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REPORT

**PHYSICS EXPERIMENT
HIGH POWER DENSITY CORE OF THE PRTR**

**J. R. WORDEN, W. L. PURCELL,
and L. C. SCHMID**

JANUARY, 1966

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PHYSICS EXPERIMENT
HIGH POWER DENSITY CORE OF THE PRTR

By

J. R. Worden, W. L. Purcell,
and L. C. Schmid

Reactor Physics Section
Physics and Instruments Department

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PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON

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PHYSICS EXPERIMENT -
HIGH POWER DENSITY CORE OF THE PRTR

I. INTRODUCTION

A high power density (HPD) core is scheduled for the Plutonium Recycle Test Reactor (PRTR) during the early part of FY-1967. This report includes a description of the core loading, an outline of the experimental program as now envisaged, a description of the methods used for analysis of the core, and the nucleonic properties of the core as deduced from its analysis.

An important feature of this core will be the production of experimental data relating reactivity characteristics and isotopic composition of the fuel as functions of time. These data will provide checks on methods of calculating the reactivity limited lifetime of reactor cores, a property that if not predicted accurately can lead to large fuel inventory costs. Computational techniques have been applied to critical experiments and burnup experiments with relatively good results. However, calculations concerning the multiplication constant of a core as a function of exposure and the reactivity limit of exposure have not been evaluated since no data from an unperturbed loading exists to test predictions of reactivity for the whole reactor.

Another important feature of this core is the opportunity to study the performance of ceramic, packed-particle fuel elements bearing plutonium in an environment similar to an operating power reactor.

The core loading has been designed to meet three goals; 1) accumulation of relatively large exposures ($\sim 10,000$ MWd/ton U)* in a reasonable time (~ 2 years), 2) a rather large change in reactivity ($> 20\% \Delta k/k$) over the lifetime of the core, and 3) provision of an environment, as free as possible from extraneous perturbations, for the basic core.

*The unit of exposure, MWd/ton U, is defined as total energy generated in megawatt-days per ton of uranium initially in the reactor.

The first goal will be met by a reduction in the number of fuel elements constituting the core of the reactor and a shortening of their length, thus increasing the specific power. The second goal will be accomplished by selection of a fuel containing relatively large quantities of fissile material. Two significant features of the core that will contribute to the realization of the third goal are 1) the use of soluble shim (boron) in the moderator for control of the reactor instead of mechanical shims, and 2) the use of a buffer zone to isolate the basic core from a fuels test region. This latter arrangement is not expected to compromise the ability to carry out program obligations for fuel testing. In fact, fuel testing capabilities will be improved because of the greater specific power.

Calculated quantities presented in this report are the results of a preliminary analysis accomplished to determine the properties of a core that would meet the goals as described above and to scope the experimental program. No experimental data has been used to normalize the calculations. The results of refined calculations and comparison with experiments are anticipated to allow more detailed planning of the experimental program and to show which parameters of the core affect the results more sensitively.

II. SUMMARY

The HPD core will consist of UO_2 -2 wt% PuO_2 fuel clusters, $58\frac{1}{2}$ in. in length, as compared to the present 88 in. Fifty-four clusters will be irradiated in the PRTR until an average exposure of approximately 10,000 MWd/ton U is attained. This exposure would be reached after about 1 yr of full time operation at 70 MW.

The experimental program with the HPD core is designed to determine both the excess reactivity and fuel composition of the core as functions of time. The core will be irradiated in increments of about 2 months duration each, at which times selected fuel rods will be analyzed to determine isotopic composition. Critical experiments will be performed at these times to determine various reactivity coefficients associated with burnup of the core.

The initial neutron multiplication factor for the HPD core would be about 1.26 with no control mechanism in the reactor.* Control will be attained by adding boron to the moderator. An initial concentration of 130 ppm by weight of natural boron will be required.

III. DESCRIPTION OF CORE

The PRTR, described in detail elsewhere,⁽¹⁾ is a pressure tube reactor which is moderated, cooled, and reflected by heavy water. Eighty-five shroud tubes penetrate the core; each contains a helium-filled gas gap, a process tube, and, normally, a 19-rod cluster of fuel rods. These channels form a hexagonal array with a lattice spacing of 8 in. The core configuration is shown in Figure 1.

The new core of the PRTR will consist of 54 clusters of 19 rods each, $58\frac{1}{2}$ in. in length as compared to the present 88 in., located around the central channel of the reactor. These fuel elements will be made of UO_2 -2 wt% PuO_2 . Natural uranium which contains 0.71 wt% U^{235} and plutonium containing about 8 wt% Pu^{240} will be used in making these elements. These 54 fuel clusters constitute the basic core for the physics experiment.

Surrounding this basic core will be a buffer region consisting of 12 elements of similar composition to the basic core. These elements will provide replacement rods for those removed from clusters in the basic core for chemical and isotopic analysis. It will also be possible to closely match the exposure on any cluster in the basic core in the event of a failure.

The next 13 positions will be available for irradiation testing of fuel elements. These elements, for the most part, will have experienced large exposures (5000 to 10,000 MWd/ton U) before the HPD core

*The calculations described in this report are based on the assumption that the radial reflector will be D_2O . If light water is used, the calculated effect will be to reduce the effective neutron multiplication by about 1%.

FIGURE 1
PRTR Core Configuration HPD Program

irradiation begins. At present these elements are expected to contain 0 to 2 wt% PuO_2 in $\text{PuO}_2\text{-UO}_2$. Some testing of new fuel element designs will be accomplished in this region. Position 1550, as shown in Figure 1, is expected to contain the Fuel Element Rupture Test Facility⁽²⁾ during power operation. Finally the last five positions will be cooled with D_2O and will be available if required for special tests.

There will be approximately 30 in. of D_2O reflector below the core; and assuming an operating moderator level of 100 in., the top reflector will be 10 in. thick. No boron will be added to either the coolant or the radial reflector.

IV. OUTLINE OF EXPERIMENTAL PROGRAM

The experimental program in the PRTR with the HPD core is designed to determine both the excess reactivity and fuel composition of the core as functions of time. A flow chart of the experiments which comprise the program is shown in Figure 2. Only the major experimental categories are shown; and, in some cases, the information gained in one experiment will dictate the desirability of obtaining other data. Also, as the information is obtained, it will become evident which experiments do not require frequent repetition. Other changes might be required depending upon information obtained from experiments to be conducted in the Plutonium Recycle Critical Facility (PRCF). These experiments in the PRCF will be conducted with fuel elements from the HPD core before the core experiment will begin. Outlines of the experiments in the PRTR are very similar to those already completed for the Critical⁽³⁾ and Power⁽⁴⁾ Tests for the zoned loadings in the PRTR. The program for the HPD core can be considered as occurring in three phases: 1) critical experiments, 2) power experiments, and 3) burnup experiments.

An outline of the critical experiments is given in Table I. These tests provide initial burnup conditions and tests of the calculational methods. Both the experimental data and the evaluation of calculational techniques will provide information for future operation of the reactor and for evaluation of Safeguards Analyses.

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TABLE I
OUTLINE OF CRITICAL AND BURNUP TESTS

<u>Experiment Number</u>	<u>Title</u>	<u>Description</u>
1	Approach to Critical	Establish initial criticality, moderator level at 100 in. or greater, boron free.
2	Boron and Fuel Addition	Add fuel until critical moderator level is lowered to 60-65 in. Add boron in small increments until critical moderator level is raised to 100 in. or greater. Continue this stepwise fuel-boron addition until 55 fuel clusters have been added. Tests 3, 4, and 6 may be conducted for intermediate fuel loadings and boron concentrations.
3	Moderator Level Sensitivity	Determine the reactivity effects of moderator height and changes. The results can be used to determine the migration area and the effect of reactor scrams.
4	Mechanical Shim Worth	Determine worth of selected shim configurations as required for supplemental means of control of excess reactivity.
5	Reflector Savings	Determine excess reactivity of the loading with and without a radial reflector.
6	Kinetics	Determine neutron lifetime and total delayed fraction of neutrons by noise analysis. Also, the reactivity of the reactor may be oscillated and cross correlations experiments conducted during this experiment.
7	Cell and Reactor Flux Measurements	Determine spatial and energy distribution of neutrons in the core. Results will be used to verify heat generation calculations for a fuel cluster and flux calculations for the core. This experiment will be done using foils.

TABLE I (contd.)

Experiment Number	Title	Description
8	Coolant Replacement	Determine reactivity effects of loss of D ₂ O coolant in the central fuel channel. Possibly this measurement would be made at positions other than the center one.
9	Fuel-Coolant Temperature Coefficient	Determine change in reactivity resulting from varying fuel and coolant temperatures together. Probably nuclear heating will be utilized for the temperature changes.
10	Moderator Temperature Coefficient	Determine reactivity change when moderator temperature is changed. Probably nuclear heating will be utilized to obtain the temperature changes.
11	Load to 67 Element Core	Add elements around basic core and determine moderator levels for a critical loading. The number of elements loaded will be calibrated against the critical moderator level at the boron concentration for 55 elements.
12	Load Core for Power Operation	Other irradiated fuel elements and test facilities will be loaded as in Test 2. The number of items loaded will be calibrated against the critical moderator level at the boron concentration for 55 elements. After the core is loaded the boron concentration is adjusted to bring the moderator level to its operating height. Other tests such as Numbers 3, 4, 5, 7, and other reactivity perturbation measurements will be made during this test.

After the critical experiments with the 55 element core are completed, the remainder of the fuel elements and irradiation facilities will be loaded.

An outline of the power experiments is given in Table II. Again, these tests serve the dual purpose of characterizing the basic core data as well as providing operational and safeguards information. Important data to be obtained from these tests are the reactivity effects of heating the core to operating temperatures, the reactivity worth of equilibrium fission products (principally xenon and samarium), and information about the stability of the core.

Scheduled burnup experiments will be conducted after about every 2 months of full power operation and will require shutting down the reactor and repeating the critical and power tests except in lesser detail. It is anticipated that these testing periods will be about 2 to 3 weeks. At these times, a fuel rod from a cluster in the basic core will be destructively analyzed to determine its isotopic composition. This rod will be replaced by a rod of similar exposure from the region outside the core. This sequence of irradiating and testing will continue until fission product poisoning and fuel element burnout prohibit further operation of the core without recharging fuel.

V. DESCRIPTION OF CALCULATIONAL MODELS

The calculations made for purposes of scoping the experiment in the PRTR are divided into two categories: 1) those describing the properties of the initial core loading, and 2) predictions of the nucleonic behavior of the core during irradiation. The results of the former calculations include neutron flux distributions, multiplication constants, control requirements, temperature coefficients, void coefficients, and kinetic parameters (neutron lifetime and delayed fractions). The second category of calculations includes the variation with exposure of most of these same parameters as well as isotopic compositions within the fuel during burnup. The methods used for each of the two categories of calculations will be described.

The core was assumed to consist initially of 55 unirradiated fuel elements. Surrounding this basic core was a region assumed to consist

TABLE II
OUTLINE OF POWER AND BURNUP TESTS

<u>Experiment Number</u>	<u>Title</u>	<u>Description</u>
1	Temperature Coefficients of Reactivity	Determine fuel, coolant, and moderator temperature coefficients of reactivity under power operating conditions.
2	Xenon Poisoning	Determine the reactivity worth of transient and equilibrium xenon. Occasionally the calibration will be made against boron concentration and more frequently using shims.
3	Samarium Poisoning and Long Term Reactivity Changes	Determine the excess reactivity at various times and at various power levels.
4	Photoneutron Flux	Determine the neutron flux level in the reactor due to fissioning of deuterium by certain fission product gammas.
5	Stability	Measure reactor transfer function under different power and burnup conditions.
6	Fuel Temperature	Determine spatial distribution of temperatures radially and axially within a cluster.
7	Power Decay	Measure transient decay of reactor power following a scram. Results will provide test of calculations of heat removal requirements during emergency conditions. It may be desirable to measure the transient by shutting the reactor down without scrambling it.
8	Reactor Flux Measurements	Determine spatial and energy distributions of neutrons in the core. Generally these measurements will be made using the PRTR wire monitor.

of D₂O and 30 empty shroud tubes. The inside diameter of the calandria is 84 in. A pure D₂O reflector, 2 ft thick, surrounds the calandria.*

A. Static Core Properties

The initial step in the analysis was a homogenization of the heterogeneous core. Cross sections were obtained for four energy groups as defined in Table III.

TABLE III
GROUP BOUNDARY ENERGIES

<u>Group</u>	<u>Lower Energy</u>	<u>Upper Energy</u>
1	11.7 keV	10 MeV
2	2.38 eV	11.7 keV
3	0.683 eV	2.38 eV
4	0	0.683 eV

The distribution, $\phi(E, \bar{r})$, of neutrons in space and energy, over the energy range 0 to 0.683 eV was calculated using the integral form of the Boltzman transport equation in one space dimension. The usual Wigner-Seitz (cylindrical cell) approximation was made for a unit cell of the core. A cylindrical model for the fuel cluster was assumed (Figure 3) with the exception of calculations of temperature coefficients and shim worths. Calculations for the exceptions are based on a solid rod equivalent of the cluster. Average cross sections, $\bar{\Sigma}_i$, were calculated for the i^{th} material from the resulting flux distribution,

$$\bar{\Sigma}_i = \frac{\int_{V_i} \int_E \Sigma(E) \phi(E, \bar{r}) dE d\bar{r}}{\int_{\text{cell}} \int_E \phi(E, \bar{r}) dE d\bar{r}}$$

*See the footnote on page 5.

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FIGURE 3
PRTR HPD Fuel Element and Reactor Cell Geometry
Annular Volume Equivalent

where the spatial integrations are over the region containing the i^{th} material (numerator) and over the entire cell (denominator).

The cross sections for the homogeneous equivalent of the cell are then given by

$$\bar{\Sigma} = \sum_i \bar{\Sigma}_i .$$

Thus, the average cross sections are defined such that, when multiplied by the integrated flux in the cell, the product gives the total reaction rate occurring within the cell. All basic cross section data $\Sigma(E)$ used for the averaging were obtained from the Battelle Northwest Master Library.⁽⁵⁾

Above the energy region where thermalization effects are important, i. e., groups 1 through 3 of Table 3, the consistent P_1 approximate solution to Boltzmann's equation was used. Shielded resonance cross sections were calculated using the narrow resonance (NR) approximation.⁽⁶⁾ An equivalent solid rod for the 19 rod cluster was selected such that the surface area of this rod is 1.23 times the surface area of the envelope (rubber band surface) of the cluster. The factor, 1.23, accounts for internal surfaces within the cluster.⁽⁷⁾

These average cross sections were then used in a diffusion theory calculation (consistent P_1) of the macroscopic neutron flux radially over the reactor. Axial leakage of neutrons was accounted for by using an axial buckling, B_z^2 , which was calculated assuming a 30 cm extrapolation length at both top and bottom of the core.

The cell calculations were made using the computer codes, THERMOS⁽⁸⁾ for Group 4, and HRG⁽⁹⁾ for Groups 1, 2, and 3. Volume weighting was applied to the isotopic densities input to HRG, and an infinite, homogeneous core was assumed for the cell calculation. Resonance calculations were made for U^{238} , Pu^{239} , and Pu^{240} . Calculation of macroscopic flux distributions over the reactor and of system eigenvalues were accomplished using HFN.⁽¹⁰⁾ The reflector cross sections were calculated assuming a source distribution in energy of fission neutrons typical of Pu^{239} fueled systems⁽¹¹⁾ and assuming the flux shape

varies as $1/E$ above 0.683 eV. Reflector cross sections for Group 4 were obtained by averaging the appropriate cross sections over the flux shape at the cell boundary as calculated using THERMOS.

B. Burnup Properties

The expected behavior of the 55 element, HPD core during irradiation was determined from one-dimensional burnup studies. Two-group diffusion theory calculations were made alternately with burnup calculations in each fuel region. The study was made using the computer code, ALTHAEA.⁽¹²⁾

The cross sections used are based on the Westcott formalism⁽¹³⁾ using the $\Delta 4$ non-Maxwellian flux shape.* Reaction rates are given by

$$R = N \sigma n v_{2200}$$

where

$$\sigma = \sigma_{2200} [g(T) + rs(T)]$$

$$n = \text{neutron density}$$

$$N = \text{atom density}$$

$$T = \text{centigrade neutron temperature.}$$

In the calculations, the Westcott formalism is modified by

$$g_{\text{eff}}(T) = g(T)/F$$

where

$$F = \text{the thermal disadvantage factor and is approximated by}$$

$$F = F_{\text{CONST}} \times \bar{\Sigma}_a + \exp(-F_{\text{EXP}} \times \bar{\Sigma}_a)$$

and

$$\bar{\Sigma}_a = \text{the macroscopic fuel absorption cross section for the Maxwellian group of neutrons.}$$

Another modification is

$$s_{\text{eff}}(T) = \frac{s(T) + bg(T)}{\sqrt{1 + \frac{\Sigma N^i \sigma_o^i}{\text{SCA}}}} - bg_{\text{eff}}(T)$$

*This description of ALTHAEA is taken essentially verbatim from Reference (14). It is included here for completeness.

where

- $b =$ the value 1.176
- $\sigma_o^i =$ the effective resonance peak cross section of the i^{th} isotope corrected for the effect of Doppler broadening
- $\text{SCA} = \Sigma_p + 1/\bar{\ell}$
- $\Sigma_p =$ the potential scattering in the fuel region
- $N^i =$ the nuclei density of the i^{th} isotope
- $1/\bar{\ell} =$ the surface-to-volume term, $S/4V$, of the fuel region modified by the Bell correction. (13)

The neutron temperature is obtained from

$$T = T_m \left(1 + (0.62) \frac{\Sigma_a/F}{\xi \Sigma_s} \right)$$

where

- $T =$ the neutron absolute temperature
- $T_m =$ the moderator absolute temperature
- $\xi \Sigma_s =$ the slowing down power.

The spectral index, r , is determined by

$$r = \frac{\frac{\bar{\Sigma}_a}{F} + \text{SNF}}{\xi \Sigma_s \sqrt{\frac{\pi T_o}{4 T}} \sum_i \Sigma_{ao}^i s_{\text{eff}}^i(T)}$$

where SNF is the macroscopic nonfuel absorption cross section per unit fuel volume, and the sum is over the isotopes in the fuel. Input data for the calculations are given in Table IV.

TABLE IV
INPUT PARAMETERS FOR ALTHAEA

<u>Parameter</u>	<u>Value</u>
F_{CONST}	7.81
F_{EXP}	1.78
SNF	0.010
$\xi \Sigma_s$	0.15
SCA	0.6

Criticality was maintained during the burnup calculation by including sufficient boron at each burnup step to make the neutron multiplication unity.

The results of the calculations, described above, are given in the next section.

VI. CALCULATED RESULTS

A. Reactivity and Boron Requirements

The effective neutron multiplication, k_{eff} , varies linearly with time and average core exposure (Figure 4). The values of k_{eff} , calculated using ALTHAEA, were reduced by approximately 4% at each time step to force agreement to the value obtained using HFN for the core at operating temperatures (listed in Table V) and zero exposure.

TABLE V
ASSUMED OPERATING TEMPERATURES

<u>Cell Component</u>	<u>Temperature, °C</u>
Fuel	980
Coolant	265
Moderator	62

FIGURE 4

Exposure Lifetime of HPD Core, 55 Elements, 70 MW

This difference in eigenvalues is due primarily to the difference in cross sections used in the two calculations. The cross section values for Pu^{239} used in ALTHAEA are based on those values recommended by Leonard⁽¹⁴⁾ because evidence exists that indicates these cross sections, when used with this calculational model, adequately predict isotopic concentrations as a function of exposure.⁽¹⁵⁾ The evidence is obtained from analytical correlation of experimental data from systems bearing plutonium. On the other hand, analysis of critical experiments⁽¹⁶⁾ indicates that the

Pu^{239} cross section values recommended by Sher ⁽¹⁷⁾ give better results for the reactivity of a given core configuration when the set is used with the THERMOS-HRG-HFN calculational model. A satisfactory explanation of this apparent paradox is presently unavailable but probably is related to the approximate nature of the calculational models.

From Figure 4, the reactivity effect of introducing equilibrium fission products, principally Xe^{135} and Sm^{149} , is $\frac{\Delta k}{k} = -4\%$.

The calculations leading to Figure 4 assume the reactor operates at a constant power of 70 MW. The maximum exposure of any fuel cluster in the core occurs in the center element and, at this position, is a factor of ~ 1.35 larger than the average core exposures used in Figure 4.

As stated previously, control of the reactor during the HPD core experiment will be accomplished by the addition of boron to the moderator. Calculations indicate that seven HPD fuel clusters are required for initial criticality with no boron in the moderator. As more clusters are loaded, it will be necessary to add boron to compensate for the excess reactivity. The ultimate moderator content will require 130 ppm natural boron for the 55 fuel elements, assuming that no fission products have yet formed and that all components of the core are at room temperature.

The amount of natural boron ($19.6\% \text{B}^{10}$) required to reduce the values of k_{eff} shown in Figure 4 to unity at each time step is shown in Figure 5. The units of boron concentration, ppm, are grams of natural boron per 10^6 grams of D_2O . The calculation did not account for the change in reflector savings of the top and bottom reflectors as boron is added. This effect would tend to reduce the total boron requirement slightly. The total reactivity worth of the axial reflectors, with no boron, is about 5%. If the boron addition effectively removed the top and bottom reflectors completely (a gross overestimate), the boron concentrations from Figure 5 would be reduced by about 20%.

FIGURE 5

Boron Requirements in Moderator for $k = 1.0$
55 Elements, 70 MW

B. Isotopic Composition of Fuel

Considerable variation occurs in isotopic composition of the fuel comprising the HPD core (Figure 6) during the expected lifetime of the core. The concentration of Pu^{239} , for example, changes by nearly a factor of 2. This will provide a stringent test of calculational models.

The nuclei densities (Figure 6) are given as a function of exposure only, because the time at which a given exposure is attained depends on

FIGURE 6

Isotopic Composition of Fuel

(Abbreviations L. S. and R. S. refer to scales on left and right, respectively.)

the position of the given fuel element within the core. To relate radial positions in the reactor to corresponding exposures, the ratio of exposure in various radial regions to average exposure in the core is shown in Figure 7. Thus, the exposure at a given radial position at time t will be the ordinate value from Figure 5 times the average core exposure at time t from either Figure 4 or Figure 6. The ratios given in Figure 7 are averages over the life of the core; and at any specific time, the actual ratio may differ from the average by as much as 3%.

FIGURE 7

Radial Exposure Relative to Core Average
HPD Core, PRTR

C. Thermal Flux Distributions

The thermal neutron flux in a unit cell of the HPD core is depressed sharply by the strongly absorbing fuel clusters (Figure 8). The flux shapes were calculated using THERMOS as described earlier. The lower curve (Figure 8) corresponds to the initial flux shape expected in any given lattice cell of the 55 element core with sufficient natural boron (130 ppm) in the moderator to control the reactor. As the core is heated to operating temperatures and power operation proceeds, the boron requirements will continually decrease and the flux shape will approach, roughly, the upper curve. This is not exact since the change in nuclear cross sections of the fuel with burnup has been neglected. This effect will cause the ratio of average flux in the moderator to average flux in the fuel to be somewhat higher than would be deduced from the upper curve of Figure 8.

FIGURE 8

HPD Core, Intracell Thermal Neutron Flux

The relative fission rate and thermal neutron flux show considerable curvature radially across the reactor. The spatial distributions (Figure 9) are shown for the initial loading with 130 ppm boron in the moderator and all core components at room temperature. During power operation, the shapes will tend to flatten such that, at end of core life, the peak-to-average ratio of the power density will be roughly 1.3 instead of the initial value of 1.6.

FIGURE 9
Radial Power Density and Thermal Flux
HPD Core, 130 ppm Natural Boron

D. Fuel Temperature Coefficient

The negative temperature coefficient of the fuel, the principle inherent shutdown mechanism in the event of a power excursion in the PRTR, is due to Doppler broadening of U^{238} and Pu^{240} resonance peaks. This coefficient, calculated for a temperature change from 20 to 980 °C is

$$\frac{1}{k} \frac{\Delta k}{\Delta T} = -1.68 \times 10^{-5} / ^\circ C,$$

and is assumed independent of boron concentration in the moderator. The major contribution to this coefficient is due primarily to U^{238} ; the contribution from Pu^{240} is small because the Pu^{240} concentration in the fuel is small compared to the U^{238} concentration.

E. Coolant Properties

The coolant temperature coefficient for a temperature change from 20 to 265 °C is calculated to be

$$\frac{1}{k} \frac{\Delta k}{\Delta T} = -1.21 \times 10^{-5} / ^\circ\text{C}$$

and is also assumed independent of boron concentration in the moderator. This coefficient includes the effects due to the decrease in coolant density and change in moderating properties of the coolant as the temperature is increased.

The effect of total removal of coolant is calculated⁽¹⁸⁾ to be -5.3 mk for the core poisoned with 81 ppm natural boron and -8.1 mk for the moderator with no boron. By linear extrapolation, the coefficient is expected to be negative for all boron concentrations anticipated during the HPD experiment.

F. Moderator Properties

In the absence of boron, the moderator temperature coefficient is primarily due to a change in moderator density. A decrease in moderator density accompanied by a rise in temperature causes an increase in the neutron leakage rate and thus reduces k_{eff} . An additional negative effect, caused by an increase in average neutron energy as the moderator is heated, occurs in the HPD core because the ratio of neutron captures to fissions increases with energy.

If a significant quantity of boron is present in the moderator, a decrease in density decreases the amount of boron poison; and thus, the moderator temperature coefficient can become positive for sufficiently large boron concentrations. In addition, this tendency for the coefficient to become positive is enhanced by the fact that the increase in average energy of the neutrons results in an increase in the ratio of neutron absorptions in the fuel to those absorptions in the moderator.

The moderator temperature coefficients for various boron concentrations are shown in Figure 10 where the coefficient ranges from $-1.16 \times 10^{-4}/^{\circ}\text{C}$ at zero boron concentration to $+1.18 \times 10^{-4}/^{\circ}\text{C}$ at a 140 ppm boron concentration. The coefficient passes through zero at about 45 ppm boron concentration. These coefficients were calculated for a change from 20°C to an assumed operating temperature of 62°C .

FIGURE 10

Moderator Temperature Coefficient Versus Boron Concentration
HPD Core

The reactivity changes accompanying a change in moderator (and associated boron) density alone are shown in Figure 11. The calculations leading to these curves were performed by assuming all cell components

at operating temperatures. Again it is seen that the effect becomes positive for sufficiently large boron concentrations.

FIGURE 11

Moderator Void Coefficients
2 wt% Mixed Oxide, 55 Element, PRTR Core

G. Kinetics Parameters

Effective values for the prompt neutron lifetime were calculated^(19, 20) for 1) the core with 140 ppm natural boron in the moderator and with no fission products and 2) the core at end of life with no born in the moderator. Both cases were calculated for all core components at operating temperatures. Radial flux and adjoint weighting were included in the calculations. The prompt neutron lifetime for the two cases is 1.57×10^{-4} sec and 3.46×10^{-4} sec, respectively. The effect of burnup and corresponding removal of boron thus increases the prompt lifetime by more than a factor of 2.

Effective values for the delayed neutron fractions are given in Table VI. The total effective delayed fraction, β_{eff} , remains nearly

constant over the life of the core because of the large fraction of fissions occurring in plutonium throughout core life.

TABLE VI
DELAYED NEUTRON FRACTIONS
HPD CORE - PRTR

<u>Group (i)</u>	<u>λ_i (sec⁻¹)</u>	<u>β_i</u> <u>140 ppm Boron</u>
1	0.0128	0.978×10^{-4}
2	0.0301	7.78×10^{-4}
3	0.124	6.06×10^{-4}
4	0.325	10.59×10^{-4}
5	1.12	3.18×10^{-4}
6	2.69	1.37×10^{-4}
7 (Photo-neutrons)	0.0016	0.542×10^{-4}
$\beta_{\text{eff}} = \sum_{i=1}^7 \beta_i = 1\beta =$		0.00305

H. Mechanical Shims

Although mechanical shims will not be used in the PRTR during power operation with the HPD core, there could be economic advantage to use them in controlling short-term reactivity changes rather than adjusting the boron concentration. For this reason, the reactivity worth of Inconel shim rods⁽¹⁾ was calculated⁽¹⁹⁾ for 1) 140 ppm boron-no fission products and 2) end of life. The results are given in Table VII. Values of total shim worth have not been corrected for rod shadowing effects and, therefore, are probably higher than would actually be obtained.

TABLE VII
SINGLE MECHANICAL SHIM WORTH

<u>Location</u>	<u>140 ppm-0 Exposure, mk</u>	<u>End of Life, mk</u>
Inner Ring	5.3	7.2
Central Ring	1.7	6.8
Outer Ring	0.8	5.5
Total - 18 shims	47	68.4

I. Whole Core Properties

The results described up to this point pertain to the 54 clusters constituting the basic core. It was assumed that the central fuel position, 1550, would contain a standard HPD fuel element so that the calculations were made for a 55 element core. During power operation, the FERTF will be located in the center and irradiations will be performed at high specific power with specially designed $\text{UO}_2\text{-PuO}_2$ fuel utilizing H_2O coolant. In addition, the reactor will contain 12 buffer elements and several test elements on the outer periphery of the core as described previously (Section III).

A calculation was made to determine the extent to which the effective neutron multiplication and neutron lifetime of the full core differ from the 55 element core. The calculation was made assuming an H_2O reflector, H_2O coolant in the FERTF and no fuel element, and 84 clusters of HPD fuel. The effective multiplication for this loading is $k_{\text{eff}} = 1.260$ or just the same as for the 55 element core. Thus the increased reactivity due to the larger core size is just compensated by the loss in reactivity associated with removal of the fuel element from the central position, replacing the D_2O with H_2O in the center process tube, and replacing the D_2O reflector with H_2O . The magnitude of each effect separately is 1 to 2%. Furthermore, the prompt neutron lifetime for the full core is only 10% smaller than the lifetime in the

55 element core (with D_2O reflector), where the moderator, in both cases, is borated such that $k_{eff} = 1.0$. Thus, it is concluded that the properties of the full core are not significantly different from the properties of the basic core.

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