawson, Comments on the Physics of Phoenix Fuel Reactors, HW-77394 (1963).

L. C. Davenport and P. F. Nichols, "P-3 Calculations of Thermal Flux Distributions in Lattices Fueled with Plutonium," Physics Research Quarterly Report, October-December, 1962, HW-76128, January 15, 1963.

P. F. Nichols, "Changes in Resonance Absorption in Pu^{240} with Temperature for Plutonium Aluminum Fuel," Physics Research Quarterly Report, October-December, 1962, HW-76128, January 15, 1963.

W. P. Stinson and L. C. Schmid, "Two Zone Subcritical Measurements with 1.8 wt% Pu-Al Fuel Rods in Light Water," Physics Research Quarterly Report, October-December, 1962, January 15, 1963.

J. R. Worden, "An Analysis of Pu-Al Fueled Graphite-Moderated Lattices," Physics Research Quarterly Report, October-December, 1962, HW-76128, January 15, 1963.

C. W. Lindenmeier, "Scattering Law for Water I," Physics Research Quarterly Report, January-March, 1963, HW-77311, April 15, 1963.

B. Arcipiani and L. C. Schmid, "Diffusion Theory and Perturbation Calculations for Reactivity Measurements in the PRCF," Physics Research Quarterly Report, April-June, 1963, HW-77871, July 15, 1963.

T. B. Thornbury, "A Measurement in the PCTR of the Relative Plutonium Content of Pu-Al Fuel Elements," Physics Research Quarterly Report, October-December, 1962, HW-76128, January 15, 1963.

L. C. Schmid, "Delayed Neutron Parameters and Reactivity Measurements in the PRCF" Physics Research Quarterly Report, July-September, 1963, HW-79054.

C. W. Lindenmeier, "Scattering Law for Water II," Physics Research Quarterly Report, July-September, 1963, HW-79054, pp. 12-20, October 15, 1963.

- K. B. Stewart, "The Modified Heavy Gas Equation and Light Water-Moderated Systems," Physics Research Quarterly Report, October-December, 1963, HW-80020.
- V. W. Gustafson, Plutonium Recycle Test Reactor Power Test Results, HW-80253 (1964).
- D. E. Christensen, N. W. Brown, and W. A. Reardon, "Plutonium Burnup Data," Trans. Am. Nucl. Soc., 7, No. 2, pp 404-405 (1964).
- W. A. Reardon and R. L. Watts, "The Relative Effect of Physics Parameter Uncertainties on Core Performance and Computed Fuel-Cycle Costs from Perturbation Methods," Trans. Am. Nucl. Soc. 7, 1, 198 (1964).
- P. L. Hofmann, G. J. Busselman, and R. H. Holeman, "Compact Water-Moderated Plutonium Burners," Trans. Am. Nucl. Soc., 7, 1, 194 (1964).
- V. W. Gustafson and R. E. Peterson, "A Computer Investigation of Doppler Delay in Mixed Oxide Fuels," Trans. Am. Nucl. Soc., 7, 2, 275 (1964).
- J. J. Regimbal, "MTR Plutonium Loading Analysis," Trans. Am. Nucl. Soc., 7, 2, 299 (1964).
- R. C. Liikala, J. R. Worden and W. A. Reardon, "An Analysis of Uranium Fueled-Light Water Moderated Critical Experiments," Physics Research Quarterly Report, January-March, 1964, HW-81659, April, 1964.
- R. C. Liikala, "Comparison of Thermalization Models," Physics Research Quarterly Report, January-March, 1964, HW-81659, April 15, 1964.
- J. R. Worden, R. C. Liikala, and W. A. Reardon, "An Analysis of Plutonium Light Water Critical Experiments," Physics Research Quarterly Report, January-March, 1964, HW-81639, April 1964.
- F. G. Dawson, "Physics Aspects of Plutonium Fuels," Proceedings Commercial Fabrication of Plutonium Fuels, Hanford Laboratories Invitational Meeting, Richland, Washington, April 2-3, 1964. USAEC Report HW-82400.
- R. E. Peterson, Neutron Kinetics Problems Associated with Mixed Oxide Fuels, USAEC Report, HW-81259, July 1964.
- R. C. Liikala, "A Comparative Study of Unit Cell Homogenization for PuO_2 - UO_2 - H_2O Lattices," Physics Research Quarterly Report, July-September, 1964, USAEC Report HW-84369, General Electric Company, Richland, Washington October 1964.
- L. C. Schmid, "Measurement of Spectral Indices in the Plutonium Recycle Test Reactor," Physics Research Quarterly Report, July-September, 1964, HW-84369, October, 1964.
- B. Arcipiani and L. C. Schmid, "Measurement of the Absorption and Fission Sensitivities in the D_2O Moderated Plutonium Recycle Critical Facility," Physics Research Quarterly Report, January-March, 1964, HW-81659, April 15, 1964.

- D. E. Christensen, "Application of Burnup Analysis to Physics Studies at Battelle-Northwest," Trans. Am. Nucl. Soc., Vol 8, No. 1, p. 8 (1965).
- D. G. Albertson and G. J. Busselman, "Analysis and Measurement of Pu-Al-Polyethylene Systems in the PCTR," Trans. Am. Nucl. Soc., 8, 2, 448 (1965).
- J. J. Regimbal, "Analysis of MTR Cycle 108 - Low Exposure Plutonium Cores," Physics Research Quarterly Report, October-December, 1964, HW-86408, January 15, 1965.
- R. A. Harris and W. A. Reardon, "Comparison of Measured and Calculated Spectra and Average Reaction Rates," Physics Research Quarterly Report, January-March, 1965, BNWL-95, 1965.
- C. M. Heeb, Theory-Experiment Comparison of Pu²⁴⁰ Resonance Absorption, AEC Report BNWL-97, Battelle-Northwest, May 1965.
- C. M. Heeb, "Comparison of Theoretical and Experimental Resonance Absorption in Pu²⁴⁰," Physics Research Quarterly Report, April-June, 1965, BNWL-149, July 15, 1965.
- G. J. Busselman, "Preliminary Analysis of Some Pu-Al Polyethylene Experiments in the PCTR," Physics Research Quarterly Report, April-June, 1965, BNWL-149, July 15, 1965.
- D. G. Albertson, "Pu-Al Polyethylene Experiments in the PCTR," Reactor Physics Quarterly Report, April-June, 1965, BNWL-149, July 15, 1965.
- F. G. Dawson, D. E. Deonigi, And E. A. Eschbach, "Plutonium Buildup and Depletion," Nucleonics, August 1965.
- J. R. Worden, W. L. Purcell and R. C. Liikala, "Modifications to the Computer Code, THERMOS and Comparative Studies on Scattering Kernels," Reactor Physics Quarterly Report, July, August, September, 1965, BNWL-193, October 1965.
- J. R. Worden, W. L. Purcell and R. C. Liikala, "Appendix A - A Correspondence Between Calculated and Measured Thermal Utilization," Reactor Physics Quarterly Report, July, August, September, 1965, BNWL-193, 1965.
- R. P. Matsen, "Integrated Burnup Equations for Uranium-Plutonium Systems;" Physics Research Quarterly Report, October-December, 1965, BNWL-222, January 1966.
- F. G. Dawson, "Physics Design Aspects of Plutonium Recycle," Presented at the Commercial Plutonium Fuels Conference, Washington, D.C., March 1-2, 1966 Conf-660308.
- R. C. Liikala, W. L. Purcell and J. R. Worden, "Sensitivity of Reactor Multiplication Values to Cross-Section Uncertainties for Thermal Systems," Proceedings of the Conference on Neutron Cross-Section Technology, March 22-24, 1966. Conf-660303

- J. R. Worden, W. L. Purcell and R. C. Liikala, "Sensitivity of Thermal Reactor Parameters to Scattering Model," Proceedings of the Conference on Neutron Cross Section Technology, March 22-24, 1966. Conf-660303.
- L. C. Schmid and G. D. Seybold, "Computer Analysis of Reactor Noise Data," Physics Research Quarterly Report, January, February, March, 1966, BNWL-284, April 15, 1966.
- R. C. Liikala and U. P. Jenquin, "Physics Survey Calculations for a $\text{PuO}_2\text{-UO}_2$ Fuel in Idealized H_2O Power Reactors," BNWL-305, July 14, 1966.
- L. J. Page, "Program GAMMAX," BNWL-CC-773, August 8, 1966.
- L. J. Page, Program BETA, "A Reactor Kinetics Code," BNWL-CC-788, August 23, 1966.
- D. H. Thomsen, "Thermalization Computational Study - Comparison of Results Obtained Using the RBU Monte Carlo, Thermos, and Program S-XIII Codes." Reactor Physics Quarterly Report, July-September, 1966, BNWL-340, October 15, 1966.
- R. C. Liikala and W. L. Purcell, "Results of Thermalization Calculations for $\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$ Lattices," Physics Research Technical Activities Quarterly Report, July-September, 1966, BNWL-340, October 15, 1966.
- W. W. Porath, "Three-Dimensional Analysis of Plutonium Fuel MTR Cores: Preliminary Results," Reactor Physics Department Technical Activities Quarterly Report, July-September, 1966, BNWL-340, October 15, 1966.
- V. O. Uotinen and S. Kobayashi, "Temperature Coefficients of $\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$ Lattices," Trans. Am. Nucl. Soc., Vol. 10, pp. 564-565 (1967).
- R. I. Smith, E. C. Davis, Jr., and L. D. Williams, "Critical Experiments with Phoenix Fuel in a MTR Mockup," Trans. Am. Nucl. Soc., 10, 537 (1967).
- D. H. Thomsen, "Analysis of a UO_2 19-Rod Cluster Experiment with the RBU Monte Carlo Code," Trans. Am. Nucl. Soc., 10, #1, 238 (1967), and BNWL-SA-1176 (1967).
- U. P. Jenquin, " $\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$ Systems-Survey Calculations," Reactor Physics Department Technical Activities Quarterly Report October, November, December, 1966, BNWL-400, April 1967.
- R. P. Matsen, "Analysis of Burnup Data," Reactor Physics Department Technical Activities Quarterly Report October, November, December, 1966, BNWL-400, April 1967.
- D. E. Christensen, "Gamma Scan Facility to Acquire Burnup Data by a Nondestructive Method," Reactor Physics Department Technical Activities Quarterly Report October, November, December, 1966, BNWL-400, April 1967.
- J. H. Lauby and J. W. Kutcher, "Measurements of ^{10}B Concentrations Using Reactivity Techniques," Reactor Physics Department Technical Activities Quarterly Report October, November, December, 1966, BNWL-400, April 1967.
- J. W. Kutcher and J. H. Lauby, "Batch Core Critical Tests in the PRTR," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1966, BNWL-400, April 1967.

V. O. Uotinen, "Experiments with Neutron Absorbing Rods," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1966, BNWL-400, April 1967.

S. Kobayashi and V. O. Uotinen, "Preliminary Calculations for H₂O-Moderated Lattices of UO₂-4 wt% PuO₂ Fuel Rods," BNWL-CC-1205, May 1967.

L. J. Page, "Theory Experiment Correlation of Burnup Data from Al-1.8 wt% Pu Fuel (Initially 6 at% Pu²⁴⁰)," BNWL-393, July 1967.

D. R. Skeen, "Program ABH: The Modified ABH Method of Cell Homogenization for Tempest," BNWL-515, July 1967.

D. R. Skeen and L. J. Page, "The Battelle Version of the THERMOS Code," BNWL-516, September 1967.

R. I. Smith, L. D. Williams and E. C. Davis, Jr., "PRCF-Phoenix Fuel Experiment," Reactor Physics Department Technical Activities Quarterly Report, January, February, March 1967, BNWL-472, October 1967.

D. E. Christensen, E. B. Reppond and W. A. Reardon, "Burnup Data from Low-Exposure Plutonium-Aluminum Fuel," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, October 1967.

D. E. Christensen, E. B. Reppond and W. A. Reardon, "Burnup Data from High-Exposure Plutonium-Aluminum Fuel," Reactor Physics Department Technical Activities Quarterly Report, January, February, March 1967, BNWL-472, October 1967.

R. P. Matsen, "Analysis of Plutonium-Aluminum Fuel Burnup Data," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, October 1967.

J. W. Kutcher and J. H. Lauby, "FERTF Zero-Power Tests in the PRTR," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, October 1967.

R. C. Liikala and W. L. Purcell, "Correlation of Effective Cross Section Ratios of Plutonium Isotopes," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, October 1967.

U. P. Jenquin, "Photoneutron Calculations for D₂O Systems," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, October 1967.

V. O. Uotinen and S. Kobayashi, "Power Peaking in UO₂-2 wt% PuO₂-H₂O Cores," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, October 1967.

S. Kobayashi and V. O. Uotinen, "Calculations for H₂O Moderated Lattices of UO₂-4 wt% PuO₂ Fuel Rods," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, October 1967.

C. M. Heeb, "The Application of Multigroup Neutron Transport Theory to Phoenix Fuel Cores," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, October 1967.

G. J. Busselman and D. N. Adrian, "Pre-experimental calculations of the PRCF - Phoenix Core," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, October 1967.

W. W. Porath, "Investigation of Power Peak Suppressors in the MTR-Phoenix Core," Reactor Physics Department Technical Activities Quarterly Report, January, February, March, 1967, BNWL-472, October 1967.

J. W. Kutcher and J. H. Lauby, "Analysis of Batch Core Critical Tests in the PRTR," Reactor Physics Department Technical Activities Quarterly Report, April, May, June, 1967, BNWL-534, January 1968.

J. W. Kutcher and G. J. Konzek, "FERTF Zero-Power Tests in the PRTR," Reactor Physics Department Technical Activities Quarterly Report, April, May, June, 1967, BNWL-534, January 1968.

D. H. Thomsen, Analysis of a UO_2 19-Rod Cluster Experiment With The RBU Monte Carlo Code," Reactor Physics Department Technical Activities Quarterly Report, April, May, June, 1967, BNWL-534, January 1968.

R. I. Smith, E. C. Davis, Jr., L. D. Williams and A. R. Maki, "PRCF-Phoenix Fuel Experiment," Reactor Physics Department Technical Activities Quarterly Report, April, May, June, 1967, BNWL-534, January 1968.

C. M. Heeb, "The Analysis of Phoenix Fuel Critical Approach Experiments," Reactor Physics Department Technical Activities Quarterly Report, April, May, June, 1967, BNWL-534, January 1968.

J. L. Carter, "HRG Data Tape," Reactor Physics Department Technical Activities Quarterly Report, April, May, June, 1967, BNWL-534, January 1968.

D. R. Skeen and A. G. Gibbs, "Current Calculation in Thermos," Reactor Physics Department Technical Activities Quarterly Report, July, August, September, 1967, BNWL-634, February 1968.

L. J. Page and D. R. Skeen, "Battelle Revised Hammer Code," Reactor Physics Department Technical Activities Quarterly Report, July, August, September, 1967, BNWL-634, February 1968.

R. I. Smith, E. C. Davis, Jr., L. D. Williams, and W. P. Stinson, "PRCF Phoenix Fuel Experiment," Reactor Physics Department Technical Activities Quarterly Report, July, August, September, 1967, BNWL-634, February 1968.

U. P. Jenquin and T. M. Traver, "Physics Calculations for Plutonium Phoenix Fuel Loadings in a Maritime Reactor," Reactor Physics Department Technical Activities Quarterly Report, July, August, September, 1967, BNWL-634, February, 1968.

U. P. Jenquin, R. C. Liikala, and W. L. Purcell, "Physics Characteristics of Plutonium Phoenix Fueled Loadings in the PRTR," Reactor Physics Department Technical Activities Quarterly Report, July, August, September, 1967, BNWL-634, February 1968.

U. P. Jenquin and W. L. Purcell, "Physics Characteristics of Thorium Fueled Loadings in the PRTR," Reactor Physics Department Technical Activities Quarterly Report, July, August, September, 1967, BNWL-634, February 1968.

D. R. Skeen and A. G. Gibbs, "Current Calculation in Thermos," Reactor Physics Department Technical Activities Quarterly Report, July, August, September, 1967, BNWL-634, February 1968.

C. M. Heeb, "PRCF-Phoenix Analysis," Reactor Physics Department Technical Activities Quarterly Report, July, August, September, 1967, BNWL-634, February 1968.

J. W. Kutcher, R. E. Harris and W. P. Stinson, PRTR Batch Core Interim Critical Tests - Series I," Plutonium Utilization Program Technical Activities Quarterly Report, September, October, November, 1967, BNWL-654, February 1968.

U. P. Jenquin and T. M. Traver, "Physics Characteristics of Reduced Density UO_2 - PuO_2 Loadings in the PRTR," Plutonium Utilization Program Technical Activities Quarterly Report, September, October, November, 1967, BNWL-654, February 1968.

V. O. Uotinen, S. Kobayashi, and U. P. Jenquin, "Temperature Coefficients of PuO_2 - UO_2 - H_2O Lattices," Plutonium Utilization Program Technical Activities Quarterly Report, September, October, November, 1967, BNWL-654, February 1968.

D. E. Christensen, E. S. Murphy, and G. Manca, "Burnup Data from Gamma Scanning," Plutonium Utilization Program Technical Activities Quarterly Report, September, October, November, 1967, BNWL-654, February 1968.

L. D. Williams and R. Martinelli, "Reactor Applications Experiments," Plutonium Utilization Program Technical Activities Quarterly Report, September, October, November, 1967, BNWL-654, February 1968.

C. W. Lindenmeier, "An Improved Short Time Approximation for Evaluating the Eggelstaff-Schofield Scattering Law Model," Plutonium Utilization Program Technical Activities Quarterly Report, September, October, November, 1967, BNWL-654, February 1968.

J. R. Worden, "Plutonium Utilization Studies," Plutonium Utilization Program Technical Activities Quarterly Report, September, October, November, 1967, BNWL-654, February 1968.

D. E. Deonigi, "Optimum Density for Plutonium Fuels," Plutonium Utilization Program Technical Activities Quarterly Report, September, October, November, 1967, BNWL-654, February 1968.

V. O. Uotinen and A. L. Pitner, "Reactivity Worth of A Central Safety Rod with and without an Internal Scatterer," BNWL-618, February 1968.

N. E. Carter, "Review of Physics Methods Used for Computing the Kinetics Behavior of Large Power Reactors," BNWL-738, January 1968.

J. W. Kutcher, R. E. Harris and W. P. Stinson, "PRTR Batch Core Interim Critical Tests - Series I," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1967, BNWL-685, March 1968.

U. P. Jenquin and T. M. Traver, "Physics Characteristics of Reduced Density UO_2 - PuO_2 Loadings in the PRTR," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1967, BNWL-685, March 1968.

V. O. Uotinen, S. Kobayashi and U. P. Jenquin, "Temperature Coefficients of PuO_2 - UO_2 - H_2O Lattices," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1967, BNWL-685, March 1968.

D. E. Christensen, E. S. Murphy, and G. Manca, "Burnup Data from Gamma Scanning," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1967, BNWL-685, March 1968.

L. D. Williams and R. Martinelli, "Reactor Applications Experiment," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1967, BNWL-685, March 1968.

G. L. Gelhaus and R. C. Liikala, "Preliminary Calculated Results for UO_2 -4 wt% PuO_2 - H_2O Lattices," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1967, BNWL-685, March 1968)

E. C. Davis and L. D. Williams, "PRCF-Phoenix Fuel Experiment," Reactor Physics Department Technical Activities Quarterly Report, October, November, December, 1967, BNWL-685, March 1968.

V. O. Uotinen, "Analysis of Temperature Coefficients," Plutonium Utilization Program Technical Activities Quarterly Report, December 1967, January, February 1968, BNWL-739, March 1968.

D. D. Matsumoto and R. H. Holeman, "ZODIAC G," Plutonium Utilization Program Technical Activities Quarterly Report, December 1967, January, February 1968, BNWL-739, March 1968.

U. P. Jenquin and T. M. Traver, "Physics Calculations for CVTR-Batch Core Fueled Loadings in the PRTR," Reactor Physics Department Technical Activities Quarterly Report, January, February, March 1968, BNWL-775, May 1968.

D. F. Newman, "The Experimental Determination of the Fast Fission Effect in a Water Moderated UO_2 - PuO_2 Lattice," Reactor Physics Department Technical Activities Quarterly Report, January, February, March 1968, BNWL-775, May 1968.

ARGONNE NATIONAL LABORATORY

P. H. Kier, B. J. Toppel and R. F. Jones, "Initial Critical Experiments of the EBWR Plutonium Recycle Program," Trans. Am. Nucl. Soc., 9, 2, 446 (1966)

P. H. Kier, et al., "Safety Analysis Associated With the Plutonium Recycle Experiment in EBWR," ANL-6841, November, 1964.

C. F. Till, "PuO₂-UO₂ Fueled Thermal Critical Experiments," Argonne National Laboratory Reactor Physics Division Annual Report, July 1, 1963 to June 30, 1964, USAEC Report ANL-7010, January 1965.

B. J. Toppel, "EBWR Plutonium Recycle Experiment," Argonne National Laboratory Reactor Physics Division Annual Report, July 1, 1962 to June 30, 1964, USAEC Report, ANL-7010, January 1965.

P. W. Kier, "EBWR Plutonium Recycle Safety Analysis," Argonne National Laboratory Reactor Physics Division Annual Report, July 1, 1963 to June 30, 1964, USAEC ANL-7010, January 1965.

P. H. Kier, "Slow Neutron Spectrum Studies by Neutron Activation," Argonne National Laboratory Reactor Physics Division Annual Report, July 1, 1964 to June 30, 1965, USAEC Report, ANL-7110, December, 1965.

P. H. Kier, "The EBWR Plutonium Recycle Experimental Program," Argonne National Laboratory Reactor Physics Division Annual Report, July 1, 1964 to June 30, 1965, USAEC Report ANL-7110, December 1965.

P. H. Kier, "Initial Critical Experiments of the Experimental Boiling Water Reactor (EBWR) Plutonium Recycle Program," Argonne National Laboratory Reactor Physics Division Annual Report, July 1, 1965 to June 30, 1966, USAEC Report ANL-7210, December 1966.

P. H. Kier, "Neutronic Aspects of Plutonium Recycle in Thermal Reactors," Power Reactor Technology and Reactor Fuel Processing, Vol. 10, No. 2, Spring 1967.

GENERAL ELECTRIC - VALLECITOS LABORATORY AND SAN JOSE

R. L. Crowther, et al., "Plutonium Utilization in Boiling Water Reactor Power Plants," Commercial Plutonium Fuels Conference, March 1-2, Washington, D.C, Conf. 66038.

M. P. Lagache, "Quantitative Autoradiography for Plutonium Fuel Analysis," Trans. Am. Nucl. Soc., 7, 1, 109, (1964).

P. Greebler, et al., "Recycle of Plutonium in Low-Enrichment Light Water Reactors," Proceedings of the Second International Conference on the Peaceful Uses of Atomic Energy, 13, 251 (P/2167), Geneva, 1958.

M. A. Robkin and R. Goldstein, "Absorption in Low Energy Resonances," Trans. Am. Nucl. Soc., 7, 2, 276 (1964).

M. A. Robkin, "Burnup Analysis of Lumpy Fuels," Trans. Am. Nucl. Soc., 8, 2, 385 (1965).

P. G. Aline and R. J. McWhorter, "The Effective Value of η for Pu^{239} ," Trans. Am. Nucl. Soc., 5, 2, 369 (1962).

P. Greebler and W. H. Harker, "Thermalization Studies on a Heterogeneous Lattice Containing Plutonium," Trans. Am. Nucl. Soc., 1, 1, 47 (1958).

J. G. Carver, "Plutonium Subcritical Experiment Program," USAEC Report GEAP-4730, General Electric Company, 1964.

P. W. Greebler, W. Harker and J. Harriman, "Neutron Thermalization Calculations for a Heterogeneous Lattice Containing Uranium and Plutonium Fuel in Water," Nucl. Sci. and Eng. 6, 128-134 (1959).

J. G. Carver and C. R. Porter, "Temperature Dependence of Nuclear Reaction Rates in a $\text{Pu-H}_2\text{O}$ Lattice," Trans. Am. Nucl. Soc., 10, 190 (1967).

E. A. Evans, D. L. Fischer, W. W. Macnabb, "Plutonium Utilization in Boiling Water Power Reactors," International Atomic Energy Agency Symposium on the Use of Plutonium As a Reactor Fuel, Brussels, March 13-17, 1967.

E. A. Evans, et al., "Plutonium Utilization in Boiling Water Reactors - A Review of EEI-GE Joint Program," Trans. Am. Nucl. Soc., 10, 2, 497 (1967).

R. L. Crowther, et al., "The Physics of Non-Uniform BWR Lattices," Joint International Conference on the Physics Problems in Thermal Reactor Design, 27-29, June 1967 at the Institution of Civil Engineers, London, SW1.

WESTINGHOUSE - APED

W. L. Orr, "A Nuclear Design Review of the Saxton Partial Plutonium Core," Trans. Am. Nucl. Soc., 9, 1, 280 (1966).

H. A. Risti and A. J. Impink, "Reactor Physics Considerations in Design of a Power Reactor Partially Loaded with $\text{PuO}_2\text{-UO}_2$," Trans. Am. Nucl. Soc., 8, 1, 209 (1965).

H. A. Risti and H. E. Krug, "Results of Comparisons of Thermal Calculational Models for Heterogeneous Light Water Moderated $\text{PuO}_2\text{-UO}_2$ Reactor Systems," Trans. Am. Nucl. Soc., 8, 1, 199 (1965).

W. L. Orr, et al., "A Comparison of Analysis with Experiment for the Saxton Plutonium Critical Experiments," Trans. Am. Nucl. Soc., 8, 2, (1965).

W. L. Orr, F. L. Langford and R. D. Leamer, "Physics Development for Plutonium Utilization in Pressurized Water Reactors," Joint International Conference on the Physics Problems in Thermal Reactor Design, 27-29 June 1967 at the Institution of Civil Engineers, London, SW1.

R. J. Allio and R. S. Miller, "Plutonium Recycle in Pressurized Water Reactors," International Atomic Energy Agency Symposium on the Use of Plutonium as a Reactor Fuel, Brussels, March 13-17, 1967.

G. N. Hamilton, et al., "Calorimetric and Spectrometric Techniques to Relate Zonewise Power Distributions in Pu-U Multizoned Cores," Trans. Am. Nucl. Soc., 8, 2, (1965).

L. Bindler and G. C. Ponchelet, "Evaluation of Computational Models and Thermal Cross-Section Data in the Analysis of $\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$ Lattices," Trans. Am. Nucl. Soc., 8, 2, (1965).

N. R. Nelson, et al., "Saxton Plutonium Quarterly Progress Reports, WCAP-3385-1 through 6, and Semi Annual Report 3385-8.

H. W. Graves, Jr., "Plutonium Utilization in Pressurized Water Reactors," Commercial Plutonium Fuels Conference, March 1-2, 1966, Conf. 660308.

F. W. Kramer, R. S. Miller, "The EEI-Westinghouse Plutonium Recycle Study Program," Trans. Am. Nucl. Soc., 10, 2, 498 (1967).

W. J. Eich, "Analysis of $\text{PuO}_2\text{-UO}_2$ Critical Experiments," Trans. Am. Nucl. Soc., 10, 1, 306 (1967).

UNITED NUCLEAR

J. R. Tomonto, "A Comparison of Measured and Calculated Isotopic Compositions as a Function of Burnup in a 1.5 wt% $\text{PuO}_2\text{-UO}_2$, Light-Water Moderated Fuel Assembly," Trans. Am. Nucl. Soc. 9, 1, 293 (1966).

J. R. Tomonto and A. H. Kazi, "An Approximation to Space-Energy Slowing Down Through Use of Multi-Group Thermal Self-Shielding Factors in $\text{PuO}_2\text{-UO}_2$ Lattice Calculations," Trans. Am. Nucl. Soc. 8, 2, (1965).

G. A. Sofer, et al., "Plutonium Utilization and Burnup Calculation Methods for Light-Water Moderated Reactors," IAEA Symp. Nucl. Mat. Mgmt., Vienna, Austria (August 1965).

James R. Tomonto and W. Jager, "Characteristics of Zoned $\text{PuO}_2\text{-UO}_2$ Fuel Rods as Recycle Fuel in Boiling-Water Reactors," Trans. Am. Nucl. Soc., 10, 2, 514 (1967).

James S. Tulenko and James R. Tomonto, "Fuel-Cycle Evaluation of Stainless-Steel and Zircaloy Cladding in Water-Moderated Systems Utilizing $\text{PuO}_2\text{-UO}_2$ Fuel," Trans. Am. Nucl. Soc., 10, 2, 515, (1967).

J. Celnik, J. R. Tomonto, and J. S. Tulenko, "Representation of Fission Products in Thermal Power Reactors Containing UO_2 and Plutonium Recycle Fuels," Trans. Am. Nucl. Soc., 10, 2, 516, (1967).

J. Celnik and S. Kellman, "Evaluation of Some Nuclear Computational Methods for Use in Plutonium Recycle," Trans. Am. Nucl. Soc., 10, 1, 305, (1967).

J. Celnik, et al., "Evaluation of Plutonium Recycle Nuclear Calculation Methods by Comparison with Experimental Data," UNC-5168, February 28, 1967.

OTHER

G. Srikantiah and J. Chernick, "Plutonium Fueled Thermal Reactors with High Conversion Ratios," Nucl. Sci. & Eng. 21, 2, 165 (1965).

T. M. Murphy, "The Influence of Plutonium on the Design of Advanced Reactors," Trans. Am. Nucl. Soc. 9, 1, 213 (1966).

R. L. Olson and W. F. Graves, "Critical Experiments with Highly Burned-up Plutonium in Heavy Water," Trans. Am. Nucl. Soc. 9, 1, 181 (1966).

D. A. Klopp, "Effective Resonance-Absorption Integrals for Plutonium-240," Trans. Am. Nucl. Soc., 6, 2, 258 (1963).

Proceedings of the Commercial Plutonium Fuels Conference held on March 1-2, 1966 at Washington, D. C., Conf. 660308.

Plutonium As A Reactor Fuel, Proceedings of a Symposium, Brussels, March 13-17, 1967, International Atomic Energy Agency, Vienna, 1967.

APPENDIX

FUEL DESCRIPTIONS

FUEL DESCRIPTION:
 ALUMINUM-PLUTONIUM ALLOY (Al-Pu) FUELED-H₂O MODERATED LATTICE EXPERIMENT

Plutonium Enrichment (wt%)	1.83	5.01	2.01
Fuel Diameter (in.)	0.500	0.500	0.500
Fuel Length (in.)	43.83	24.0	36.0
Clad I.D. (in.)	0.506	0.506	0.506
Clad Thickness (in.)	0.030	0.030	0.030
Clad Material	Zircaloy-2	Zircaloy-2	Zircaloy-2
Mass of Al-Pu per rod (g/rod)	389.32	219.94	318.19
Mass of Pu per rod (g/rod)	7.12	11.01	6.383

Average Plutonium Composition (wt%)

²³⁹ Pu	93.97	94.09	81.01
²⁴⁰ Pu	5.58	5.33	16.46
²⁴¹ Pu	0.44	0.55	2.31
²⁴² Pu	0.014	0.03	0.20

Average Aluminum Composition (wt%)

Al	97.21	Assumed the same as	97.57
Ni	1.61	1.8 wt% Fuel	1.63
Fe	0.49		0.30
Si	0.69		0.10

FUEL DESCRIPTION: MIXED OXIDE ($\text{UO}_2\text{-PuO}_2$) FUELED- H_2O MODERATED LATTICE EXPERIMENTS

	<u>Type 1</u>	<u>Type 2</u>	<u>Type 3</u>	<u>Type 4</u>
Enrichment: $\frac{\text{Uranium}}{\text{Plutonium}}$	Depleted to ~0.16 1.50	Natural 2.00	Natural 4.0	Natural 6.60
Fuel Diameter (in.)	0.372	0.508	0.4975	0.3374
Fuel Length (in.)	48.50	36.0	36.0	36.6
Clad I.D. (in.)	0.372	0.508	0.4975	0.3445
Clad Thickness (in.)	0.027	0.030	0.0335	0.02325
Clad Material	Zircaloy-2	Zircaloy-2	Zircaloy-2	Zircaloy-4
Mass of $\text{UO}_2\text{-PuO}_2$ per rod (g/rod)	828.0	1128.0	1084.54	546.576
Mass of PuO_2 per Rod (g/rod)	12.42	22.56	43.20	36.074

Average Plutonium
Composition (wt%)

		<u>8% Fuel</u>	<u>16% Fuel</u>	<u>24% Fuel</u>		
^{239}Pu	91.41	91.62	81.11	71.76	75.38	90.49
^{240}Pu	7.83	7.65	16.54	23.50	18.10	8.57
^{241}Pu	0.73	0.70	2.15	4.08	5.08	0.89
^{242}Pu	0.03	0.03	0.20	0.66	1.15	0.04
^{238}Pu	-	-	-	-	0.28	-

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FUEL DESCRIPTION:
URANIUM DIOXIDE (UO₂) FUELS UTILIZED IN REACTOR APPLICATIONS EXPERIMENTS

	<u>PNL-CNEN Experiments</u>	<u>WAPD-ESADA Experiments</u>	<u>WAPD-SAXTON Experiments</u>
Uranium Enrichment (wt% ²³⁵ U)	2.32	2.719	5.742
Fuel Diameter (in.)	0.440	0.400	0.357
Fuel Length (in.)	36.0	48.0	36.6
Clad I.D. (in.)	0.440	0.4054	0.361
Clad Thickness (in.)	0.030	0.03145	0.015
Clad Material	Aluminum	Zircaloy-4	304-SS
Mass of UO ₂ per Rod (g/rod)	825.6	1028.02	604.25
Mass of ²³⁵ U per Rod (g/rod)	16.86	905.93	30.578

FUEL DESCRIPTION: PLUTONIUM FUELED-D₂O MODERATED LATTICE EXPERIMENTS

<u>Fuel Type</u>	<u>19-Rod UO₂-PuO₂ Cluster</u>	<u>19-Rod Al-Pu Cluster</u>	<u>19-Rod UO₂ Cluster</u>
Enrichment:			
Uranium	Natural	-	Natural
Plutonium	2.0 wt%	1.81 wt%*	-
Fuel Diameter (in.)	0.505	0.500	0.505
Fuel Length (in.)	58.5	87.5	88.0
Clad I.D. (in.)	0.505	0.505	0.505
Clad Thickness (in.)	0.030	0.030	0.030
Clad Material	Zircaloy-2	Zircaloy-2	Zircaloy-2
Center-to-Center Rod Spacing (in.)	0.637	0.637	0.637
Fuel Material Mass (g/rod)	1792.0 of UO ₂ -PuO ₂	271.91 of Pu	2518 of UO ₂
Fuel Material Composition (wt%)			
²³⁹ Pu	90.477	93.97	
²⁴⁰ Pu	8.186	5.58	
²⁴¹ Pu	1.273	0.44	
²⁴² Pu	0.064	0.014	

*Aluminum contains 1.35 wt% Ni, 1.10 wt% Si, and 0.33 wt% Fe.

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