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SAXTON PLUTONIUM PROGRAM

NUCLEAR DESIGN OF THE SAXTON

PARTIAL PLUTONIUM CORE

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

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ABSTRACT

The purpose of the Saxton Plutonium Program is to develop information concerning the use of plutonium enriched fuel in pressurized water reactors. The nuclear design requirements were to establish a fuel enrichment and core configuration that would provide a design life of 8,250 MWD and an average irradiation in the plutonium fuel of 10,000 MWD/tonne. It was necessary to meet these objectives within a linear power limitation of 16 kw/ft.

The nuclear design work reported concerns three areas. Initially, parameter studies were carried out to determine the performance characteristics of the Saxton reactor containing a partial core of plutonium. Later, more specific nuclear design calculations were carried out to set the enrichment specification and determine the available performance for a core containing nine central plutonium fuel assemblies with twelve surrounding uranium assemblies. Finally, a series of critical experiments using the design fuel and a supporting program of analysis were used to confirm the specified nuclear design.

The Saxton Core II nuclear design is expected to have a lifetime capability of 9,300 MWD and achieve an average burnup in the plutonium fuel of 13,600 MWD/tonne. The supporting critical experiment program provided significant beginning-of-life physics information about the fuel and the basis for improving the correlation of analysis with experiment for plutonium fueled systems.

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1.0 INTRODUCTION

1.1 General Description of the Saxton Plutonium Program

1.1.1 Purpose

The purpose of the Saxton Plutonium Program is to develop information concerning the use of plutonium enriched fuel in pressurized water reactor systems through the design, fabrication, and operation of a partial core of $\text{PuO}_2\text{-UO}_2$ fuel in the Saxton reactor. The in-pile performance of the fuel will be evaluated and a post-irradiation examination of fuel samples will be made. A basic program objective is to achieve as high a burnup in the plutonium fuel as possible during a two-year irradiation period.

1.1.2 Program Sequence

The program sequence can be divided into four phases. During the nuclear design phase, the requirement was to develop the specifications for the partial loading of the mixed-oxide ($\text{PuO}_2\text{-UO}_2$) fuel assemblies to be used in Saxton Core II. It was necessary to determine a plutonium loading that would provide the desired lifetime within the power limitations set by the plant, thermal, and hydraulic design. Goals of 10,000 MWD/tonne average burnup in the plutonium fuel with a maximum burnup of 20,000 MWD/tonne were established.

The second phase consisted of a series of design-oriented critical experiments conducted at the Westinghouse Reactor Evaluation Center (WREC) using the same fuel rods as those to be used in the irradiation test. The purpose

of these experiments was to verify the nuclear design of the unirradiated fuel before installation in the Saxton reactor.

The third phase concerns the operation of the partial plutonium core in the Saxton reactor. During this phase, a series of zero and low power physics measurements will be made to verify reactivity, control rod and boron worth, flux and power distributions, and temperature, pressure, and power coefficients. Similar data will be obtained during periodic shut-downs to determine the changes in characteristics with irradiation.

In the fourth phase, a post-irradiation examination of fuel samples will be made and compared to analytic predictions.

1.1.3 Scope of the Report

This report describes the work carried out in the first two phases, the nuclear fuel design and the planning and analysis of the WREC critical experiments. A complete description of the results of each phase is included in the main body of the report. Additional detail in each area is included in the Appendices. Sufficient critical experiment information is included to provide comparisons with the analysis and to permit further study should it later prove desirable. Also included is a brief summary of the post-critical experiment analysis of the design core carried out in preparation for the zero power testing of Saxton Core II.

1.2 General Description of the Saxton Reactor

The Saxton Experimental Reactor^{/1} was designed to investigate areas of interest in the development of pressurized water reactors. The experimental program has included investigations in fuel development and chemical shim reactivity control.

A brief summary of some of the more important design and performance parameters of the original Saxton core, Saxton Core I, is contained in Table 1.1. (A more detailed summary is included in Appendix G.)

TABLE 1.1

SAXTON Core I Description

Core

Number of UO ₂ Fuel Assemblies	21
Equivalent Core Diameter, inches	28.07
Core Length, inches	36.6
Number of Control Rods	6

Fuel and Fuel Rod

UO ₂ Enrichment, % U-235	5.7
Fuel Tube Diameter, inches	0.391
Clad Material	304 SS
Clad Thickness, inches	0.015
Fuel Rod Pitch, inches	0.580

Performance

Thermal Power, MW	20
Moderator Temperature, °F	530
Operating Pressure, psi	2000

Saxton Core I was operated for a total of 8630 MWD and accumulated an average burnup of $\approx 9/00$ MWD/tonne. However, the projected burnup

capability of Saxton Core I was 12,800 MWD/tonne.

Since Saxton was designed as an experimental facility, it has an extensive in-core instrumentation system. The system is capable of measuring outlet temperature, flow rate, and pressure drop. In addition, the activation of manganese in carbon steel wires in conjunction with analysis is used to determine nuclear power distributions throughout the core.

Figure 1.1 is a diagram of the core showing the location and type of in-core instrumentation. There are also five subassemblies available for special enrichment rods or instrumentation. These subassemblies may be examined without removing the reactor head since they can be removed through the fuel element ports.

The second core to be operated in the Saxton reactor, designated Saxton Core II, contains a partial core loading of plutonium fuel. Because of the experimental nature of the Saxton reactor, the operation of this core offers a unique opportunity to investigate the performance of plutonium fuel in a pressurized water reactor system. The use of the same fuel rods in a critical experiment program conducted in advance of the irradiation test has provided significant beginning-of-life physics information about the fuel.

1.3 Initial Ground Rules

At the start of the nuclear design of the partial plutonium core, the following ground rules were established:

- 1) The nuclear parameters affecting the operational safety margin on the Saxton reactor should not be significantly changed by the introduction of plutonium fuel.

Figure 1.1

SAXTON IN-CORE INSTRUMENTATION

- PITOT TUBE AND THERMOCOUPLE
- ⊙ THERMOCOUPLE
- ⊕ FLUX WIRE THIMBLE

- 2) Control of the Saxton reactor by either control rods alone or chemical shim alone, or by any combination of the two should not be restricted by the presence of plutonium bearing fuel in the reactor.
- 3) The Saxton Core II, comprised in part of plutonium fuel, should be capable of a minimum of 8250 megawatt days (thermal) of operation and should have a useful life not greatly in excess of 10,000 megawatt days.
- 4) Specific power level limitations imposed on the Saxton reactor should not be exceeded as a result of introduction of plutonium bearing fuel.
- 5) The fraction of total reactor power produced in a mixed oxide ($\text{PuO}_2\text{-UO}_2$) fuel should be maximized.
- 6) The presence of plutonium bearing fuel in the Saxton reactor should not jeopardize the successful execution of other experiments scheduled to be carried out concurrently with the Saxton Plutonium Program.

1.4 Analysis Methods and Cross Sections

1.4.1 Analysis Methods

The nuclear analysis in each phase was carried out using standard methods that have been verified in the design and operational analysis of the Saxton reactor and other pressurized water reactors. Details concerning the computer codes that were used are contained in the references, however,

to clarify the subsequent discussion a brief description of the more important codes is included in the following paragraphs.

LEOPARD^{/2}

The LEOPARD code determines fast and thermal spectra based on a modified MUFT^{/3} SOFOCATE^{/4} model. The thermal spectrum is the same as that given by a Wigner-Wilkins SOFOCATE calculation except for the treatment of disadvantage factors. Disadvantage factors are determined using a modified form of the Amouyal-Benoist calculation at 172 energy levels from zero to 0.625 ev. LEOPARD computes a fast spectrum based on a consistent B-1 MUFT IV calculation. The resonance integral for U-238 is determined from "a mixed oxide" correlation that is in good agreement with Hellstrand's measurements for uranium metal and uranium dioxide at any temperature. An iterative procedure in the code is followed to determine a self-shielding or "L" factor for U-238 that is consistent with the calculated resonance integral. An "L" factor of 1.0 is used for all other elements except Pu-240. The "L" factor for Pu-240 is calculated from an equation that is also based on a U-238 correlation.

LASER^{/5}

The LASER code determines fast and thermal spectra based on a modified MUFT-THERMOS^{/6} model. The thermal calculation consists of the THERMOS code, a cell transport theory code in space and energy, with an energy range from zero to 1.855 ev. Thus, the Pu-240 resonance at ≈ 1.05 ev is included in the thermal range. The thermal energy mesh includes 35 energy

points spaced non-uniformly. The mesh contains fine energy intervals under the 0.3 ev Pu-239 and 1.05 ev Pu-240 resonances and thus eliminates the necessity of using weighted effective cross sections.

Both LASER and LEOPARD use a consistent B-1 MUFT IV calculation in the fast energy group. The difference lies in the energy level at which the fast group ends.

PDQ-3⁷

The PDQ-3 code solves the few-group, time independent, neutron diffusion equations in X-Y geometry. Group constants were obtained from LEOPARD.

TURBO*⁸

The TURBO* code solves the few-group, two dimensional (X-Y geometry) neutron diffusion equations in combination with a pointwise burnup calculation to determine reactivity-lifetime relationships. The microscopic library generated in LEOPARD is used to determine time dependent group constants for use in TURBO*.

LUX

The LUX code is a modification of the CANDLE⁹ one-dimensional few-group depletion code and is used to determine reactivity as a function of lifetime. The LEOPARD microscopic library is used.

1.4.2 Cross Sections

During the course of the work, the following four different thermal cross section sets were used:

- 1) Leonard^{/10} cross-sections normally contained in the LEOPARD II production code.
- 2) Cross sections reported by Westcott^{/11} at the 1964 Geneva Conference.
- 3) Cross sections from Winfrith^{/12}.
- 4) Cross sections compiled by Sher^{/13} in 1965.

The energy dependence of the Leonard, Westcott, and Sher thermal cross sections was the same. The absorption and fission cross sections were normalized to the reported 2200 m/sec values. The thermal cross section data from Winfrith which has a different shape as a function of energy was obtained from Reference 12.

2.0 SUMMARY AND CONCLUSIONS

2.1 Summary

2.1.1 Nuclear Design Chronology

To determine whether a nuclear fuel design could be developed that would provide the desired performance within the constraints imposed by the initial ground rules, parameter studies were carried out to evaluate the effect of plutonium enrichment and core location on lifetime, installed reactivity, power peaking, and power sharing in the two-fueled regions.

At that time there was considerable uncertainty as to the accuracy of existing methods of analysis and cross sections when applied to a plutonium core. Therefore, calculations were carried out in parallel with the design to compare the analysis with six available mixed-oxide (PuO_2 - UO_2) experiments conducted at Hanford. An average discrepancy of $\approx 2.6\%$ $\Delta k/k$ was found and an allowance was included in the subsequent design calculations to compensate for this discrepancy.

Based on the initial parameter studies, a reference configuration was established in which nine plutonium fuel assemblies were installed in the center of the core with twelve conventional uranium fuel assemblies installed on the periphery. The range of the possible plutonium enrichments was narrowed to permit a more detailed investigation of burnup and power distribution effects.

One-dimensional and two-dimensional burnup calculations were then carried out to set the plutonium enrichment specifications. These

studies showed that a plutonium enrichment that provided sufficient lifetime when installed in the center of the core resulted in an increase in the radial nuclear hot channel factor over that of the conventional Saxton uranium loading. The decision to install the plutonium fuel assemblies in the center of the core was based on the desire to maximize the irradiation exposure of the mixed-oxide ($\text{PuO}_2\text{-UO}_2$) fuel. Thus, the enrichment specification required a compromise between the irradiation, power peaking, and lifetime objectives.

In view of the large uncertainty in the calculation of installed reactivity in a core containing plutonium and because of the importance of power peaking and power sharing effects upon the performance of the Saxton core, the WREC critical program was planned to provide experimental information in each of these areas.

The results from the critical experiment program confirmed the validity of the nuclear fuel design. In general, the associated analysis and experiment were in good agreement. The small corrections indicated by the experimental results were factored into revised performance predictions for the design core. In addition, criticality studies carried out during the critical experiments program led to a significant improvement in the correlation of analysis with experiment. As a result, revised cross sections were adopted for use in the analysis required in following the operating history of Saxton Core II.

2.1.2 Core Description - Saxton Core II

The nuclear design sequence described in the previous paragraphs led to the core configuration shown in Figure 2.1. As shown in this figure,

Figure 2.1: Saxton Core II Fuel Configuration

the Saxton Core II fuel loading is made up of nine central mixed-oxide ($\text{PuO}_2\text{-UO}_2$) fuel assemblies and 12 outer UO_2 fuel assemblies. The mixed-oxide contains 6.6 w/o PuO_2 in natural UO_2 . The UO_2 assemblies are enriched to 5.7 w/o U-235. Of the nine plutonium assemblies, two contain vibratory compacted fuel and are installed on the flats in positions D-2 and D-4. The remaining seven assemblies contain pelletized fuel. The plutonium fuel rods are clad with Zircaloy-4 with the exception of some 30 fuel rods which are clad with 304 stainless steel. These special stainless steel rods are distributed in the central plutonium region to meet specific irradiation objectives. All of the plutonium assemblies contain inconel grids.

The 12 UO_2 assemblies consist of five assemblies with the original Core I design (304 SS grids). Of these, one assembly (E-1) is a special 51 rod hollow UO_2 assembly designed to accommodate a supercritical test loop. This assembly accumulated ≈ 2100 MWD/tonne burnup during Saxton Core I operation. At the start of the plutonium program, it will contain a stainless steel plug. The seven remaining UO_2 assemblies are of the new design (inconel grids). These assemblies are located at the corners of the core to reduce the reactivity loss associated with the higher neutron cross section and the increased material volume of the new grid structure.

The special L-shaped assemblies (UO_2 fueled) used in Saxton Core I will also be used in Saxton Core II. In addition, the original fuel bearing control rod followers (UO_2) will also be used in Saxton Core II.

However, to reduce the reactivity loss due to fuel burnup, control rods were interchanged (rod 1 for rod 2 and rod 5 for rod 6) placing the fuel followers with the lowest burnup in the center of the core.

Four of the fuel assemblies, one plutonium and three uranium, contain removable 3x3 subassemblies.

Specifications for the fuel and fuel rods are included in Appendix A.

2.1.3 Performance Capability and Limitations

The Saxton Plutonium Core is expected to have a lifetime capability of 9300 MWD or 9500 hours at 23.5 MWt which is approximately 12% higher than the design requirement of 8250 MWD.

For a design life of 8250 MWD, an average burnup in the plutonium fuel of 13,600 MWD/tonne should be achieved. The burnup in the maximum rod in the plutonium region is expected to be $\approx 20,000$ MWD/tonne. These burnup values equal or exceed the original design objectives. Since the core has a small increase in design life over that required, slightly higher burnups are possible if the core is allowed to run to the end of life.

To insure meeting the lifetime requirements, it was necessary to increase the nuclear hot channel factor over that of Saxton Core I.

Consequently, it may be necessary to restrict the reactor power level initially to ≈ 22 MWt to avoid exceeding a linear power limit of 16 kw/ft. However, this limitation, if required, would be short lived due to the rapid reduction of the power peaks with burnups. It is expected that

the core could be operated at the design level of 23.5 MWt after an equivalent three full power months operation at ≈ 22 MWt. The need for a limit on the power will be determined during the initial test program for Saxton Core II. The hot, clean, installed reactivity at power (23.5 MWt) is expected to be $11.2\% \Delta k/k$. A total rod worth of $17.6\% \Delta k/k$ is anticipated. At the start of life, the following boron requirement is expected:

Hot, clean, zero power	2230 ppm
Cold, clean, zero power	2700 ppm

2.1.4 Scope of the WREC Critical Program¹⁴ and Supporting Analysis

An outline of the measurements program carried out at the WREC is included in Table 2.1. The following statements briefly summarize the extent of the experimental work:

- 1) A series of preliminary experiments using the UO_2 fuel was carried out. Critical buckling measurements were made in three different lattice pitches. The effect on reactivity and local power peaking resulting from perturbations such as water slots and control rods was investigated. The moderator temperature coefficient was measured to $70^\circ C$. Foil traverses were made with U-238 and Au foils. Pulsed neutron experiments were conducted.
- 2) For the single region PuO_2 - UO_2 cores, critical buckling measurements were made in five different lattices. The effect of perturbations due to water slots and control rods was investigated. Foil traverses using U-238, Au, and Dy foils were

Table 2.1

Single-Region Experiments PuO ₂ -UO ₂ Fuel		Multi-Region Experiments ³ PuO ₂ -UO ₂ Inside, UO ₂ Outside
<u>0.56-Inch Lattice</u>	<u>0.795-Inch Lattice</u>	<u>Clean</u>
a. Cylindrical core-critical rods b. Buckling by fuel rod scans (19 x 19 core) 1 c. Power map, foil traverses-U238, Au, Dy (19 x 19 core) d. Water slot in center-reactivity, power map 2 e. Aluminum slab in center-reactivity, power map f. Five control rods in center-reactivity, power map (21 x 21 core) g. Moderator temperature coefficient h. Vipac vs pelletized fuel-reactivity, power map i. 3 x 3 UO ₂ insert-reactivity, power map, flux-Dy j. Fuel rod circumferential flux-Dy wire k. Pulsed neutron experiments-clean and borated l. Boron worth to 50 ppm (19 x 19 core) m. Boron worth to 337 ppm (21 x 21 core) n. Borated core buckling (21 x 21 core) o. Noise analysis (19 x 19 core)	a. Cylindrical core-critical rods b. Buckling by fuel rod scans (12 x 12 core) c. Power map <u>0.52-Inch Lattice</u> a. Cylindrical core-critical rods b. Buckling by fuel rod scans (22 x 23 core) c. Pulse neutron experiments (23 x 23 core) <u>0.735-Inch Lattice</u> a. Cylindrical core-critical rods b. Buckling by fuel rod scans (13 x 13 core) <u>1.04-Inch Lattice</u> 4 a. Cylindrical core-critical rods b. Buckling by fuel rod scans (11 x 11 core)	a. Fuel substitution in steps to the reference core (11 x 11 PuO ₂ -UO ₂ , 19 x 19 core)-reactivity b. Power map, foil traverses-U238, Dy c. Aluminum slab at boundary-reactivity, power map d. Water slot-reactivity, 3 positions (11 x 11 PuO ₂ -UO ₂ , 21 x 21 core) e. Five control rods-reactivity, 5 positions (11 x 11 PuO ₂ -UO ₂ , 21 x 21 core) f. Five control rods at fuel interface-power map g. Moderator temperature coefficient h. Pulse neutron experiments (11 x 11 PuO ₂ -UO ₂ in 19 x 19 and 21 x 21 cores) i. Noise analysis <u>Borated Core</u> a. Fuel and boron addition in steps to the reference core (19 x 19 PuO ₂ -UO ₂ , 27 x 27 overall, 1453 ppm boron) - reactivity b. Power map, foil traverses-U238, Dy c. Water slot experiment at fuel interface-reactivity, power map d. Aluminum slab experiment at fuel interface-reactivity, power map e. L-shaped UO ₂ inserts in PuO ₂ -UO ₂ region simulating Saxton design-Manganese wire activation at design flux wire locations and core power map (1425 ppm boron) f. 3 x 3 UO ₂ insert in PuO ₂ -UO ₂ region (1425 ppm boron) reactivity, power map
<u>UO₂ Fuel</u>	<u>0.795-Inch Lattice</u>	<u>UO₂ Inside, PuO₂-UO₂ Outside</u>
<u>0.56-Inch Lattice</u>	<u>0.795-Inch Lattice</u>	<u>Clean</u>
a. Cylindrical core-critical rods b. Buckling by fuel rod scans (19 x 19 core) c. Power map, foil traverses-U238, Au (19 x 19 core) d. Water slot in center-reactivity, power map e. Aluminum slab in center-reactivity, power map f. Five control rods in center-reactivity, power map (21 x 21 core) g. Moderator temperature coefficient h. 3 x 3 PuO ₂ -UO ₂ insert-reactivity, power map i. 7 x 7 PuO ₂ -UO ₂ insert-reactivity j. Fuel rod circumferential flux-Dy wire k. Pulsed neutron experiments	a. Cylindrical core-critical rods b. Buckling by fuel rod scans (13 x 14 core) c. Power map <u>0.52-Inch Lattice</u> 4 a. Cylindrical core-critical rods b. Buckling by fuel rod scans (cylindrical core)	a. Fuel substitution in steps to inverted reference core (11 x 11 UO ₂ inside, 19 x 19 overall) b. Power map, foil traverses-U238, Dy c. Water slot-reactivity, 3 positions (11 x 11 UO ₂ , 21 x 21 core) d. Five control rods, 3 positions (11 x 11 UO ₂ , 21 x 21 core) e. Pulse neutron experiments (11 x 11 UO ₂ , 21 x 21 core) <u>Borated Core</u> a. Quarter-core step change in fuel position from 2-region core to inverted 2-region core-reactivity b. Full core change to inverted reference configuration (19 x 19 UO ₂ , 27 x 27 overall, 1252 ppm boron) c. Power map, foil traverses-U238, Dy
1. U-238 foils counted for fission activity and Np-239 decay 2. Power measurements by fuel rod scans 3. Carried out in 0.56-inch lattice 4. Not part of the Saxton Plutonium Program. Included to complete the summary of the buckling data available		

made. Moderator temperature coefficient measurements to 58°C were carried out. Differences in power and reactivity between vibratory-compacted and pelletized fuel and between different clad materials were evaluated. Boron worth in the moderator was measured. Pulsed neutron and noise analysis experiments were conducted.

3) In the multi-region cores composed of separate regions of each fuel type, a series of experiments to determine the power sharing between the two fuels was carried out. The experiments included configurations in which the $\text{PuO}_2\text{-UO}_2$ fuel was placed in an inner region with UO_2 in the outer region, as in the Saxton reference design, and also in configurations with an inverted geometry in which the UO_2 was installed in an inner region with the $\text{PuO}_2\text{-UO}_2$ fuel on the outside. Heat-rate and foil irradiation experiments were conducted to determine the relationship of measured gamma activity after shutdown to relative power in the two fuels. The effects of perturbations due to water slots and control rods were investigated. Pulsed neutron techniques and noise analysis experiments were conducted. Moderator temperature coefficient measurements were made. Both clean and borated configurations were used.

A program of analysis was carried out in support of the WREC critical experiments. Standard methods of analysis were used in a full investigation of the experiments outlined in Table 2.1. With few exceptions,

the agreement between analysis and experiment was good. The following statements briefly summarize some of the more significant results:

- 1) The analysis carried out in advance of the critical program to determine the fuel rod and boron concentration requirements was in excellent agreement with the subsequent experimental results. For the 0.56-inch $\text{PuO}_2\text{-UO}_2$ critical lattice, 346 fuel rods were required for criticality as compared to an expected requirement of 356 fuel rods. This good agreement meant the planned experimental program could be carried out without major revisions. The as-planned program and the pre-analysis results are reported in Appendix D.
- 2) Buckling measurements made over a range of H/Pu ratios from 230 to 1077 provided the basis for a detailed criticality study. The results of the study led to the adoption of methods and cross sections whereby the discrepancy between analysis and experiment was reduced to approximately 1% over the entire H/Pu range.
- 3) Comparison of measured and calculated power peaking effects in both single and multi-region cores show that the analysis methods used in the design overestimate power peaking in both UO_2 and $\text{PuO}_2\text{-UO}_2$ fuels. As a result, power peaking calculations for the design core are conservative. Therefore, it may be possible to increase the initial reactor power level to more than 22 MWt if confirmed by the power tests in Saxton Core II.

- 4) Power sharing between the $\text{PuO}_2\text{-UO}_2/\text{UO}_2$ fueled regions in multi-region cores is important in establishing the maximum core power level. Comparisons of calculated and measured power distributions were made in the multi-region experiments of Table 2.1. Two methods were used to translate gamma activity to rod power. The analytic and experimental power distributions agreed within $\approx 1\%$ using a foil method and within $\approx 5\%$ using a thermal method. With either method, there is no adverse effect on the power capability of Saxton Core II. When both power peaking and power sharing are considered, the calculated design core local power peak is expected to be between 0 and 5% higher than will actually occur during operation.
- 5) The calculated boron worth in both single region and multi-region cores was in excellent agreement with experimental results. The measured control rod worths were also in good agreement with analysis. The largest discrepancy occurred in evaluating rod worth in the plutonium regions. For these experiments, the analysis underestimated rod worth.

2.1.5 Nuclear Characteristics Summary

The use of plutonium fuel assemblies in Saxton Core II alters the nuclear characteristics of the system. The following statements briefly summarize the major reactivity and kinetic effects that occur with a partial plutonium core.

- a) The Saxton plutonium core will have a more negative moderator temperature coefficient than with a conventional uranium loading. This effect results in a shift in power distribution with temperature.
- b) The negative Doppler coefficient will be larger with $\text{PuO}_2\text{-UO}_2$ than with just UO_2 fuel.
- c) The part-plutonium core will have a larger positive pressure coefficient and a more negative void coefficient than the conventional uranium core.
- d) Boron concentration and control rod worths are decreased in the partial plutonium core.
- e) The delayed neutron fraction (β_{eff}) and prompt neutron lifetime when averaged over the core at the beginning of life will be smaller with a partial plutonium core than with a full loading of uranium fuel. They both remain constant throughout the core life because of the change in radial power distribution and the plutonium buildup in the outer uranium assemblies. However, there is a marked change in characteristics with temperature and rod position because of the shift in power distribution.

2.2 Conclusions

As a result of the analytic and experimental work carried out during the course of the nuclear design, the following conclusions were reached:

- (1) The nuclear design of the partial plutonium core meets the design requirements as to lifetime and fuel irradiation. An adequate margin is included to allow for reasonable deviations in the expected performance and still meet the design objectives.
- (2) The performance of a critical experiment using the design fuel served three purposes. First, it provided confirmation of the nuclear design. This confirmation was particularly important from the standpoint of reducing the large uncertainty as to the initial reactivity of the design core. Second, it provided the necessary physics information to serve as a basis for improving the correlation of analysis with experiment for plutonium fueled systems. Third, the experimental and analytical program identified a number of design considerations that are important in utilizing plutonium fuels in thermal reactors.
- (3) The good agreement between analysis and experiment obtained at the close of the critical experiments program using standard pressurized water reactor design methods provides assurance as to the adequacy of the methods in the determination of initial reactivity and beginning of life power distributions. However, data obtained during Saxton Core II operation will be required before any conclusions can be reached as to the adequacy of the analysis in predicting core life.

- (4) While good agreement between analysis and experiment was demonstrated for the low enrichment experiments conducted at Hanford and the high enrichment experiments conducted at the WREC, additional experiments at intermediate enrichments and with variations in the Pu-240 content are needed to insure satisfactory correlation can be obtained at enrichments throughout the possible range of interest.

3.0 Parameter Studies

3.1 Purpose

The parameter studies carried out at the start of the nuclear design effort had the following three basic objectives:

- 1) To determine the feasibility of achieving the desired performance within limitations set by the defined ground rules.
- 2) To establish tentative design specifications such as fuel enrichment and fuel assembly location in order to determine a reference configuration suitable for additional study.
- 3) To evaluate the adequacy of the available analytic methods and cross sections so that any deficiencies that were found could be factored into the final enrichment specification.

3.2 Analytic Results

The results of the parameter study to determine a reference configuration can be divided into two parts. The first concerns calculations carried out for a uniform loading throughout the core. The purpose was to determine the basic characteristics of the Saxton reactor when fueled with plutonium. The extent of these calculations and the methods used are summarized in Table 3.1.

As shown in this table, the calculations can be further separated into three groups. The calculations in the first group were made to determine

Table 3.1

Parameter Studies in a Uniformly Loaded Saxton Core

Group 1 - K_{eff} , Lifetime, Inverse Boron Worth, Relative Clad Worth, and Equilibrium Xenon Worth as a Function of Fuel Enrichment

Figure No.	Parameter Investigated	As a Function of (Range)	With Variations In	Remarks
3.1	1. K_{eff} (Hot*, Clean, 23.5 MWt) 2. Inverse Boron Worth (ppm/% ρ)	Plutonium** Atom Fraction (a/o) (0 - 20)	Boron Concentration a) 0 ppm b) 1000 ppm	LEOPARD Calculation
3.2	1. K_{eff} (Hot, Clean, 23.5, 1000 ppm B) 2. $\Delta\rho$ Zirconium vs. SS-304	Plutonium Atom Fraction (a/o) (0 - 20)	Clad Material and Thickness a) 15 mil Zr clad b) 23 mil Zr clad c) 15 mil SS-304 clad	LEOPARD Calculation
3.3	Lifetime at 23.5 MWt (Thousands of Hours)	Plutonium Atom Fraction (%) (0 - 15)	Clad Material and Thickness a) 23 mil Zr clad b) 15 mil SS-304 clad	LEOPARD Calculation
3.4	1. ΔK_{eff} %, 0 Hours to 100 Hours (At 23.5 MWt) 2. $\Delta\rho$ %, 0 Hours to 100 Hours (At 23.5 MWt)	Plutonium Atom Fraction (%) (0 - 20)	--	Equilibrium Xenon Worth - LEOPARD Calculation
3.5	Lifetime at 23.5 MWt (Thousands of Hours)	U-235 Enrichment (%) (Nat'l - 3.0)	Plutonium Atom Fraction (%) a) 3.98% Pu - 23 mil Zr clad b) 3.98% Pu - 15 mil SS-304 clad c) 5.04% Pu - 15 mil SS-304 clad	LEOPARD Calculation
3.6	K_{eff} (Hot, Clean, 23.5 MWt, 1000 ppm B)	U-235 Enrichment (%) (Nat'l - 3.0)	Plutonium Atom Fraction (%) a) 3.98% Pu - 15 mil SS-304 clad b) 3.98% Pu - 23 mil Zr-2 clad c) 5.04% Pu - 15 mil SS-304 clad	LEOPARD Calculation

Group 2 - Variation of Core Characteristics as a Function of Plutonium Atomic Fraction

Figure No.	Section	Parameter Investigated	With Variations In	Remarks
B-1	Appendix B	1. K_{eff} (Hot, Clean, 23.5 MWt) 2. Fast to Thermal Flux Ratio 3. Wigner-Wilkins $\frac{1}{\nu}$ Factor	Boron Concentration a) 0 ppm b) 1000 ppm	SS-304 Clad LEOPARD Calculations
B-2	Appendix B	1. "g" Factor 2. Macroscopic Fission Cross Section Thermal Weighted	--	SS-304 Clad LEOPARD Calculation

Group 3 - Variation of Core Characteristics as a Function of Lifetime

Figure No.	Section	Parameter Investigated	With Variations In	Remarks
B-3	Appendix B	1. Fast to Thermal Flux Ratio 2. Wigner-Wilkins $\frac{1}{\nu}$ Factor 3. Weighted Macroscopic Absorption Cross Section 4. Weighted Macroscopic Fission Cross Section	Boron Concentration with Time	Conventional Saxton Core 5.74 a/o U-235 CG-304 Clad LEOPARD Calculation
B-4	Appendix B	1. Fast to Thermal Flux Ratio 2. Wigner-Wilkins $\frac{1}{\nu}$ Factor 3. Weighted Macroscopic Absorption Cross Section 4. Weighted Macroscopic Fission Cross Section	Boron Concentration with Time	3.98 a/o Pu SS-304 Clad LEOPARD Calculation

* $T_{mod} = 530^{\circ}\text{F}$

** Pu Isotopic Composition, %

Pu-239 = 88.15

Pu-240 = 10.24

Pu-241 = 1.61

Pu-242 = -

the effect of variations in plutonium enrichment on such parameters as reactivity, boron worth, relative clad worth, lifetime, and equilibrium xenon requirements. Also included in this group were calculations to determine the effect on lifetime and k_{eff} for a variation in the U-235 content for two different plutonium enrichments. The objective was to determine if there were significant advantages in using enriched UO_2 in the mixed-oxide fuel ($\text{PuO}_2\text{-UO}_2$). The results of these calculations are summarized in Figures 3.1 through Figure 3.6.

The second group concerns the variation of core characteristics with plutonium enrichment. The characteristics include macroscopic cross sections, self-shielding factors (g factors), fast-to-thermal flux ratios, Wigner-Wilkins $\frac{1}{v}$ factors, and calculated k_{eff} . The third group contains similar parameters evaluated as a function of lifetime for two different plutonium enrichments. The results of this portion of the study are summarized in Figures B-1 through B-4 in Appendix B.

The second part of the parameter study concerns the calculations carried out in multi-region cores. The extent of these calculations and the methods used are summarized in Table 3.2. The analysis was carried out for three basic reference configurations. The number of mixed-oxide fuel assemblies assumed was based on estimates as to the amount of plutonium fuel to be made available. The configurations described on page 3-11 were used in the analysis.

Figure 3.1 - Effect of Boron on Reactivity in a Saxton Core
Uniformly Loaded with Uranium Enriched with Plutonium

Figure 3.2 - Relative Worths of Zirconium Clad in a Saxton Core
Uniformly Loaded with Natural Uranium Enriched with Plutonium

Figure 3.3 - Relative Lifetimes of Saxton Cores Uniformly Loaded with Natural Uranium Fuel Enriched with Plutonium

Figure 3.4 - Apparent Worths of Equilibrium Xenon in a Saxton Core Uniformly Loaded with Natural Uranium Enriched with Plutonium

Figure 3.5 - Effective Multiplication in a Saxton Core Uniformly Loaded with Uranium Enriched in Both U-235 and Plutonium

Figure 3.6 - Relative Lifetimes of Saxton Cores Uniformly Loaded with Fuel Enriched in Both Plutonium and Uranium-235

Table 3.2

Parameter Studies in Multi-Region Saxton Cores
with Three Different Fuel Arrangements

Group 1 - Investigation of Parameters as a Function of Plutonium Atomic Fraction (a/o)

Figure No.	Parameter Investigated
3.7	1. E_{eff} (Hot, Clean, 23.5 MWT) 2. Lifetime (Hours at 23.5 MWT) 3. Initial Boron Concentration (ppm)
3.8	1. Inverse Boron Worth (ppm/% ρ) 2. Equilibrium Xenon Defect at 23.5 MWT (% ρ)
3.9	Relative Assembly Power Level
3.10	1. Peak to Average Power Ratio 2. Fraction of Power in Plutonium Region
3.11	Delayed Neutron Fraction

Group 2 - Investigation of Parameters as a Function of Lifetime (Thousands of Hours at 23.5 MWT)

Figure No.	Parameter Investigated	With Variations In
3.12	Fraction of Power in Plutonium Region 1. Corner Assemblies 2. Center Assemblies	Plutonium Enrichment a) 3.98 a/o b) 5.04 a/o c) 7.96 a/o d) 15.0 a/o (corners)
3.13	Peak to Average Power in Plutonium	Plutonium Enrichment a) 3.98 a/o b) 5.04 a/o c) 7.96 a/o d) 15.0 a/o (corners)
3.14	Peak to Average Power in Uranium	Plutonium Enrichment a) 3.98 a/o b) 5.04 a/o c) 7.96 a/o d) 15.0 a/o (corners)

LUX Calculations

Configuration Descriptions

- a) 8 $\text{PuO}_2\text{-UO}_2$ assemblies in center, 13 UO_2 assemblies in outer region.
- b) 8 $\text{PuO}_2\text{-UO}_2$ assemblies in corner locations, 13 UO_2 assemblies in center and on flats.
- c) 1 $\text{PuO}_2\text{-UO}_2$ assembly in center, 8 $\text{PuO}_2\text{-UO}_2$ assemblies in each corner location, 12 UO_2 assemblies in remaining locations.

- a) 8 $\text{PuO}_2\text{-UO}_2$ assemblies in the center of the core with 13 conventional Saxton UO_2 assemblies in the outer region.
- b) 8 $\text{PuO}_2\text{-UO}_2$ assemblies in the corner locations with 13 conventional Saxton UO_2 assemblies installed in the center of the core and in the middle of the peripheral flats.
- c) 1 $\text{PuO}_2\text{-UO}_2$ assembly in the center of the core, 8 $\text{PuO}_2\text{-UO}_2$ assemblies in the corner locations with 12 conventional Saxton UO_2 assemblies installed in the remaining core locations.

The various parameters were investigated as a function of plutonium enrichment and lifetime. The results of the study are summarized in Figures 3.7 through Figure 3.14. The results of one-dimensional LUX studies to determine the relative power distribution as a function of lifetime for configuration (a) and (b) are included in Appendix B, Figures B-6 through B-9. In these calculations the plutonium loading was 5.04 a/o Pu. A LUX calculation at the beginning of life for a fuel loading of 3.98 a/o is included in Figure B-5 for comparison.

An alternate configuration in which eight $\text{PuO}_2\text{-UO}_2$ assemblies (5.04 a/o Pu, Zr clad) are installed around a center highly depleted UO_2 assembly was investigated using PDQ-3 to determine boron and control rod worth and local power peaking effects. The resulting PDQ-3 power distributions are shown in Figures B-10 and B-11. The reactivity results are summarized in the tabulation on page 3-20.

ED. SK. 322285-B

Figure 3.7 Calculated k_{eff} (Hot, Clean, 23-1/2 MWt), Initial Boron Concentration, and Lifetime as a Function of Plutonium Loading and Assembly Location

ED. SK. 32279-B

Figure 3.8 - Inverse Boron Worth and Equilibrium Xenon Defect as a Function of Plutonium Loading and Assembly Location

ED. SK. 322283-B

Figure 3.9 - Relative Assembly Power Level as a Function of Plutonium Loading and Assembly Location

CD. SK. 322284-B

Figure 3.10 - Peak-to-Average Power Ratio and Fractional Plutonium Power as a Function of Plutonium Loading and Assembly Location

ED. SK. 322281-B

Figure 3.11 - Delayed Neutron Fraction ($\bar{\beta}$) as a Function of Plutonium Loading and Assembly Location

ED. SK. 322286-B

Figure 3.12 - Fractional Plutonium Power as a Function of Lifetime for Variations in Enrichment and Assembly Location

ED. SK. 322282-B

Figure 3.13 - Peak-to-Average Power Ratio in Plutonium as a Function of Lifetime with Variations in Enrichment and Assembly Location

ED. SK. 322280-B

Figure 3.14 - Peak-to-Average Power Ratio in Uranium as a Function of Lifetime with Variations in Plutonium Enrichment and Assembly Location

Fig. No.	Boron Conc.	Rod Conf.	Calculated Maximum Hot Spot*	Calculated k_{eff}	Rod or Poison Worth, $\% \Delta k/k_{initial}$
B-10	1000	All Out	2.01	1.060	0.0
B-10	2000	All Out	2.06	0.996	6.0
B-11	1000	Rod 5- In	2.06	1.024	3.4
B-11	1000	Rods 2,5 In	-	0.980	7.5
B-11	1000	Rods 1,2, 5,6 In	-	0.929	12.4
* Calculated Maximum Hot Spot for Saxton Core I = 1.91					

Power distributions for three enrichment variations in eight $\text{PuO}_2\text{-UO}_2$ assemblies installed in corner locations were determined using PDQ-3. The analysis shows that a Pu enrichment of 5.04 a/o Pu is needed in order to limit the maximum hot spot in the central uranium region to 1.91, the same value as that of Saxton Core I. (The PDQ-3 power distributions for Saxton Core I is included in Appendix G.) As the plutonium enrichment in the corner elements is increased to more than 5.04 a/o Pu, the maximum hot spot decreases. The PDQ-3 power distributions for the three enrichments are included in Appendix B, Figure B-12.

3.3 Reference Configuration

As a result of these studies, it was concluded that to achieve the desired burnup in the plutonium fuel it would be necessary to install the mixed-oxide ($\text{PuO}_2\text{-UO}_2$) assemblies in the middle of the core. Consequently, a reference configuration was established in which nine

Zircaloy-4 clad plutonium assemblies are installed in the center of the core with twelve conventional Saxton uranium fuel assemblies installed on the periphery. Figure 3.15 shows the resulting power distributions as calculated by PDQ-3 for three different plutonium enrichments. Also shown in this figure is the value for the maximum hot spot in the plutonium region. A comparison of the maximum hot spot values calculated in the three cases indicates that to maintain the same value as that achieved in Saxton Core I, it would be necessary to specify a plutonium enrichment of 4.00 a/o Pu. However, with this enrichment, it would not be possible to obtain the desired lifetime. Consequently, the decision was made to permit an increase in the hot channel factor in order to meet the burnup and lifetime objectives.

3.4 Parallel Criticality Calculations

To evaluate the adequacy of the available analytic methods and cross sections, a comparison of analysis and experiment was made for six mixed-oxide ($\text{PuO}_2\text{-UO}_2$) experiments conducted at Hanford. The LEOPARD code was used in the analysis of six different 1.5 w/o $\text{PuO}_2\text{-UO}_2$ critical and approach-to-critical experiments using the experimentally determined buckling as input to the code. Table 3.3 summarizes the comparison between analysis and experiment for two different Pu-239 cross section sets. The Leonard cross sections normally used in LEOPARD resulted in a calculated k_{eff} higher than the experimental value for each of the six lattices. To account for the reactivity discrepancy that was observed, an allowance was included in subsequent installed reactivity and lifetime predictions.

9 PuO₂-UO₂ assemblies (5.04 a/o Pu, Zr clad) in center positions, 1000 ppm Boron Eigenvalue: 1.077

NOTE: The underlined value in each assembly is the average power in that assembly relative to the average power in the core. The relative power is also shown for individual fuel rods near the uncontrolled corners of assemblies (where hot spots usually occur). If the hot rod is elsewhere, the value and location are shown.

9 PuO₂-UO₂ Assemblies (4.000 A/o Pu, Zr Clad) in center positions 1000 PPM Boron Eigenvalue: 1.065

9 PuO₂-UO₂ assemblies (6.0 a/o Pu, Zr clad) in center positions, 1000 ppm Boron Eigenvalue: 1.095

Figure 3.15 - PDQ-3 Calculated Power Distributions for Three Enrichment Variations in Nine Central PuO₂-UO₂ Assemblies

An earlier cross section set with a higher capture-to-fission ratio (α_{49}) resulted in better agreement with the experiment. The α of this cross section set was in agreement with a then latter review of the 2200 m/sec nuclear parameters by Sher.¹⁵ However, Leonard cross sections continued to be used as the reference cross section set because of a more favorable comparison of analysis with burnup for the Yankee core.

During the WREC critical experiments program, a more extensive criticality study was made using a number of different cross section sets and alternate methods of analysis. Details concerning this criticality study are included in Section 5.2. The end result of the study was the adoption of alternate cross sections for use in the correlation of analysis with the WREC critical experiments and in the calculations to be made in following operation of the Saxton Core II. However, Leonard cross sections were used throughout the nuclear design calculations.

Table 3.3

Comparison of Hanford Experiments and LEOPARD Calculations*

<u>Lattice Spacing (inches)</u>	<u>H/Pu (atom ratio)</u>	<u>Calculated k_{eff} Leonard cross-sections</u>	<u>Calculated k_{eff} Earlier cross-sections</u>
0.55	230	1.0165	0.9988
0.60	326	1.0240	1.0052
0.71	567	1.0314	1.0113
0.80	794	1.0297	1.0040
0.90	1077	1.0272	1.0062
0.93	1169	1.0271	1.0061

Experimental $k_{eff} = 1.0$.

4.0 Nuclear Fuel Design Calculations

4.1 Introduction

4.1.1 Calculation Sequence

The completion of the parameter studies described in the previous section led to the establishment of a reference configuration in which nine plutonium fuel assemblies were installed in the center of the reactor with 12 uranium fuel assemblies installed on the periphery. An enrichment range from 5.5 to 6.0 w/o $\text{PuO}_2\text{-UO}_2$ was then selected for more detailed calculations of reactivity and lifetime. The available lifetime for the two enrichments was determined by means of one-dimensional (LUX) calculations. Two-dimensional burnup calculations in X-Y geometry (TURBO*) were carried out for the higher enrichment. At the completion of this work, a fuel enrichment of 6.0 w/o $\text{PuO}_2\text{-UO}_2$ was specified.

Later, the fuel source was changed which resulted in a change in the isotopic composition of the plutonium fuel. The following list compares the two fuel compositions.

Isotope	Original Specification, % (Used in Parameter Study)	Revised Specification, %
Pu-239	88.15	90.44
Pu-240	10.24	8.60
Pu-241	1.61	0.91
Pu-242	-	0.05

The effect of the change in isotopic composition was to increase the initial installed reactivity by $\approx 0.6\% \Delta k/k$, increase local power peaking by $\approx 1\%$, and to increase slightly the calculated lifetime. Since the net effect on core performance was small, no change was made in the enrichment specification.

Detailed calculations of reactivity coefficients were then carried out using the revised isotopic composition. Experimental information from the operation of Saxton Core I, included in Appendix G, provided a check on these calculations. During the course of this work, a number of changes were made in the mechanical design such as a small reduction in fuel pellet diameter and an increase in the amount of material in the Core II inconel grid design. As a result, a review was made of the available reactivity, the reactivity effect of the various changes, and the size of the uncertainties associated with the reactivity and lifetime predictions. Based on this review, the enrichment specification was increased from 6.0 w/o $\text{PuO}_2\text{-UO}_2$ to 6.6 w/o $\text{PuO}_2\text{-UO}_2$.

Sufficient additional calculations were then made to revise the reactivity and lifetime predictions, the expected power distributions, and to show that the reactivity coefficients were not altered as a result of the enrichment change.

After the performance of the critical experiments at the WREC using the design fuel, final revisions were made in the reactivity and lifetime predictions and revised methods and cross sections were adopted for use in the operational analysis of Saxton Core II.

4.1.2 Scope of the Analysis Description

The description of the design calculations reported in this section is divided into two parts. The first part concerns the more recent calculations to determine the boron concentration requirements and control rod worths expected in the initial testing of Saxton Core II. These calculations were carried out after the completion of the critical experiments program and are based on the use of cross sections and methods that were selected because of their favorable comparison of analysis with the WREC experiments. Therefore, these results represent the best predictions available for comparison of the initial test results with the analysis.

The second part concerns the calculations carried out in advance of the critical experiments program. These results were later modified on the basis of the WREC experimental data. The calculations include the predictions of installed reactivity, lifetime, local power peaking, regional power sharing, and reactivity coefficients. Additional detail concerning these calculations is included in Appendix C.

4.2 Post-Critical Experiment Analysis Summary

Calculations were made using a LEOPARD-5/BUBBLE¹⁶ /PDQ-3 computer code sequence to determine the boron concentration requirements and control rod worths to be expected in the initial zero power tests of Saxton Core II. The core configuration used in the analysis is shown in Figure 4.1. LEOPARD-5 which contains Sher (1965) cross sections was selected on the basis of a favorable comparison of analysis with the WREC critical experiments (Section 5.2.5). To account for the small

Figure 4.1: PLAN VIEW OF SAXTON CORE SHOWING FUEL ASSEMBLY LOCATIONS.

differences between analysis and experiment using these cross sections, a reactivity bias was included. The BUBBLE code was chosen because it provides a flexible means of duplicating analytically the experimental history of Saxton Core II. The basic purpose of the code is to generate macroscopic cross sections as a function of boron concentration and burnup and punch them out in a form that can be used in PDQ-3. The code is essentially an interpolation and extrapolation procedure using calculated data from LEOPARD-5 within a defined burnup-boron concentration matrix.

For the zero-power calculations, a series of LEOPARD-5 calculations was carried out for each major fuel region in the core at four different boron concentrations. The BUBBLE code was then used to punch out PDQ-3 input constants at the desired boron concentrations. Number densities for fuel areas that are too small to dominate the spectrum in which they occur, such as fuel followers and sub-assemblies, were combined in the code with macroscopic cross sections from the nearest dominate fuel area to produce macroscopic constants. Group constants for the water slots were determined in BUBBLE from a LEOPARD-5 calculation matrix of water and steel at various steel volume fractions and boron concentrations.

An 87x87 PDQ-3 matrix was used in the analysis which is essentially the same matrix as that used throughout the Saxton Core II nuclear fuel design. A variation in axial buckling with boron concentration was included. To correct for the longitudinal grid structure which could not be included in the PDQ-3 calculation in X-Y geometry, a previously determined value of $0.3\% \Delta k/k$ was used.

Group constants for the control rods were determined using the procedure outlined in Appendix E. The accuracy of the method was demonstrated by the good agreement between analysis and experiment for the control rod experiments at the WREC in both single region and multi-region cores.

A summary of the zero-power reactivity calculations for Saxton Core II is included in Table 4.1. Table 4.2 contains a summary of the reactivity effect of various rod patterns and changes in boron concentration derived from these results.

Based on the post-critical study, the following boron requirements were anticipated:

<u>Core Condition</u>	<u>Boron, ppm</u>
Cold, zero-power	2895 ppm
Hot, zero-power	2460 ppm

Upon the completion of these calculations, a check was made using the LEOPARD-5/BUBBLE/PDQ-3 calculation sequence for the hot, zero-power, just critical condition for Saxton Core I (1804 ppm boron). Using this new sequence the calculated reactivity was $\sim 1.2\%$ $\Delta k/k$ higher than that indicated by the experiment. As described in Appendix G, the calculated k_{eff} for this core condition was in excellent agreement with the experiment using the analytic procedure that was used throughout the nuclear design of Saxton Core II. In both calculations the longitudinal grid structure was neglected. Consequently, in the original calculations in which good agreement was obtained, discrepancies in the basic method of

TABLE 4.1

Summary of Saxton Core II Zero Power Reactivity Calculations
Carried Out After Completion of the WREC Critical Experiments

Cold,¹ Zero Power

<u>Control Rods Inserted</u>	<u>PPM Boron</u>	<u>Calculated k_{eff}^2</u>	<u>Calculated k_{eff}^3</u>
0	2300	1.0392	1.0272
0	2950	0.9966	0.9846
2,5	2950	0.9267	0.9147
2,5	1800	1.0042	0.9922
1,3,4,6	1800	1.0182	1.0062
1,2,3,4,5,6	1800	0.9238	0.9118
1,2,3,4,5,6	2300	0.8917	0.8797

Hot,⁴ Zero Power

<u>Control Rods Inserted</u>	<u>PPM Boron</u>	<u>Calculated k_{eff}^2</u>	<u>Calculated k_{eff}^3</u>
0	2300	1.0081	0.9961
0	2650	0.9906	0.9786
0	1225	1.0701	1.0581
2,5	2650	0.9144	0.9024
2,5	1225	0.9904	0.9784
2,5	1800	0.9597	0.9477
1,2,3,4,5,6	1225	0.8968	0.8848

1. 100 F, 350 psi
2. 0.006 was subtracted from the calculated value for the reactivity bias and inconel grids.
3. 0.018 was subtracted from the calculated value for the reactivity bias, inconel grids, and the reactivity discrepancy in the comparison of Saxton Core I using the same analysis method.
4. 533 F, 2000 psi.

TABLE 4.2

Calculated Boron and Control Rod Worth
for Saxton Core II

Cold,¹ Zero Power

<u>Initial Rods</u>	<u>Insert Rods</u>	<u>Remove Rods</u>	<u>Initial Boron, ppm</u>	<u>Final Boron, ppm</u>	<u>Reactivity² Change, % $\Delta k/k$</u>
0	-	-	2300	2950	4.13 (157.3 ppm/1% $\Delta\rho$)
0	2,5	-	2950	2950	7.25
0	All	-	2300	2300	15.33
All	-	2&5	1800	1800	9.64
All	-	1,3,4,6	1800	1800	8.33
All	-	-	1800	2300	3.45 (145 ppm/1% $\Delta\rho$)

Hot,³ Zero Power

<u>Initial Rods</u>	<u>Insert Rods</u>	<u>Remove Rods</u>	<u>Initial Boron, ppm</u>	<u>Final Boron, ppm</u>	<u>Reactivity² Change, % $\Delta k/k$</u>
0	-	-	1225	2300	5.86 (183.5 ppm/1% $\Delta\rho$)
0	-	-	2300	2650	1.81 (193.4 ppm/1% $\Delta\rho$)
0	2,5	-	2650	2650	7.90
0	2,5	-	1225	1225	7.58
0	All	-	1225	1225	17.56

1. 100 F, 350 psi

2. reactivity change = $\% \ln \frac{k_1}{k_2}$

3. 533 F, 2000 psi

analysis were compensated for by neglecting the longitudinal grid structure. When the new sequence was adopted, the reactivity base point for Saxton Core I was changed, thereby, introducing a reactivity error in the subsequent calculations. Therefore, the error obtained using the new sequence for the Saxton Core I base point was treated as a reactivity bias and was subtracted from the calculated zero-power reactivity values for Saxton Core II. The results are included in Table 4.1. Using these revised values the calculated boron concentration requirements are:

<u>Core Conditions</u>	<u>Corrected Boron</u>
Cold, zero-power	2700
Hot, zero-power	2230

4.3 Nuclear Fuel Design Summary

4.3.1 Available Reactivity

The system reactivity was determined by means of PDQ-3 two-group diffusion calculations in X-Y geometry. As shown in Table 4.3, the hot, clean reactivity at power is expected to be $11.2 \pm 0.8\% \Delta k/k$. This value includes an allowance for the discrepancy between the analysis and the WREC critical experiments and corrections for the differences in the analytic configuration and that actually installed. A core diagram of the configuration corresponding to the reactivity summary is shown in Figure 4.1. The following statements explain each item in the Table 4.3.

TABLE 4.3

Hot, Clean Reactivity - Saxton Core II

<u>Item</u>	<u>% $\Delta k/k$</u>
a. Calculated reactivity, PDQ-3 analysis, $(k-1)/k$	13.4
b. Change in follower locations	+ 0.4 $\pm$ 0.3
c. Analysis-to-experiment discrepancy	- 1.4 $\pm$ 0.6
d. Core configuration changes	- 0.8 $\pm$ 0.4
Supercritical loop	- 0.25
Partially burned element	- 0.05
Vibratory-compacted fuel location	- 0.0
Grid material change	- 0.3
Fuel density change	- 0.2
e. Non-uniform moderator temperature	- 0.2 $\pm$ 0.2
f. Non-uniform Doppler	- 0.2 $\pm$ 0.2
<hr/>	
Available Reactivity	11.2 $\pm$ 0.8

- a) The calculated reactivity of the system was determined by means of PDQ-3 calculations with group constants determined using Leonard cross sections. The analysis included the presence of partially depleted fuel followers and L-sections. An exposure equivalent to 20,000 MWD/tonne burnup was assumed for the followers and 9,000 MWD/tonne was assumed for the L-sections.
- b) Two fuel followers in the center of Saxton Core I were moved to outer positions in Saxton Core II. In this change the more heavily burned fuel in the center of the core was replaced by fuel having a lower burnup. The effect of the fuel change was to increase the installed reactivity.
- c) The comparison of the WREC critical experiment results with the analysis using the same methods as those used in the Saxton Core II design calculations is described in Section 5.2. A discrepancy of approximately 2% $\Delta k/k$ was found for the unborated 0.56 inch lattice which has approximately the same H/Pu ratio as the Saxton design at temperature. Applying a calculated weighting factor of 0.7 to account for a partial core loading results in an allowance of 1.4% $\Delta k/k$ for the discrepancy.
- d) The changes of item (d) include the presence of the supercritical assembly with a steel plug in position E1, the partially burned element in E1, and the movement of the vibratory-compacted fuel assemblies from corner positions to flat positions. Also included are changes in grid material from stainless steel to Inconel 718 and a change in fuel density.

- e) The non-uniform moderator temperature distribution is calculated to be a small negative effect.
- f) The effect of a more negative Doppler coefficient due to a non-uniform fuel temperature distribution was not considered in previous reactivity tabulations. It is expected to be small negative effect.

As described in Section 3.4, the original allowance to account for a possible discrepancy between analysis and experiment was based on a comparison of a number of Hanford approach-to-critical experiments. For these experiments at a lower plutonium fuel loading than that specified for the Saxton reactor, an average discrepancy of $\approx 2.6\% \Delta k/k$ was found. To compensate for this discrepancy, an equivalent full core error of $\approx 3\% \Delta k/k$ was assumed which was approximately equal to the maximum error found in the comparison of the Hanford experiments. A calculated factor of 0.7 was applied for a partial core loading. As a result, a correction of 2.0% was used in the original reactivity predictions for the design core. The discrepancy of approximately 2% for the WREC experiments described in Item (c) above with the resulting allowance of 1.4% $\Delta k/k$ in the reactivity prediction points out that only a small change in the performance estimates were necessary as a result of the WREC experimental data obtained with the design fuel.

4.3.2 Available Lifetime

The available lifetime for the Saxton reactor containing a partial core of plutonium fuel was determined by both one-dimensional (radial) and two-dimensional (X-Y) burnup calculations. The reactivity as a function of

lifetime for an enrichment of 6.6 w/o $\text{PuO}_2\text{-UO}_2$ as determined from a LUX radial calculation is shown in Figure 4.2. This figure also shows the possible variation in lifetime that would result for the maximum and minimum expected values of initial reactivity of Table 4.3. (Additional details concerning the lifetime calculations are included in Appendix C.) The calculated just-critical boron concentration as a function of lifetime is shown in Figure 4.3.

Lifetime calculations for Saxton Core I, described in Appendix G, resulted in a calculated lifetime approximately 1000 hours less than that actually available as indicated by an extrapolation of the experimental data.

While no comparable burnup information is available for the plutonium fuel, the lifetime calculation is believed to be conservative.

4.3.3 Initial Power Distribution

The peak power in the core will occur in the plutonium region at the beginning of life. Figure 4.4 shows the core radial power distribution as calculated by PDQ-3. The analysis was carried out for a plutonium enrichment of 6.5 w/o PuO_2 ; however, the difference in power levels for the final enrichment is small, approximately 1% in local power. The local-to-core average power in each fuel rod in the central plutonium assembly is shown in Figure 4.5. The maximum peak rod power listed in this figure is higher than that shown in the diagram to account for a change in boron concentration from 1000 ppm to the at-power beginning of life requirement of ≈ 2000 ppm and to account for the difference in the

Figure 4.3

PDQ Power Distribution

Configuration: 9 PuO₂-UO₂ Assemblies (6.5 w/o PuO₂) in Center Positions
 No rods, 1000 ppm Boron, Depleted Followers and L-Sections

The underlined value in each assembly is the average power in that assembly relative to the average power in the core. The relative power is also shown for individual fuel rods near the uncontrolled corners of assemblies (where hot spots usually occur).

Figure 4.4 Saxton-Plutonium Core Power Distribution

Conditions: 9 PuO₂-UO₂ Assemblies (6.5 w/o PuO₂) in Center
 No Rods, 1000 ppm Boron
 Depleted Followers and L-Sections

2.03	1.85	1.88	2.14		2.14	1.91	1.90	2.10
1.84	1.66	1.66	1.76	1.89	1.76	1.67	1.68	1.90
1.87	1.66	1.63	1.64	1.66	1.64	1.63	1.67	1.92
2.15	1.77	1.64	1.62	1.62	1.63	1.65	1.77	2.14
	1.94	1.66	1.63	1.62	1.63	1.66	1.90	
	1.94	1.673	1.63	1.63	1.63	1.65	1.77	2.15
	1.93	1.71	1.68	1.67	1.65	1.64	1.68	1.89
	2.00	1.94	1.95	1.95	1.78	1.67	1.68	1.87
					2.16	1.89	1.96	2.06

Peak Rod Power = 2.36

(This value includes a correction for an increase in boron from 1000 ppm to 2000 ppm, a correction for the difference between calculated and measured power in Saxton Core I, and a 10% safety factor)

Figure 4.5 Center Assembly Power Distribution

calculated and measured maximum power from Saxton Core I. This comparison also includes a 10% safety factor to allow for a possible error in translating the flux wire measurements to local fuel rod power (Appendix G).

The maximum peak radial power of 2.36 for Saxton Core II compared to a Saxton Core I value of 2.03 shows that it was necessary to increase the nuclear hot channel factor over that of a conventional all uranium core to insure meeting the lifetime requirements. Consequently, it may be necessary to limit the reactor power level initially to avoid exceeding a linear power density limit of 16 kw/ft. The necessity for an initial power limitation will be determined by measurements during the initial test program for the design core.

Measurements were made during the WREC critical experiments program to determine the magnitude of the possible error in the power peaking calculation. Two factors are significant in establishing the maximum local power in a multi-region core. The first concerns local power peaking in regions of excess moderation.

To investigate this effect, power peaking measurements were made near water slots in both single region and multi-region cores. In addition, an experiment was conducted in which simulated fuel followers and flux wire thimbles were installed in a two region core. This experiment approximates the maximum power peaking condition of the design core, power peaking near flux wire thimbles. These experiments and the associated analysis, described in Section 5.4, showed that the methods of analysis used in the nuclear fuel design overpredict the power

peaking in regions of excess moderation for both UO_2 and $\text{PuO}_2\text{-UO}_2$ fuels. The largest discrepancy occurred in the plutonium fueled cores. In the experiment containing simulated flux wire thimbles, an overprediction of approximately 5% results when using the same methods as those used in the nuclear fuel design.

A second factor concerns the power sharing between the two different fueled regions. If more power than expected is produced in the plutonium region, it would be necessary to limit the total core power in order to limit the maximum local power to 16 kw/ft. Power sharing measurements in the WREC multi-region cores were made using two different methods to interpret the experimental results. The two methods are described in Section 5.5. The deviation between analysis and experiment was $\approx 5\%$ using a thermal method and approximately 1% using a foil method. If the thermal method is the more accurate, the power peaking in the plutonium fuel would be 5% higher than indicated by the analysis. If the foil method is the more accurate, the analysis and experiment are in excellent agreement. At this time there is no justification for selecting one method over the other even though the analysis and experiment are in better agreement if the foil method is used.

When both power peaking and power sharing are considered, the increase in local power caused by a possible discrepancy in determining the correct power sharing between the two fuels (by the thermal method) is compensated for by an overprediction of power peaking in regions of excess moderation. If the thermal method is subsequently verified by nuclear and thermal measurements in the design core, the beginning of life radial power peak-

to-average will be 2.36 as shown in Figure 4.5 and it will be necessary to limit the initial power to 21.6 MWt to avoid exceeding a linear power limit of 16 kw/ft. On the other hand, if the foil data proves to be the more accurate, it should be possible to increase the core power level by approximately 5% to 22.7 MWt.

4.3.4 Lifetime-Power Relationship

If it is necessary to limit the total core power at the beginning of life because of local power peaking, this limitation will be short lived since the initial peak power in the plutonium region reduces rapidly with time due to burnup. The following list summarizes the maximum values of the peak rod radial power to core average radial power for both the plutonium and uranium fuel regions as a function of time. The values were derived from a TURBO* two-dimensional burnup calculation.

Hours Operation 23.5 MWt	PuO ₂ -UO ₂ Fuel	UO ₂ Fuel
0	2.36	1.22
6,000	1.88	1.26
12,000	1.63	1.23
Average for 8250 MWD	2.00	1.24

The two-dimensional TURBO* burnup calculation was carried out at an enrichment of 6.0 w/o PuO₂-UO₂ with the original isotopic composition. Since the effect of these changes was comparatively small, ($\approx 5\%$ in local power) the calculations are useful in following the change in peak power with time. Figure 4.6 compares the core radial power distribution at the beginning of life, after 6000 hours, and after 12,000 hours from

Conditions: 6,000 hours, 800 ppm boron, All Rods Out, 23.5 MWt

Note: The underlined value in each assembly is the average power in that assembly relative to the average power in the core. The relative power is also shown for individual fuel rods near the uncontrolled corners of assemblies (where hot spots usually occur). If the hot rod is elsewhere, the value and location are shown.

Conditions: 0 hours, 2400 ppm boron, All Rods Out, 23.5 MWt

Conditions: 12,000 hours, 0 ppm boron, All Rods Out, 23.5 MWt

Figure 4.6 - TURBO*, Relative Assembly Power at the Beginning of Life, After 6,000 Hours Operation, and After 12,000 Hours Operation at 23.5 MWt

the TURBO* calculation. The individual fuel rod power for the center element for the same time periods are shown in Figure 4.7.

The TURBO* lifetime calculation sequence assumed chemical shim operation. In the first step sufficient boron was added to reduce the calculated k_{eff} to approximately 1.0. Then a series of calculation steps consisting of a burnup period followed by boron dilution to restore the k_{eff} to approximately 1.0 was used. For the final depletion step, no boron was present in the core. The calculation sequence that was followed is summarized in Figure C-1 in Appendix C. A comparison with a one-dimensional LUX calculation for the same core configuration and fuel composition is also included in Appendix C.

The rapid reduction in peak power as a function of operating time at 23.5 MWt is shown in Figure 4.8. A reactor operating sequence that does not exceed the 16 Kw/ft limit is also shown in Figure 4.8. This figure shows that a power level of 23.5 MWt will be allowed following six months operation at about 22 MWt and a load factor of 0.5. Thereafter, the maximum linear power density decreases to approximately 13.0 Kw/ft after 8250 MWD operation. In these calculations, the effect of a non-uniform axial burnup has not been considered. Therefore, in actual practice the peak power will decrease faster than is shown in Figure 4.8.

4.3.5 Fuel Burnup

A design lifetime objective of 8250 MWD and the listed average value for the peak rod power in each region results in the following predicted burnup values.

2.01	1.88	1.90	2.13		2.12	1.93	1.91	2.05
1.87	1.73	1.73	1.82	1.92	1.82	1.74	1.75	1.91
1.90	1.73	1.69	1.70	1.72	1.70	1.70	1.74	1.93
2.14	1.82	1.70	1.69	1.69	1.69	1.71	1.82	2.13
	1.96	1.72	1.69	1.69	1.69	1.72	1.93	
	1.96	1.73	1.70	1.69	1.69	1.71	1.83	2.15
	1.95	1.76	1.73	1.73	1.71	1.70	1.74	1.92
	2.06	1.95	1.96	1.98	1.84	1.74	1.75	1.90

Conditions: 0 hours, 2400 ppm Boron,
All Rods Out, 23.5 MWt

2.17	1.92	1.90	2.04
------	------	------	------

1.662	1.577	1.592	1.720		1.697	1.582	1.572	1.698
1.574	1.479	1.468	1.524	1.588	1.523	1.470	1.483	1.573
1.582	1.467	1.426	1.431	1.442	1.433	1.431	1.471	1.584
1.721	1.528	1.431	1.413	1.412	1.415	1.434	1.526	1.701
	1.604	1.448	1.414	1.409	1.414	1.445	1.592	
	1.591	1.453	1.421	1.415	1.416	1.435	1.530	1.727
	1.585	1.477	1.455	1.452	1.437	1.433	1.475	1.600
	1.643	1.588	1.597	1.612	1.537	1.476	1.489	1.589

Conditions: 6,000 hours, 800 ppm Boron,
All Rods Out, 23.5 MWt

1.733	1.600	1.588	1.677
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1.466	1.414	1.419	1.484		1.457	1.390	1.382	1.417
1.413	1.351	1.337	1.371	1.405	1.364	1.329	1.339	1.383
1.419	1.338	1.299	1.297	1.303	1.296	1.296	1.330	1.392
1.487	1.376	1.299	1.278	1.274	1.279	1.297	1.367	1.460
	1.415	1.311	1.278	1.270	1.276	1.306	1.409	
	1.402	1.317	1.287	1.279	1.281	1.302	1.376	1.490
	1.398	1.337	1.319	1.315	1.305	1.306	1.345	1.427
	1.424	1.400	1.407	1.422	1.384	1.347	1.362	1.424

Conditions: 12,000 hours, 0 ppm Boron,
All Rods Out, 23.5 MWt

1.498	1.431	1.426	1.479
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Figure 4.7 - TURBO* Center Assembly Fuel Rod Power Distribution at the Beginning of Life, After 6,000 Hours Operation, and After 12,000 Hours Operations at 23.5 MWt

	Burnup in PuO ₂ -UO ₂ Fuel (MWD/tonne)	Burnup in UO ₂ Fuel (MWD/tonne)
Average	13,600	7,300
Average in Peak Rod	20,000	11,500

These burnup values were set by the anticipated irradiation time available. If the lifetime calculation shown in Figure 4.2 is correct, the design capability of the core is 9300 MWD or slightly higher than the design objective. Consequently, if the core is allowed to run to the end of life, the following burnup values are possible.

	Burnup in PuO ₂ -UO ₂ Fuel (MWD/tonne)	Burnup in UO ₂ Fuel (MWD/tonne)
Average	15,000	8,300
Average in Peak Rod	22,500	13,000

4.3.6 Reactivity Coefficients

4.3.6.1 Doppler Coefficient

Doppler coefficient calculations were carried out as a function of the effective fuel temperature where the effective fuel temperature is defined as that temperature which gives the correct experimental power coefficients of reactivity when employed in standard design calculations. The relationship between pellet temperature and effective fuel temperature is based on the work of W. T. Sha^{/17} who correlated effective fuel temperature for

Doppler broadening of U-238 resonances with experimental power coefficient measurements in the Yankee, Saxton, and BR-3 reactors. The temperature power relationships developed for the Saxton reactor are shown in Figure 4.9.*

The Doppler coefficient was determined by completing a series of LEOPARD calculations for values of T_{eff} ranging from 800°F to 2000°F in 200 degree increments. The resulting group constants were then used in AIM-5 radial diffusion calculations. Figure 4.10 summarizes the results for variations in T_{eff} alone and for the variation in component temperatures shown in Figure 4.9. The difference between the two curves is due to the change in moderator content resulting from the small change in clad dimensions with temperature. Figure 4.11 shows the derived power coefficient based on the Doppler calculations and the temperature relationships of 4.9.

Calculations were also made to determine the fraction of the total Doppler effect due to fuel temperature changes in each region separately. Approximately 70% of the total reactivity change due to Doppler is the result of temperature changes in the central plutonium region while 30% is the result of temperature changes in the outer uranium region.

4.3.6.2 Moderator Coefficient

The moderator temperature coefficient was determined by a sequence of LEOPARD and AIM-5 calculations. Figure 4.12 summarizes the results

* Initial temperature calculations for the Zr-clad $\text{PuO}_2\text{-UO}_2$ fuel rods resulted in approximately the same average fuel temperature as that of the 304 SS clad UO_2 fuel rods of Saxton Core I. Later, the pellet-to-clad gap was increased for the $\text{PuO}_2\text{-UO}_2$ rods causing an increase in average fuel temperature. The effect of this change was factored into the reactivity summary of Table 4.3 and has been included in the at power calculations in preparation for following the operation history of Saxton Core II.

for the partial plutonium core. The figure also includes a comparison of the measured values of the moderator temperature coefficient with the analysis for the conventional Saxton all uranium loading.

Temperature coefficient measurements were made in both single region and multi-region cores during the WREC critical experiments program. The measured temperature coefficient in a single region $\text{PuO}_2\text{-UO}_2$ assembly was more negative than that of a UO_2 fueled assembly at the same lattice pitch. This result is in agreement with that expected from the analysis. The magnitude of the measured temperature coefficient difference for the two fuels and the correlation of the analysis with experiment for the temperature coefficient measurements at low temperature are not regarded as being of particular significance to the design because of the uncertainties associated with this type of measurement and because of the large temperature difference between the design temperature and the temperature of the experiment. Buckling measurements made at a wide range of H/Pu ratios are more easily interpreted and provide an indication as to the effect of density changes upon reactivity. Based on an evaluation of all applicable WREC experiments, there is no need to modify the design core moderator temperature coefficient as a result of the experimental data.

A study was carried out during the course of the nuclear design to determine the factors contributing to the more negative temperature coefficient in the plutonium fueled region than in the region containing uranium. The results show the factors can be separated into a thermal and epithermal component. In the epithermal energy range, the plutonium fuel has a more negative temperature coefficient because of the effects of

higher epithermal absorption. In the thermal energy range, three changes in fuel rod design and fuel composition account for the more negative temperature coefficient. The effects of the changes are approximately equal in magnitude.

- 1) change in the spectrum on the Pu-239 capture-to-fission ratio
- 2) change in cladding material from 304 SS to Zr-4
- 3) change in U-235 content from 5.7% to 0.714%.

Pressure and Void Coefficient

Table 4.4 compares the pressure and void coefficients for the conventional uranium loading with that of the core containing nine plutonium fuel assemblies. The analysis was carried out using a LEOPARD and AIM-5 sequence. Measured values of the pressure coefficient at various boron concentrations are included in the table. The analysis and experiment are in good agreement for comparable conditions.

4.3.7 Boron and Control Rod Worth

4.3.7.1 Boron Worth

The LEOPARD-AIM-5 calculated boron worth in Saxton Core II as a function of boron concentration is shown in Figure 4.13. The good agreement between analysis and experiment obtained using the same procedure for the borated WREC critical experiments (Section 5.6.5) provides some assurance as to the validity of these results.

TABLE 4.4

Pressure and Void Coefficients

Pressure Coefficient: $\frac{1}{k} \frac{\partial k}{\partial P}$ ($\% \Delta k/k/\text{psi}$) $\times 10^6$, Moderator = 530°F

Saxton Core II1000 ppm
BoronMaximum BoronSaxton Core I1000 ppm
BoronCleanRodded,
0 Boron

+ 3.5

Analysis

+ 1.30 (2000 ppm)

+ 2.1

+ 3.1

+ 5.6 (all rods)

Experiment

+ 1.45 (1600 ppm)

+ 2.4

+ 4.6 (partial rods)

Void Coefficient: $\frac{1}{k} \frac{\partial k}{\partial V}$ ($\% \Delta k/k/\%$ void), Moderator = 530°F

Saxton Core II1000 ppm
Boron2000 ppm
BoronSaxton Core I1000 ppm
BoronCleanFully Rodded,
0 Boron

- 0.27

- 0.10

- 0.18

- 0.26

- 0.43

Figure 4.13

4.3.7.2 Control Rod Worth

The nominal mode of control for the Saxton plutonium core is expected to be chemical shim. However at times during the program, it may be necessary to employ control rods alone, chemical shim alone, or combinations of the two. In any mode of control, soluble poison will be required for cold, clean core shutdown.

The control rod worths were determined by means of a series of PDQ-3 calculations using a procedure that satisfactorily determined the rod worth in Saxton Core I. (See Appendix G.) The calculated total bank worth for the plutonium core was 16.9% $\Delta k/k$. Stuck rod calculations showed that the total bank worth with the most effective rod stuck in the fully withdrawn position was 11.7% $\Delta k/k$. Figures 4.14 and 4.15 show the effect on the beginning of life power distributions for stuck rod and dropped rod conditions.

An alternate procedure was used to generate group constants for the control rods for the comparison of analysis with experiment during the WREC critical experiments program. Because of the favorable comparison with the experimental results, this procedure was applied to a re-evaluation of rod worths in the design core. As a result, the total bank worth is expected to be slightly higher than was previously determined, 17.6% $\Delta k/k$ as compared to 16.9% $\Delta k/k$. A summary of the best values for the expected rod worths is included in Table 4.2.

PDQ Power Distributions

Configuration: 9 PuO₂-UO₂ Assemblies (6.0 w/o PuO₂) in Center Positions
Rods 1,2,3,4,6 in, 1000 ppm Boron, Depleted Followers and
L-Sections

The underlined value in each assembly is the average power in that assembly relative to the average power in the core. The relative power is also shown for individual fuel rods near the uncontrolled corners of assemblies (where hot spots usually occur). The peak power includes a correction for the discrepancy between analysis and experiment from Saxton Core I.

Stuck Rod Power Distribution

Figure 4.14

Configuration: 9 PuO₂-UO₂ Assemblies (6.0 w/o PuO₂) in Center Positions
Rod 6 in, 1000 ppm Boron, Depleted Followers and L-Sections

Figure 4-15

4.3.8 Kinetic Characteristics

β_{eff} was evaluated throughout life from a TURBO* calculation. A constant value of ≈ 0.0049 was found. The β_{eff} was determined by weighting the delayed group yields of the various fissioning materials by the fraction of neutrons from each isotope. An importance factor, derived from LEOPARD calculations using different source spectra, was applied to account for the difference in importance of delayed and prompt neutrons. Table 4.5 summarizes the power and neutron source fraction in each of the two regions for each isotope at the beginning of life. The table also includes delayed neutron data and the region and core average β_{eff} and prompt neutron lifetime.

TABLE 4.5

Isotopic Power and Neutron Fractions, Delayed
Neutron Data, β_{eff} and Prompt Lifetime

Beginning of Life

<u>Region</u>	<u>Isotopic Power Fractions</u>				<u>Neutron Source Fractions</u>			
	<u>U-235</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-241</u>	<u>U-235</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-241</u>
Core Average	0.4301	0.0504	0.5078	0.0112	0.4008	0.0514	0.5352	0.0125
PuO ₂ -UO ₂	0.1302	0.0503	0.7998	0.0177	0.1188	0.0490	0.8132	0.0190
UO ₂	0.9390	0.0520	0.0088	0.0001	0.9340	0.056	0.0099	0.0001

Delayed Neutron Data

<u>Delayed Neutron Group, i</u>	<u>λ_i, sec⁻¹</u>		<u>β_i</u>			
	<u>U-235</u>	<u>Pu-239</u>	<u>U-235</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-241</u> ⁽³⁾
1	0.0124	0.0128	0.000215	0.000204	0.000074	0.000091
2	0.0305	0.0301	0.001424	0.002151	0.000626	0.000775
3	0.111	0.124	0.001274	0.002543	0.000443	0.000549
4	0.301	0.325	0.002568	0.006092	0.000685	0.000848
5	1.13	1.12	0.000748	0.003533	0.000181	0.000224
6	3.00	2.69	0.000273	0.001178	0.000092	0.000114
			0.0065	0.0157	0.0021	0.0026

β_{eff} and Prompt Neutron Lifetime

<u>Region</u>	<u>β_{eff}</u>	<u>Lifetime, μ sec⁽⁴⁾</u>
Core Average	0.0049	11.7 ⁽⁵⁾
PuO ₂ -UO ₂	0.0035 ⁽¹⁾	8.6
UO ₂	0.0075 ⁽²⁾	15.7

1. Includes partially burned uranium fuel followers
2. Includes partially burned uranium L-sections.
3. Relative abundance for Pu-239 used.
4. Calculation made with core containing 1000 ppm boron. (LEOPARD II)
5. Weighting based on power fractions of U = .439, contains L-sections and followers, Pu = .561.

5.0 ANALYSIS OF THE WREC CRITICAL EXPERIMENTS

5.1 Introduction

5.1.1 Description of the WREC Critical Experiments

5.1.1.1 Purpose

The purpose of the WREC critical experiments was to evaluate the nuclear design methods used in establishing the plutonium loading and to provide an improved base point for reactivity, lifetime, and power distribution predictions for Saxton Core II. Thus, these experiments represented engineering criticals directed toward establishing a satisfactory core configuration and mode of operation for the subsequent irradiation test. Two types of fuel were available for these experiments, the conventional Saxton UO_2 fuel (5.74 w/o U-235) and the mixed-oxide fuel (PuO_2 - UO_2) containing 6.6 w/o PuO_2 . The fuel rod specifications are included in Appendix A.

5.1.1.2 Planning

In planning the measurements program, the philosophy was to assign priority to those measurements that would provide information directly useful in meeting the performance objectives of the design core. Areas of particular importance to the Saxton design were (a) initial reactivity, (b) power peaking effects, (c) power sharing, and (d) kinetic and reactivity response.

a) Initial Reactivity

As pointed out previously, a large uncertainty existed as to the initial reactivity of the reference Saxton design as indicated by the comparison of analysis and experiment for six mixed-oxide critical and approach-to-critical experiments conducted at the Hanford Laboratories. For these more lightly loaded experiments (1.5 w/o PuO_2 compared to 6.6 w/o PuO_2 for Saxton), an average discrepancy of $\approx 2.6\% \Delta k/k$ was found. While an allowance for this discrepancy was included in the reactivity and lifetime predictions for the design core, the absence of any mixed-oxide experiments near the proposed loading meant a large reactivity uncertainty would be necessary until experiments at the design loading were completed.

b) Power Peaking Effects

The enrichment specification for the plutonium fuel rods required a compromise between irradiation, power limitations, and lifetime objectives. The decision to install the plutonium fuel assemblies in the center of the core was based on the desire to maximize the irradiation exposure of the mixed-oxide fuel.

However, since the plutonium region has a larger absorption cross section than the outer uranium region, the thermal flux undergoes a marked change at region boundaries and peaks to a greater extent

than normal in water slots near the plutonium. The plutonium fission cross section is also larger than the uranium. Consequently, local power peaking occurs in the plutonium rods at region boundaries and water slots. Therefore, measurements were needed to determine if the analysis adequately predicts power peaking effects and to indicate the most desirable method of correcting any unanticipated adverse power effects should they exist.

c) Power Sharing

The relative power produced in each of the two different fueled regions is important in establishing the power level at which the core can operate. If more power than expected is produced in the plutonium region where the hot-spot occurs, it will be necessary to reduce the total core power to avoid exceeding the hot-spot limitation.

d) Kinetic and Reactivity Response

There are a number of important differences in the reactivity and kinetic response in assemblies containing plutonium alone, uranium alone, and combinations of the two fuels. Some examples of the expected differences were the following:

- (1) The cores containing plutonium were calculated to have a more negative temperature and void coefficient.

(2) The analysis indicated that identical boron concentrations and control rods are worth less in plutonium cores.

(3) β_{eff} is less in the plutonium core.

5.1.1.3 Measurements Program

The planned measurements program was designed to investigate each of these areas. The program included single region criticals using each fuel, and multi-region criticals using both fuels in separate regions. The program also included experiments in which the relative positions of the $\text{PuO}_2\text{-UO}_2$ fuel and UO_2 fuel were interchanged. The purpose of these experiments was to aid in the evaluation of the data and to provide information that would be required if it became necessary to change the reference core design to one in which the $\text{PuO}_2\text{-UO}_2$ fuel was installed in the outer region.

5.1.2 Analysis Sequence

The analysis carried out in support of the WREC experimental program was divided into three phases. The first phase was concerned with the analysis needed in planning the experiments and establishing the expected fuel loading and boron concentration requirements for the specified critical configurations. A summary of this analysis is included in Appendix D together with an outline of the measurements program as originally planned. Also included in Appendix D are the results of transient studies required prior to the operation of a plutonium core at the WREC.

The second phase consisted of the analysis carried out during the experimental program. The purpose of this work was to process the experimental data, to aid in the interpretation of the experimental results, and to provide a basis for the selection of the most appropriate methods and cross-sections for use in the post-critical comparisons of analysis with experiment. As an example, since the kinetic characteristics of the two types of fuel are different, it was necessary to determine β_{eff} for each experimental configuration in order to correctly evaluate the reactivity measurements. Calculations of this type are included in a discussion of the applicable experiment. Because of the discrepancy between analysis and experiment for the Hanford mixed-oxide experiments, a criticality study including both the Hanford and WREC assemblies was carried out. This study, described in Section 5.2, provided the basis for the subsequent third phase analysis.

The third phase consisted of the detailed analysis of the experiments outlined in Table 2.1. The analytic results of this phase are reported according to the following functional headings:

1. Buckling Measurements
2. Power Peaking Effects
3. Power Sharing Experiments
4. Reactivity Measurements and Analysis
5. Flux Measurements and Analysis
6. Kinetics

5.1.3 Experimental Data

To properly compare the analysis with experiment, it is necessary to understand how the experiments were conducted and what results were obtained. For this reason, the experiments carried out under each functional heading are described. In addition, since the interpretation of the reactivity measurements is dependent upon the accuracy of the calculations of the kinetic characteristics, sufficient basic data for each experimental configuration is included to permit a re-evaluation of the reactivity measurements should it later prove desirable. (Experiment specifications are included in Appendix A and a description of experimental methods and results is included in Appendix F.) A more complete description of the experimental results is included in Reference 14.

There is some duplication in the comparison of analysis with some of the experiments. In those cases where the clarity is improved by the duplication, the results are included under both functional headings.

5.2 Criticality Study

5.2.1 Scope

Since the initial comparison of analysis with six Hanford mixed-oxide ($\text{PuO}_2\text{-UO}_2$) critical and/or approach-to-critical experiments using the LEOPARD code with Leonard cross sections (Section 3.4) resulted in a large reactivity discrepancy, a detailed investigation was carried out

during the critical experiment program to determine the most appropriate methods and cross sections to be used in the post-critical analysis of the WREC experiments and in the analysis to be performed in following the operating history of the Saxton reactor.

The studies were directed primarily to an investigation of the following effects:

1. Variations in the thermal energy treatment of the unit cell.
2. Variations in the scattering kernel for H_2O .
3. Variations in the thermal parameters for U-235, Pu-239, and Pu-241.

Calculations were carried out for five Hanford mixed-oxide (1.5 w/o PuO_2 - UO_2) experiments, five WREC mixed-oxide (6.6 w/o PuO_2 - UO_2) critical experiments, and two WREC UO_2 (5.7 w/o U-235) critical experiments. The investigation included variations in both method and cross sections.

5.2.2 Methods and Cross Sections

Criticality calculations were made with the LASER and LEOPARD codes. The basic difference between the two codes is in the calculation performed in the thermal energy group. In LASER, the thermal calculation consists of the THERMOS code with an expanded thermal energy group to cover the range from zero to 1.855 ev. Thus, the Pu-240 resonance at ≈ 1.05 ev is included in the thermal range. Two versions of LASER were used during the criticality study. The primary difference between the two versions

was in the thermal energy mesh. The improved LASER*, available at the close of the experimental program, contained a non-uniform thermal energy mesh with very fine energy intervals under the Pu-239 and Pu-240 resonances. The effect of the revised mesh for both the Hanford and WREC experiments was to improve the reactivity correlation as well as to significantly reduce the trend with pitch apparent in earlier calculations.

The thermal spectrum in LEOPARD, on the other hand, is determined by a Wigner-Wilkins SOFOCATE calculation with disadvantage factors determined using a modified form of the Amouyal-Benoist calculation at 172 energy levels from zero to 0.625 ev. cutoff.

The following four different cross section sets were used during the study:

1. Leonard cross-sections normally contained in the LEOPARD II production code.
2. Cross-sections reported by Westcott at the 1964 Geneva Conference and included in a revised LEOPARD II.
3. Cross-sections from Winfrith.
4. Cross-sections compiled by Sher and designated Sher (1965) and included in LEOPARD V.

5.2.3 Calculations Using Leonard Cross-Sections

5.2.3.1 Reactivity Results

The reactivity results from LASER and LEOPARD for the Hanford mixed-oxide experiments are summarized in Table 5.1 and Figure 5.1. The

* Analytic results reported in this report were obtained with the improved version of LASER.

Figure 5.1 - A Comparison of Analysis with Experiment for the Hanford Mixed-Oxide Experiments Using the LEOPARD and LASER Codes

LASER results illustrate the effect of a variation in the scattering kernel. (The LASER free-gas kernel is equivalent to that contained in LEOPARD). As shown in Table 5.1, the difference between LASER with the Nelkin kernel and LEOPARD is from 1% to 1.5% in reactivity over a range in which the H/Pu ratio varies from 230 to 1077. The difference due to the scattering kernel in LASER is about 0.3% $\Delta k/k$ and remains approximately constant throughout. The LEOPARD results of Figure 5.1 show a definite trend with H/Pu ratio while the trend with LASER is reduced.

TABLE 5.1

A Comparison of Analysis with Experiment for the Hanford Mixed-Oxide Experiments Using the LEOPARD and LASER Codes

<u>Pitch</u>	<u>H/Pu</u>	<u>Ke_{Leop}</u> [*]	<u>Ke_{Las Nel}</u>	<u>Ke_{Las F.G.}</u>
0.55	230	1.0165	1.0095	1.0134
0.60	326	1.0240	1.0126	1.0162
0.71	567	1.0314	1.0170	1.0200
0.80	794	1.0297	1.0156	1.0183
0.90	1077	1.0272	1.0159	1.0186

The results of the analysis for the UO₂ critical experiments carried out at the WREC are given in Table 5.2. A small temperature correction

* These calculations contain a built-in reactivity bias of 1.0036.

is included since the experiment was conducted at a temperature of 18°C while the analysis was carried out at 20°C. The analysis using both codes is in good agreement with the experiment.

Table 5.2

Comparison of Analysis with Experiment for the WREC
UO₂ Critical Experiments Using the LEOPARD and LASER Codes

<u>Pitch (in)</u>	<u>Ke_{Leop}</u>	<u>Ke_{Las Nel}</u>	<u>Ke_{Las F. G.}</u>
0.560	1.0048	1.0002	1.0038
0.792	1.0002	0.9983	1.0041

The comparison of analysis with experiment for the WREC mixed-oxide (PuO₂-UO₂) critical experiments is summarized in Table 5.3.

Table 5.3

A Comparison of Analysis with Experiment for the WREC PuO₂-UO₂
Critical Experiments Using the LEOPARD and LASER Codes

<u>Pitch (in)</u>	<u>Ke_{Leop F. G.}</u>	<u>Ke_{Las Nel}</u>
0.520	0.9998	1.0053
0.560	1.0223	1.0231
0.735	1.0268	1.0194
0.792	1.0313	1.0227
1.040	1.0314	1.0266

For the 0.56-inch pitch, the LASER and LEOPARD results are in close agreement. For the other lattices, the difference in calculated reactivity using the two methods is between 0.5% and 1% Δk/k.

5.2.3.2 Standard Deviation of Results

As a means of comparing the correlation of analysis with experiment, the standard deviation about the mean of the calculated k_{eff} was determined.

Comparisons were made for the following:

1. Five Hanford $\text{PuO}_2\text{-UO}_2$ experiments
2. Five WREC $\text{PuO}_2\text{-UO}_2$ critical experiments
3. The Hanford and WREC $\text{PuO}_2\text{-UO}_2$ experiments combined
4. Two WREC UO_2 critical experiments.

The calculated standard deviations are summarized in Table 5.4. The results indicate the LASER code is preferred from the standpoint of a smaller spread in the calculations.

TABLE 5.4
Standard Deviations of the Reactivity
Results Using Leonard Cross-Sections

<u>$\text{PuO}_2\text{-UO}_2$ Fuel</u>		
	<u>LEOPARD_{Leon}</u>	<u>LASER_{Leon}</u>
σ_{Hanford}	5.26×10^{-3}	2.73×10^{-3}
σ_{WREC}	11.75×10^{-3}	7.42×10^{-3}
$\sigma_{\text{HAN and WREC}}$	9.26×10^{-3}	6.19×10^{-3}
<u>UO_2 Fuel</u>		
	<u>LEOPARD</u>	<u>LASER</u>
σ_{WREC}	2.31×10^{-3}	0.974×10^{-3}

5.2.3.3 Explanation of the Differences Between LASER and LEOPARD Results

The heterogeneous thermal treatment of the unit cell in LASER leads to a spatially varying spectrum which in most of the cases considered is harder in all regions of the cell than the mean spectrum given by LEOPARD, particularly for the cores with plutonium. This spatially varying spectrum and the use of the Nelkin kernel instead of the Free-Gas kernel for light water leads to different reaction rates in LASER than with LEOPARD. The differences in Pu-239 and Pu-240 reaction rates and in the thermal leakage account for the differences between the LASER and LEOPARD reactivity results. The relative Pu-239 and Pu-240 reaction rates calculated by the two codes for both the Hanford and WREC experiments are included in Table 5.5. The resulting reactivity effect for differences in reaction rate and leakage are listed in Table 5.6. The relative Pu-240 reaction rates are also shown graphically in Figure 5.2.

5.2.4 Calculations with Different Sets of Thermal Parameters

During the study, a number of different sets of thermal parameters for U-235, Pu-239, and Pu-241 were used in three slightly different versions of the LEOPARD code. A summary of the various thermal parameters and the particular version of the code in which they were used is included in Table 5.7.

The results obtained with LEOPARD using the different thermal parameters are summarized in Table 5.8 while LASER results are summarized in Table 5.9. The variation in LEOPARD results are due primarily to the change in cross-sections, however, there is a small effect due to differences in the code. The standard deviation of the various results are summarized

Table 5.5

Absorption, Capture and Fission Reaction Rates Normalized
to One Atom for Hanford and WREC Lattices

Lattice Pitch (in)	R _{Fis} Pu-239 Leopard (1 atom)	R _{Fis} Pu-239 Laser Nel (1 atom)	R _{cap} Pu-239 Leopard (1 atom)	R _{cap} Pu-239 Laser Nel (1 atom)	R _{abs} Pu-239 Leopard (1 atom)	R _{abs} Pu-239 Laser Nel (1 atom)	R _{abs} Pu-240 Leopard (1 atom)	R _{abs} Pu-240 Laser Nel (1 atom)	Pu-239 Cell Averaged No. Densities ($\times 10^{-24}$)	Pu-240 Cell Averaged No. Densities ($\times 10^{-24}$)
<u>Hanford</u>										
0.55	75.60	75.25	36.20	36.41	111.80	111.66	118.41	109.89	1.2083 10^{-4}	1.0322 10^{-5}
0.60	101.00	100.26	46.47	46.36	147.47	146.62	125.24	122.84	1.0157 10^{-4}	8.6765 10^{-6}
0.71	151.15	149.89	65.81	65.58	216.96	215.47	132.28	140.52	7.2513 10^{-5}	6.1946 10^{-6}
0.80	185.48	183.93	78.62	78.32	264.10	262.25	134.70	149.31	5.7127 10^{-5}	4.8801 10^{-6}
0.90	216.87	215.52	90.15	89.95	307.02	305.47	135.99	156.28	4.5140 10^{-5}	3.8562 10^{-6}
<u>WREC</u>										
0.52	29.97	30.11	15.19	15.21	45.16	45.32	59.66	51.78	4.4345 10^{-4}	4.2160 10^{-5}
0.56	37.55	37.90	18.66	18.64	56.61	56.54	64.45	57.34	3.8261 10^{-4}	3.6373 10^{-5}
0.735	71.77	70.96	32.43	32.11	104.20	103.07	75.14	71.37	2.2196 10^{-4}	2.1102 10^{-5}
0.792	81.66	81.07	36.12	36.04	117.78	117.11	76.83	74.37	1.9132 10^{-4}	1.8188 10^{-5}
1.04	118.50	117.97	50.06	50.00	168.56	167.97	80.58	82.96	1.1085 10^{-4}	1.0538 10^{-5}

Table 5.6

Reactivity Values Associated with Differences in Pu-239
and Pu-240 Reaction Rates and Differences in Thermal Leakage

<u>Lattice Pitch (in)</u>	<u>$\Delta K_{\text{Pu-239}}$ (LEOP→LASER)</u>	<u>$\Delta K_{\text{Pu-240}}$ (LEOP→LASER)</u>	<u>Leakage (LEOP→LASER)</u>
Hanford			
0.55	-.34%	+.29%	-.29%
0.60	-.68%	+.06%	-.35%
0.71	-.44%	-.15%	-.41%
0.80	-.44%	-.22%	-.44%
0.90	-.34%	-.26%	-.38%
WREC			
0.52	+.19%	+.77%	-.16%
0.56	-.12%	+.58%	-.13%
0.735	-.59%	+.17%	-.03%
0.792	-.46%	+.10%	-.08%
1.04	-.32%	-.07%	-.05%

Figure 5.2 - Adsorption Reaction Rate per Atom for Pu-240 Determined Using LEOPARD and LASER Codes

TABLE 5.7

Cross Section Parameters at 2200 M/Sec for
Three Cross Section Sets

		Cross Sections (2200 m/sec)		
		Leonard*	Wescott Geneva** Conf. 1964	Sher (1965)***
U-235	σ_a	679.1	678.4 ± 1.9	$679 \pm 0.30\%$
	σ_f	580.5	577.5 ± 1.6	$578 \pm 0.35\%$
	α	0.1699	0.1748 ± 0.0018	$0.195 \pm 0.19\%$
	ν	2.4388	2.44242 ± 0.0066	$2.442 \pm 0.25\%$
	η	2.0846	2.0790 ± 0.0055	$2.078 \pm 0.24\%$
Pu-239	σ_a	1008.2	1010.6 ± 4.3	$1015 \pm .40\%$
	σ_f	752.8	745.9 ± 3.3	$742 \pm 0.47\%$
	α	0.3393	0.3548 ± 0.0047	$0.369 \pm 0.46\%$
	ν	2.8904	2.8759 ± 0.0020	$2.898 \pm 0.38\%$
	η	2.1582	2.1227 ± 0.0089	$2.116 \pm 0.43\%$
Pu-241	σ_a	1371.2	1376.1 ± 24.7	
	σ_f	963.2	1012.7 ± 6.7	
	α	0.4235	0.3589 ± 0.0252	
	ν	3.0209	2.9779 ± 0.0205	
	η	2.1221	2.1913 ± 0.0439	

*Used in LEOPARD II code

**Used in revised LEOPARD II code

***Used in LEOPARD V with Wescott Pu-241 cross sections

Table 5.8

LEOPARD Reactivity Comparisons with Different Thermal Cross-Sections

<u>Lattice Pitch (in)</u>	<u>LEOPARD (Leonard)</u>	<u>LEOPARD (Westcott Geneva 1964)</u>	<u>LEOPARD (Sher 1965)</u>
<u>Hanford</u>			
0.55	1.0165	1.0039	1.0049
0.60	1.0240	1.0102	1.0102
0.71	1.0314	1.0170	1.0155
0.80	1.0297	-	1.0129
0.90	1.0272	1.0133	1.0098
<u>WREC</u> *			
0.52	0.9998	0.9890	0.9892
0.56	1.0223	1.0103	1.0101
0.735	1.0268	1.0128	1.0114
0.792	1.0313	1.0175	1.0159
1.04	1.0314	1.0167	1.0143

* Aluminum mid-plate omitted.

Table 5.9

LASER Reactivity Comparisons with Different Thermal Cross Sections

<u>Lattice Pitch (in)</u>	<u>Fuel Rod Diameter (in)</u>	<u>Moderator/ Fuel Vol. Ratio</u>	<u>H/Pu Atom Ratio</u>	<u>LASER (NELKIN, LEONARD)</u>	<u>LASER (NELKIN, WESTCOTT)</u>	<u>LASER (NELKIN, WINFRITH)</u>	<u>LASER (NELKIN SHER, 1965)</u>
<u>Hanford</u>							
0.55	0.372	1.10	230	1.0095	1.0002	0.9923	1.0009
0.60	0.372	1.56	326	1.0126	1.0029	0.9946	1.0026
0.71	0.372	2.71	567	1.0170	1.0068	0.9983	1.0058
0.80	0.372	3.79	794	1.0156	1.0053	0.9968	1.0043
0.90	0.372	5.14	1077	1.0159	1.0056	0.9971	1.0044
* <u>WREC</u>							
0.52	0.337	1.68	76	1.0053	0.9970	0.9897	0.9964
0.56	0.337	2.16	98	1.0231	1.0140	1.0062	1.0114
0.735	0.337	4.70	211	1.0194	1.0092	1.0007	1.0067
0.792	0.337	5.67	254	1.0227	1.0122	1.0036	1.0094
1.04	0.337	10.80	486	1.0266	1.0158	1.0069	1.0139

* Aluminum mid-plate omitted.

in Table 5.10. These calculations show that the use of the LASER program with a Nelkin kernel for light water and either the 1965 SHER cross-sections or the Winfrith cross-sections give the best agreement with experimental results. This conclusion is based on a comparison of both the mean value for k_{eff} and standard deviation (σ) values. It can be seen that the total reactivity confidence interval corresponding to 3σ is decreased from 1% in reactivity going from LEOPARD to LASER and from LEONARD cross-sections to 1965 SHER or Winfrith cross-sections. This reduction of 3σ is due primarily to the use of the LASER program rather than to the choice of the thermal cross-sections. The reduction in discrepancy in the mean value of k_{eff} is due equally to the choice of the cross-sections and to the use of LASER rather than LEOPARD.

5.2.5 Method Selection

In selecting a method for use in following the operating history of Saxton Core II, both LASER and LEOPARD were considered. While the LASER results were in better agreement with the combined Hanford and WREC experiments, the difference between the two codes was small for the WREC experiments alone. For the lattice corresponding to the H/Pu ratio of the Saxton design, the LEOPARD calculated k_{eff} was slightly closer to the experimental value than that obtained with LASER. Consequently, the code selection was based primarily on the ease with which the code could be applied and the costs involved in its use. Since LEOPARD is a production-oriented code that lends itself to the repetitive calculations required in following the operating history of the design core, LEOPARD was selected.

Table 5.10

Comparison of the Different Standard Deviations Determined
from LEOPARD and LASER Calculations

Cross Sections	σ LEOPARD	σ LASER	Lattices	$\bar{K}_{Leop}$	$\bar{K}_{Laser}$
LEONARD	$5.26 \cdot 10^{-3}$	$2.73 \cdot 10^{-3}$	HANFORD	1.0258	1.0141
"	$11.75 \cdot 10^{-3}$	$7.92 \cdot 10^{-3}$	WREC	1.0223	1.0194
"	$9.26 \cdot 10^{-3}$	$6.19 \cdot 10^{-3}$	H & W*	1.02404	1.0168
WESTCOTT	$4.81 \cdot 10^{-3}$	$2.55 \cdot 10^{-3}$	HANFORD	1.0111	1.0039
"	$10.33 \cdot 10^{-3}$	$6.69 \cdot 10^{-3}$	WREC	1.0091	1.0096
"	$8.40 \cdot 10^{-3}$	$5.99 \cdot 10^{-3}$	H & W	1.0100	1.0071
SHER 1965	$3.54 \cdot 10^{-3}$	$1.69 \cdot 10^{-3}$	HANFORD	1.0107	1.0036
"	$9.57 \cdot 10^{-3}$	$6.17 \cdot 10^{-3}$	WREC	1.0076	1.0076
"	$7.37 \cdot 10^{-3}$	$4.93 \cdot 10^{-3}$	H & W	1.0092	1.0056
Winfrith	-	$2.13 \cdot 10^{-3}$	HANFORD	-	0.9958
"	-	$6.25 \cdot 10^{-3}$	WREC	-	1.0014
"	-	$5.40 \cdot 10^{-3}$	H & W	-	0.9986

* Combination of the Hanford and WREC data.

During the WREC critical program, a revised version of LEOPARD II was used with Wescott 1964 Geneva Conference cross-sections. Later a new version, LEOPARD V, became available. This version contained the changes incorporated in the revised LEOPARD and the more recent set of cross-sections by SHER (1965). To compare the revised LEOPARD with the LEOPARD V code, duplicate calculations were then carried out for a number of the WREC critical lattices. The comparison is summarized in Table 5.11 and shows that the two versions are essentially equivalent. Since the two versions of the code are equivalent and since LEOPARD V is expected to become the standard production code in the future, it was adopted for use in the analysis of Saxton Core II. The code was modified so that it could be used in a mechanized analysis sequence consisting of LEOPARD-V/BUBBLE/PDQ-3 (Section 4.2).

5.3 Buckling Measurements and Analysis

5.3.1 Experimental Scope

Critical buckling measurements were made for five different lattices with the $\text{PuO}_2\text{-UO}_2$ fuel and three different lattices for the conventional Saxton UO_2 fuel. The experimental procedure that was followed is outlined in Appendix F. The basic lattice used for a major part of the experimental program was that containing approximately the same H/Pu ratio as the Saxton design at temperature, the 0.56-inch lattice. In this lattice, the buckling was measured in two separate experiments as a check on the precision of the measurements. The buckling for a borated configuration containing 337 ppm boron was also measured in the 0.56-inch lattice. Table 5.12 contains a summary of the experimental results.

Table 5.11

A Comparison of Calculated k_{eff} Using the Revised LEOPARD and LEOPARD-5
Codes for the WREC Mixed-Oxide (PuO_2-UO_2) Critical Experiments*

Lattice Pitch (inches)	LEOPARD-5** Sher, 1965 Cross Sections	Revised LEOPARD Geneva 1964 Cross Sections
0.52	0.988913	0.98867
0.56	1.00959	1.00981
0.735	1.01093	1.01230
0.792	1.015017	1.01663
1.04	1.013812	1.01620

* k -bias = 1.0 for both sets of calculations and the effect of
of the aluminum mid-plate was also included.

** Calculations made by L. Strawbridge.

Table 5.12

BUCKLING AND REFLECTION SAVINGS

Lattice Loading	Critical No. of Fuel Rods	Critical Buckling		Reflector Savings		Water Temperature (°C)	Critical Water Height (cm.)
		Total (cm ⁻²) x 10 ³	Axial	Total (2x)	Axial (cm.)		
<u>UO₂, 5.74 w/o M-235, SS Clad</u>							
0.52" Pitch ⁺	449 (cylindrical)	11.76 ± 0.18	0.904 ± 0.01	13.73 ± 0.19	8.49 ± 0.27 (Bott.) 2.36 ± 0.27 (Top)	19.2	93.63
0.56" Pitch	361 (19x19)	12.71 ± 0.14	1.13 ± 0.01	14.31 ± 0.25	8.77 ± 0.14 (Bott.) 2.65 ± 0.05 (Top)	18.0	82.05
0.792" Pitch*	182 (13x14)	13.68 ± 0.19	1.05 ± 0.01	12.43 ± 0.41	7.78 ± 0.08 (Bott.) 0.21 ± 0.04 (Top)	17.3	88.96
<hr/>							
<u>PuO₂-UO₂, 6.6 w/o PuO₂, Zr-4 Clad</u>							
0.52" Pitch	506 (22x23)	10.88 ± 0.13	1.07 ± 0.01	15.76 ± 0.44 (22 Rods) 14.51 ± 0.37 (23 Rods)	8.56 ± 0.29 (Bott.) 4.86 ± 0.29 (Top)	25.8	82.90
0.56" Pitch ^Δ	361 (19x19)	12.21 ± 0.11	1.14 ± 0.01	14.82 ± 0.17	7.97 ± 0.57 (Bott.) 4.67 ± 0.15 (Top)	17.0	80.80
	361 (19x19)	12.09 ± 0.15	1.11 ± 0.01	15.37 ± 0.28	8.22 ± 0.34 (Bott.) 4.36 ± 0.34 (Top)	15.75	81.79
0.56" Pitch 337 ppm Boron	441 (21x21)	11.23 ± 0.10	0.964 ± 0.01	13.98 ± 0.18	8.38 ± 0.15 (Bott.) 4.99 ± 0.15 (Top)	18.0	86.06
0.735" Pitch	169 (13x13)	15.96 ± 0.19	1.61 ± 0.01	12.78 ± 0.25	6.83 ± 0.17 (Bott.) 3.27 ± 0.17 (Top)	24.1	68.41
0.792" Pitch	144 (12x12)	15.93 ± 0.22	1.34 ± 0.01	12.90 ± 0.20	6.47 ± 0.12 (Bott.) 3.05 ± 0.12 (Top)	16.1	76.76
1.04" Pitch ⁺	121 (11x11)	12.84 ± 0.14	1.25 ± 0.01	12.27 ± 0.10	6.23 ± 0.09 (Bott.) 3.09 ± 0.09 (Top)	19.9	79.50

Notes:

1. All pitches are square.
2. Water heights are referenced to the bottom of the fuel.
3. Bottom axial reflector is water and aluminum; top axial reflector is air and fuel.
4. One-quarter inch thick aluminum grid plate at 45.7 cm. from bottom of the fuel.
5. Errors are ± one standard deviation.
6. *Core was irradiated on a positive period. Positive reactivity was 0.15% Δk/k.
7. ^ΔTwo separate experiments performed.
8. ⁺Not part of the Saxton Plutonium Program, but included for completeness.

The number of fuel rods required to achieve criticality in a fully inundated cylindrical core was determined for both fuels for a number of different lattices. Table 5.13 summarizes the fuel rod requirements.

5.3.2 Analytic Results

The measured bucklings of Table 5.12 were used in the revised version of LEOPARD with the 1964 Geneva Conference cross-sections. These results are compared to an experimental k_{eff} of 1.0 in Table 5.14. As a further comparison of analysis with experiment, the calculated PDQ power distributions for the 0.56-inch lattice for both fuels was used to determine a calculated buckling using the same procedure that was used in processing the experimental power data. The following list compares the analytic and measured buckling and is useful in pointing out the magnitude of the buckling error that produces the observed discrepancy in reactivity.

<u>Lattice</u>	<u>Fuel</u>	<u>Calculated Radial B^2 (cm^{-2})</u>	<u>Measured Radial B^2 (cm^{-2})</u>
0.56	UO ₂	0.01110	0.01158
0.56	Pu-UO ₂	0.01074	0.01107

5.4 Power Peaking Effects

5.4.1 Experimental Scope

The effect on reactivity and local power peaking resulting from perturbations such as water slots and control rods was investigated in both single and multi-region cores. Table 5.15 summarizes the perturbation

Table 5.13

Number of Fuel Rods Required for Criticality for
Cylindrical Cores of Various Lattice Pitches

Fuel	Pitch* (inch)	No. of Fuel Rods To Produce Criticality	Moderator Temperature (°C)
5.74 w/o U-235 SS Clad	0.52 ⁺	447	17.6
	0.56	339	17.3
	0.792	173	17.3
6.6 w/o PuO ₂ Zr-4 Clad	0.52	472	24.0
	0.56	337	17.5
	0.735	151	28.0
	0.792	133	17.0
	1.04 ⁺	117	20.0

* All Pitches are Square.

⁺ Not Part of the Saxton Plutonium Program.

Table 5.14

Calculated k_{eff} for the WREC PuO_2-UO_2 and UO_2
 Critical Experiments Using the LEOPARD Code and
 the 1964 Geneva Conference Cross-Sections

<u>PuO_2-UO_2 Fuel</u>			
<u>Lattice Pitch (in)</u>	<u>H/Pu</u>	<u>Measured Buckling (cm^{-2}) $\times 10^3$</u>	<u>Calculated k_{eff}</u>
0.52	76	10.88 ± 0.13	0.9890
0.56	98	12.15 ± 0.08	1.0103
0.56*	98	11.23 ± 0.10	1.0148
0.735	211	15.96 ± 0.19	1.0128
0.792	254	15.93 ± 0.22	1.0175
1.040	486	12.84 ± 0.14	1.0167

<u>UO_2 Fuel</u>			
<u>Lattice Pitch (in)</u>	<u>H/U-235</u>	<u>Measured Buckling (cm^{-2}) $\times 10^3$</u>	<u>Calculated k_{eff}</u>
0.52	77	11.76 ± 0.18	0.9936
0.56	99	12.71 ± 0.14	1.0059
0.792	259	13.68 ± 0.19	1.0021

* 337 ppm boron

Table 5.15
Summary of Perturbation Experiments

Single-Region Cores						
Core Size		Fuel	Perturbation			Type Measurements
19 x 19		UO ₂	Water Slot in Center, 5 Rods Removed			Reactivity, Power
19 x 19		UO ₂	Aluminum Slab in Center, 5 Rods Removed			Reactivity, Power
21 x 21		UO ₂	5 Ag-In-Cd Control Rods in Center			Reactivity, Power
19 x 19		PuO ₂ -UO ₂	Water Slot in Center, 5 Rods Removed			Reactivity, Power
19 x 19		PuO ₂ -UO ₂	Aluminum Slab in Center, 5 Rods Removed			Reactivity, Power
21 x 21		PuO ₂ -UO ₂	5 Ag-In-Cd Control Rods in Center			Reactivity, Power
Multi-Region Cores						
Core Size	Region 1		Region 2 Fuel	Boron ppm	Perturbation	Type Measurements
	Size	Fuel				
19 x 19	11 x 11	PuO ₂ -UO ₂	UO ₂	0	Aluminum Slab at PuO ₂ -UO ₂ Boundary (5 PuO ₂ -UO ₂ Rods Removed)	Reactivity, Power
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	0	5 Control Rods at PuO ₂ -UO ₂ Boundary (5 PuO ₂ -UO ₂ Rods Removed)	Reactivity, Power
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	0	Water Slot as a Function of Position a) Center; b) PuO ₂ -UO ₂ Boundary; c) UO ₂ Boundary	Reactivity
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	0	5 Control Rods as a Function of Position a) Center; b) PuO ₂ -UO ₂ Boundary; c) UO ₂ Boundary; d) UO ₂ -2nd Row; e) UO ₂ -3rd Row	Reactivity
27 x 27	19 x 19	PuO ₂ -UO ₂	UO ₂	1453	Water Slot at PuO ₂ -UO ₂ Boundary (5 PuO ₂ -UO ₂ Rods Removed)	Reactivity, Power
27 x 27	19 x 19	PuO ₂ -UO ₂	UO ₂	1453	Aluminum Slab at PuO ₂ -UO ₂ Boundary (5 PuO ₂ -UO ₂ Rods Removed)	Reactivity, Power
19 x 19	11 x 11	UO ₂	PuO ₂ -UO ₂	0	Water Slot as a Function of Position a) Center; b) UO ₂ Boundary; c) PuO ₂ -UO ₂ Boundary	Reactivity
19 x 19	11 x 11	UO ₂	PuO ₂ -UO ₂	0	5 Control Rods as a Function of Position a) Center; b) UO ₂ Boundary; c) PuO ₂ -UO ₂ Boundary	Reactivity

experiments that were carried out and the type of measurements that were made in the 0.56-inch lattice.

In the single-region core, a water slot was formed by removing five fuel rods in a line in the center of the core. Power measurements were made in the adjacent rods before and after the water slot was formed. Experiments were also carried out with a 1/4-inch aluminum slab installed in the water slot to displace part of the water. In the multi-region cores, experiments were carried out with the water slot, aluminum slab, and control rods at the boundary between the two fuel regions.

5.4.2 Analytic Results

5.4.2.1 Single-Region Cores

Three different methods were used in the analysis of the initial UO_2 perturbation experiments in order to select a method for use in the evaluation of the remaining experiments of Table 5.15. In the first method, the group constants used for the water slots were obtained from a LEOPARD calculation using the material composition of the slot alone. These constants, designated soft spectrum constants, were determined by the use of a flux spectrum that was not representative of the spectrum existing in the slot since the influence of the adjacent fuel was not considered.

In the second method, constants for the slot were determined by defining a unit cell for the fuel rods surrounding the slot and including in the LEOPARD calculation an extra region composed of the

materials contained within the slot. Group constants for the slot were then determined from the group averaged microscopic cross sections and the number density of the slot materials.

In the third method, a different Dancoff calculation from that of the normal lattice was used for the fuel rods surrounding the slot to account for the reduction in fuel rod shadowing that occurs in the slot. Group constants for the slot were determined by the use of an extra region in LEOPARD.

The results obtained using the three methods for the single-region UO_2 experiments are compared in Table 5.16. The comparison of calculated and experimental power peaking was made for the rod nearest the slot since this is the position where the maximum error occurs. Both analysis and experiment are normalized to a point that is not influenced by the central perturbation. These comparisons show that the use of group constants determined from an extra region in LEOPARD improves both the power peaking and reactivity correlation, however, the use of a different Dancoff in the rods surrounding the slot has little effect. The discrepancy in power peaking using a soft spectrum is approximately twice that obtained using an extra region. In both cases, the analysis overpredicts power peaking.

Power distribution comparisons using extra-region group constants for the water slot and aluminum slab are shown in Figure 5.3. A complete summary of the power distribution and reactivity comparisons for the various methods is included in Tables E-1 and E-2 in Appendix E.

Table 5.16

A Comparison of the Discrepancy in Calculated and Measured Power Peaking
and Reactivity in UO_2 Perturbation Experiments for Different Analysis Methods

Power Peaking Comparisons (Analysis/Experiment)				
Perturbation	Soft-Spectrum*	Extra-Region*	Dancoff, Imperfect Lattice**	
			Soft Spectrum	Extra-Region
Water Slot	1.056	1.026	1.053	1.027
Aluminum Slab	1.010	1.003	-	-

Reactivity Comparisons					
Perturbation	Measured Reactivity % $\Delta k/k$	Soft-Spectrum* % $\Delta k/k$	Extra-Region* % $\Delta k/k$	Dancoff, Imperfect Lattice**	
				Soft Spectrum % $\Delta k/k$	Extra-Region % $\Delta k/k$
Water Slot	+ 0.26	+ 0.317	+ 0.233	+ 0.306	+ 0.223
Aluminum Slab	- 0.27	- 0.233	- 0.256	-	-

*Calculated with Leonard

**Calculated with 1964 Geneva Conference cross-sections.

There is no difference in the calculated reactivity change or power peaking effects for a perturbation in the UO_2 cores using the two cross-section sets.

Similar calculations were carried out for the single-region $\text{PuO}_2\text{-UO}_2$ perturbation experiments. The results comparing reactivity effects and the calculated and experimental power peaking for the rod nearest the slot are summarized in Table 5.17. Power distribution comparisons using the extra-region constants for the perturbation are shown in Figure 5.4. Tables E-3 and E-4 in Appendix B contain a summary of the reactivity and power distribution results for the various methods of analysis.

The single-region perturbation experiments and the associated analysis illustrate the following:

1. Power peaking is greater in the cores containing $\text{PuO}_2\text{-UO}_2$ fuel than in those containing UO_2 . For the water slot experiments, the power peaking in $\text{PuO}_2\text{-UO}_2$ is approximately 6% higher than that in UO_2 .
2. The discrepancy between analysis and experiment is also greater in the $\text{PuO}_2\text{-UO}_2$ configurations than in those containing UO_2 .

Power peaking in the design core is expected to occur in the central plutonium region at the beginning of life. The observed discrepancy between analysis and experiment indicates the design calculation is conservative.

Additional power peaking information is available for the multi-region configurations of Table 5.15. This information is included in the

Table 5.17

A Comparison of the Discrepancy in the Calculated and Measured Power Peaking
and Reactivity in $\text{PuO}_2\text{-UO}_2$ Perturbation Experiments for Different Methods of Analysis

Power Peaking Comparisons
(Analysis/Experiment)

Perturbation	Soft-Spectrum	Extra-Region	Dancoff, Imperfect Lattice	
			Soft-Spectrum	Extra-Region
Water Slot	1.075	-	1.073	1.042
Aluminum Slab	1.047*	1.031	-	-

Reactivity Comparisons

Perturbation	Measured Reactivity % $\Delta k/k$	Soft-Spectrum % $\Delta k/k$	Extra-Region % $\Delta k/k$	Dancoff, Imperfect Lattice	
				Soft-Spectrum % $\Delta k/k$	Extra-Region % $\Delta k/k$
Water Slot	+ 0.49	+ 0.597	-	+ 0.577	+ 0.49
Aluminum Slab	- 0.04	- 0.05*	- 0.09	-	-

*Leonard Cross-Sections

discussion of the reactivity measurements and analysis, Section 5.6, and power sharing, Section 5.5.

5.5 Power Sharing

5.5.1 Experimental Scope

The relative power produced in each of the two fueled regions in the Saxton design is important in establishing the maximum power level at which the core can operate. If more power than expected is produced in the central plutonium region, it will be necessary to reduce the total core power to avoid exceeding an imposed hot-spot limitation. Therefore power measurements were made in various multi-region configurations during the WREC critical program. Table 5.18 summarizes the extent of the power sharing experiments. Also included in Table 5.18 are four multi-region experiments designed to investigate the effect of various perturbations such as water slots and control rods. These experiments also provide power sharing information.

5.5.2 Power Measurements

In configurations composed of different types of fuel, it is not possible to determine the correct relative power in each fuel by a gamma scan of fuel rods if the gamma source and decay characteristics of the two fuels are different. Such is the case with uranium and plutonium fuels.

Therefore, to translate measured gamma activity to fuel rod power, two different methods were developed. The two methods are described in detail in WCAP-3385-54. In the first method, designated the thermal method,

TABLE 5.18
SUMMARY OF THE MULTI-REGION EXPERIMENTS
Power Sharing Experiments

Configuration	Region 1			Region 2		Core Size	Boron ppm	Date of Experiment	Foil* Data	Description
	Size	Fuel	Rods	Fuel	Rods					
-- $\frac{1(a)}{1(b)}$ --	11x11	PuO ₂ -UO ₂	121	UO ₂	240	19x19	0	4/29/65 6/03/65	U-238 } Np-239 } Dys	Reference Core - clean, two power maps were made in this configuration
2	19x19	PuO ₂ -UO ₂	361	UO ₂	368	27x27	1,453	5/14/65	U-238 } Np-239 } Dys	Reference Core - borated
3	19x19	PuO ₂ -UO ₂	352	UO ₂	368	27x27	1,425	5/21/65		3x3 UO ₂ insert in PuO ₂ -UO ₂ region
4	11x11	UO ₂	121	PuO ₂ -UO ₂	240	19x19	0	5/06/65	U-238 } Np-239 } Dys	Reference Inverted Core - clean
5	19x19	UO ₂	361	PuO ₂ -UO ₂	368	27x27	1,252	5/28/65	U-238 } Np-239 } Dys	Reference Inverted Core - borated
6	3x3	PuO ₂ -UO ₂	9	UO ₂	352	19x19	0	4/09/65		PuO ₂ -UO ₂ Insert Experiment
7	3x3	UO ₂	9	PuO ₂ -UO ₂	352	19x19	0	4/26/65	Dys	UO ₂ Insert Experiments - also used to compare power produced in Vipac and pelletized fuel
8	19x19	PuO ₂ -UO ₂ (UO ₂)	339 (18)	UO ₂	367	27x27	1,425	5/21/65	Mn Wire	L shaped UO ₂ insert in PuO ₂ -UO ₂ region - Mn wire activation in simulated Saxton des. core

Power Peaking Experiments

9	11x11	PuO ₂ -UO ₂	116	UO ₂	240	19x19	0	4/30/65		5 PuO ₂ -UO ₂ rods removed at fuel region interface = aluminum slab installed
10	11x11	PuO ₂ -UO ₂	116	UO ₂	320	21x21	0	5/04/65		5 PuO ₂ -UO ₂ rods removed at fuel region interface = 5 ² Ag-In-Cd rods installed
11	19x19	PuO ₂ -UO ₂	356	UO ₂	368	27x27	1,453	5/14/65		5 PuO ₂ -UO ₂ rods removed at fuel region interface = water slot formed
12	19x19	PuO ₂ -UO ₂	356	UO ₂	368	27x27	1,453	5/17/65		5 PuO ₂ -UO ₂ rods removed at fuel region interface = aluminum slab installed

*U-238 foils counted for fission activity and Np-239 decay

thermally insulated fuel rods were irradiated in a low power critical facility (CES) and the temperature rise of thermocouples attached to the fuel rod cladding was measured. Then, after shutdown, the same fuel rods were gamma scanned and the ratio of heat rate to gamma activity as a function of time after shutdown was determined. In the second method, designated the foil method, foils composed of two different foil materials were irradiated and subsequently gamma scanned. Later, the number of fission events occurring in each foil was determined by a gamma count of the I_{a-140} activity. Time dependent ratios were developed for relating gamma activity to rod power as shown in Figure 5.5. The curve of the ratio for the foil method is similar in shape but approximately 5% below that of the thermal method. At this time there is no reason for selecting one method in preference to the other. Consequently, analytic comparisons were made with both methods.

5.5.3 Analytic Results

5.5.3.1 Power Sharing Experiments

The analysis of the multi-region experiments listed in Table 5.18 was carried out using the LEOPARD-PDQ-3 sequence and the 1964 Geneva Conference cross-sections. A comparison of the PDQ-3 analysis with experiment for seven power sharing configurations (Configurations 1 through 7, Table 5.18) is shown in Figures 5.6 through 5.13. These figures compare the experimental power distribution with the analysis along the flats and diagonals of each core as determined by both the thermal and foil method of relating gamma activity to rod power. In these figures, (pages 5-41 through 5-48) the curves represent the point-

wise power distribution and are used to illustrate the shape of the power curve near region boundaries. The horizontal line segments represent the integral of the analytic power over a region corresponding to one fuel rod. These integrated values should be directly comparable to the experimentally determined rod power.

For the reference cores with an inner region of $\text{PuO}_2\text{-UO}_2$, Configuration 1(b) and Configuration 2, both two-group and four-group calculations were made. A comparison of the two methods is included in Figure 5.6 and 5.8 for a traverse along the flats of both cores. The comparison shows a slight improvement in the correlation between calculated and measured power when a four-group analysis is used.

The results of the power sharing studies show good agreement between analysis and experiment for both the foil and heat rate methods of translating the gamma measurements to rod power. The difference between the measured and calculated values averages approximately 1% using the foil method and approximately 5% for the heat rate method. Based on either result, little adverse effect on the performance of the design core is expected as a result of a discrepancy in power sharing .

Additional details concerning the power sharing experiments including the core diagrams and the measured power for the fuel rods in each configuration are included in Appendix E.

Figure 5.6 - A Comparison of Calculated and Measured Power Distributions in a 19 x 19 Core with an 11 x 11 $\text{PuO}_2\text{-UO}_2$ Inner Region and a UO_2 Outer Region

Figure 5.7 - A Comparison of Calculated and the Averaged Measured Power Distributions from Configurations 1(a) and 1(b)

Figure 5.8 - A Comparison of Calculated and Measured Power Distributions in a 27 x 27 Core with a 19 x 19 $\text{PuO}_2\text{-UO}_2$ Inner Region and a UO_2 Outer Region

Figure 5.9 - A Comparison of Calculated and Measured Power Distributions in a 27 x 27 Core with a 19 x 19 $\text{PuO}_2\text{-UO}_2$ Inner Region Containing a 3 x 3 UO_2 Insert and a UO_2 Outer Region

Figure 5.10 - A Comparison of Calculated and Measured Power Distributions in a 19 x 19 Core with an 11 x 11 UO_2 Inner Region and a PuO_2-UO_2 Outer Region

Figure 5.11 - A Comparison of Calculated and Measured Power Distributions in a 27 x 27 Core with a UO_2 Inner Region and a $\text{PuO}_2\text{-UO}_2$ Outer Region

Figure 5.12 - A Comparison of Calculated and Measured Power Distributions in a 19 x 19 UO_2 Core with a 3 x 3 $\text{PuO}_2\text{-UO}_2$ Insert

Figure 5.13 - A Comparison of Calculated and Measured Power Distributions in a 19 x 19 $\text{PuO}_2\text{-UO}_2$ Core with a 3 x 3 UO_2 Insert

5.5.3.2 Manganese Flux Wire Experiment

Fuel rod power in the Saxton reactor is normally determined by manganese wire activation measurements that are used in conjunction with appropriate PDQ calculations. The local rod power is determined by multiplying the measured flux wire activation by the ratio of calculated rod power to calculated wire activation. The calculations to normalize the data and determine power from the various flux wire measurements are carried out using the INCORE computer code.

An experiment was carried out during the critical program to determine if two differently fueled regions would lead to difficulty in interpreting the flux wire measurements. The core consisted of a 19×19 PuO_2 - UO_2 inner region with a UO_2 outer region forming a 27×27 core overall. To mock-up the center element of the Saxton design reactor, L-shaped UO_2 inserts simulating control rod fuel followers were installed. Aluminum tubes containing manganese wire inserts were installed at positions approximating the flux wire thimble locations in the design core. Figure 5.14 is a diagram of the experimental configuration. The PDQ-3 analysis for this configuration was carried out using extra-region constants for the water and aluminum of the simulated flux wire thimbles. The experimental and calculated manganese activation and core power distributions are compared in Table 5.19.

Configuration: 27 x 27 Core
 19 x 19 $\text{PuO}_2\text{-UO}_2$ Inner Region
 UO_2 Outer Region

Figure 5.14 Core Diagram for Two-Region Power Shaping Experiment
 Containing Simulated Fuel Followers and Flux Wire Thimbles.

Table 5.19

Special Power Sharing Experiment - Analysis and Experiment

<u>Mn Wire</u>			<u>PuO₂-UO₂ Fuel Rods</u>			<u>UO₂ Fuel Rods</u>			
<u>No.</u>	<u>Experiment</u>	<u>Analysis</u>	<u>No.</u>	<u>Experiment</u>	<u>Analysis</u>	<u>Experiment</u>		<u>Analysis</u>	
						<u>Foil Method</u>	<u>Thermal Method</u>		
26	1.426	1.426	21	1.000	1.000	67	0.302	0.286	0.321
27	1.031	1.015	20	1.052	1.033	68	0.277	0.263	0.253
28	1.344	1.363	19	1.061	1.093	69	0.323	0.307	0.303
29	1.144	1.115	18	0.762	0.742	70	0.381	0.361	0.375
30	1.192	1.174	17	0.998	0.992	71	0.591	0.560	0.617
			13	0.951	0.928	72	0.567	0.538	0.554
			9	0.807	0.803	73	0.421	0.399	0.411
			6	0.643	0.640	74	0.522	0.495	0.496
			4	0.552	0.542	75	0.573	0.543	0.565
			10	1.049	1.085	76	0.593	0.563	0.573
			7	0.826	0.804	77	-	-	0.560
			5	0.763	0.745	78	0.755	0.716	0.754
			11	0.706	0.693	79	0.810	0.768	0.794
			12	0.792	0.770	80	0.792	0.751	0.777
			14	0.974	0.969	81	0.796	0.755	0.784
			15	0.756	0.740	82	0.780	0.740	0.760
			16	0.836	0.807				
			23	-	0.729				
			34	0.848	0.873				
			22	0.857	0.856				

The procedure to be followed in interpreting the activation measurements in Saxton Core II was applied to the data obtained in the WREC flux wire experiment. Figure 5.15 compares the measured fuel rod power with that which would be determined from the activation measurements, the PDQ-3 calculation, and the use of the INCORE code. The figure shows the per cent difference between the INCORE results and the power measurements using both the foil and thermal methods of normalization. Standard deviations between the power measurements and INCORE were calculated as:

$$\sigma = \frac{\sum (D_i)^2}{n-1}^{1/2}$$

$$n = \text{fuel rod measurement}$$

$$D_i = \frac{\text{INCORE-Measurement}}{\text{Measurement}}$$

with the following results:

foil method

$$\sigma_{\text{overall}} = 2.9\%, \quad \sigma_{\text{UO}_2} = 3.9\%, \quad \sigma_{\text{PuO-UO}_2} = 2.1\%$$

thermal method

$$\sigma_{\text{overall}} = 3.9\%, \quad \sigma_{\text{UO}_2} = 5.5\%, \quad \sigma_{\text{PuO}_2\text{-UO}_2} = 2.9\%.$$

$$\left(\frac{\text{INCOFE-Meas.}}{\text{Meas.}} \right) \times 100\%$$

5-53

Figure 5.15 - Comparison of Measured Relative Fuel Rod Power with Incore Results

Based on an evaluation of this experiment, it was concluded that the method to be used was satisfactory and that no modification was necessary for interpreting the Saxton Core II flux wire data.

This experiment also provided information as to power peaking in the $\text{PuO}_2\text{-UO}_2$ fuel rods near the simulated flux wire thimbles. The comparative power data of Table 5.19 for rods near the flux wires, Rods 19, 10, 34, and 22, shows the calculated power is approximately 3% higher than that measured. In this calculation extra-region constants were used for the simulated flux wire thimbles. With soft-spectrum constants, as used in the design core calculations, the power peaking would be approximately 5% higher than that measured.

5.5.3.3 Perturbation Experiments

Configurations 9-12 of Table 5.18 were designed primarily to investigate power peaking effects; however, they also provide power sharing information. Figures 5.16 through 5.19 compare the PDQ-3 analysis with experiment. The results show an overprediction of local power peaking which is consistent with the results of the single region comparisons.

Additional detail concerning these experiments is included in Appendix E.

Figure 5.16 - A Comparison of Calculated and Measured Power Distributions Across a 19 x 19 Core with an 11 x 11 $\text{PuO}_2\text{-UO}_2$ Inner Region, a UO_2 Outer Region, and an Aluminum Slab at the Fuel Region Boundary

Figure 5.17 - A Comparison of Calculated and Measured Power Distributions Across a 21 x 21 Core with an 11 x 11 PuO₂-UO₂ Inner Region, a UO₂ Outer Region, and Control Rods Installed at the Fuel Region Boundary

Figure 5.18 - A Comparison of Calculated and Measured Power Distributions Across a 27 x 27 Core with a 19 x 19 $\text{PuO}_2\text{-UO}_2$ Inner Region, a UO_2 Outer Region, and a Water Slot at the Fuel Region Boundary

Figure 5.19 - A Comparison of Calculated and Measured Power Distributions in a 27 x 27 Core with a 19 x 19 $\text{PuO}_2\text{-UO}_2$ Inner Region, a UO_2 Outer Region, and an Aluminum Slab at the Region Boundary

5.6 Reactivity Measurements and Analysis

5.6.1 Experimental Scope

During the experimental program, extensive reactivity measurements were made in both single and multi-region configurations. In general, the reactivity measurements can be grouped into four categories:

1. Reactivity worth of changes in core geometry as in step changes in loading from a one-zone to a two-zone core.
2. Reactivity worth of core perturbations such as water slots.
3. Reactivity worth of control rods and soluble poison.
4. Temperature coefficient measurements in single and multi-region cores.

The interpretation of experimental results in configurations containing fuels with different kinetic characteristics may result in a larger error than if a single fuel type were used. For this reason, a complete description of the experimental procedure and experimental results for the various configurations is included in Appendix F. The analysis on which the reactivity interpretation is based is described in the following paragraphs.

5.6.2 Reactivity - Period Characteristics

To determine reactivity from critical experiment period measurements, it was necessary to determine β_{eff} , the effective fraction of delayed neutrons.

For single-region cores, β_{eff} was determined from LEOPARD calculations using the following two equations:

$$\bar{\beta} = \frac{\sum_j \sum_i \beta_{ij} (\bar{v} \Sigma_{fj} \bar{\phi})}{\sum_j (\bar{v} \Sigma_{fj} \bar{\phi})}$$

$$\beta_{\text{eff}} = I \cdot \bar{\beta}$$

where

β_i = the delayed neutron fraction for each group i , 6 delayed groups were used

j = number of fissional isotopes

I = an importance factor to account for difference in worth of the delayed and prompt neutrons.

The importance factor is the ratio of the calculated LEOPARD multiplication factor using the delayed neutron spectrum to that determined using the prompt neutron spectrum. The prompt neutron lifetime is also obtained from LEOPARD.

The kinetic parameters for the single-region cores are summarized in the following list:

Kinetic Parameter*	UO ₂ Fuel	PuO ₂ -UO ₂ Fuel
$\bar{\beta}$	0.00696	0.00293
I	1.144	1.155
β_{eff}	0.00796	0.00338
l , micro-seconds	14.49	8.15

* 500 ppm boron

The calculated kinetic parameters were used in the Inhour code to determine the desired reactivity-period relationships. A comparison of the reactivity vs. period curves for the single-region cores is shown in Figure 5.20.

The kinetic characteristics for the two-region cores composed of different types of fuel distributed spatially were computed by the following procedure. Group constants for each region were determined using LEOPARD for both the prompt and delayed spectra. Then, a two-group cylindrical AIM-5 calculation was made using the prompt group constants. A buckling search was used to adjust the transverse buckling until a calculated k_{eff} of 1.0 was obtained. A second AIM-5 calculation was then run in three groups using the previously determined transverse buckling. In this calculation the groups consisted of a prompt group, a delayed group, and a thermal group. In each region the prompt neutron source fraction was set equal to one while the delayed neutron source fraction was set equal to β for the particular fuel composition. In the calculation no transfers occurred from prompt group to delayed group. Instead, both prompt and delayed groups transferred directly to the thermal group. The resulting calculated k_{eff} equals $1 + \beta_{\text{eff}}$. A core averaged $\bar{\beta}$ was then determined from an integration of the β -weighted neutron source fraction over the core. An importance factor was determined from the equation $I = \beta_{\text{eff}}/\bar{\beta}$. Table 5.20 summarizes the calculated kinetic parameters for the clean configurations loaded during the experimental program. The reactivity vs. period curves for a number of two region 19×19 clean configurations are compared to similar curves for the single-fuel configurations in Figure

Figure 5.20

Table 5.20

Calculated Kinetic Parameters for WREC Critical Configurations Containing
 PuO_2 - UO_2 and UO_2 Fuels

Configurations in ² 0.56" pitch square lattice	β_1	β_2	β_3	β_4	β_5	β_6	β	β_{eff}	I	ℓ (v-sec)
11x11 PuO_2 - UO_2 19x19 overall	0.000167	0.001187	0.001051	0.002092	0.000685	0.000254	0.005435	0.005836	1.07378	12.80
15x15 PuO_2 - UO_2 19x19 overall	0.000133	0.000997	0.000852	0.001640	0.000546	0.000209	0.004376	0.004627	0.94577	12.99
19x19 PuO_2 - UO_2							0.002913	0.003379	1.160	8.50
11x11 PuO_2 - UO_2 21x21 overall	0.000173	0.001223	0.001089	0.002178	0.000711	0.000969	0.005636	0.006091	1.0808	13.14
11x11 UO_2 19x19 overall	0.000128	0.000966	0.000819	0.001566	0.000523	0.000202	0.004203	0.005334	1.2683	10.70
15x15 UO_2 19x19 overall	0.000160	0.001148	0.001010	0.001999	0.000656	0.000244	0.005216	0.006558	1.2572	12.43
19x19 UO_2							0.006943	0.007940	1.144	15.38
11x11 UO_2 21x21 overall	0.000122	0.000934	0.000787	0.001492	0.000501	0.000195	0.004031	0.005131	1.2729	10.40

5.20. Figure 5.21 shows the variation of β_{eff} with fuel loading and position. Reactivity vs. period curves for two 21 x 21 two-region cores are shown in Figure 5.22.

Similar calculations were made for the borated reference core consisting of a 19 x 19 $\text{PuO}_2\text{-UO}_2$ inner region with 368 UO_2 rods in an outer region forming a 27 x 27 rod core. The resulting reactivity-period curve for this configuration is shown in Figure 5.23. The calculations were repeated for the inverted configuration in which the UO_2 is installed in the inner region with the $\text{PuO}_2\text{-UO}_2$ installed in the outer region. The following list compares the calculated kinetic characteristics for each core. Also included are the values of β for each region.

<u>Core 1 ($\text{PuO}_2\text{-UO}_2$ Inside)</u>	<u>Core 2 (UO_2 Inside)</u>	
$\beta_{\text{eff}} = 0.00406$	$\beta_{\text{eff}} = 0.00623$	$\beta, \text{PuO}_2\text{-UO}_2 = 0.00294$
$\bar{\beta} = 0.00429$	$\bar{\beta} = 0.00530$	$\beta, \text{UO}_2 = 0.00698$
$I = 0.94441$	$I = 1.175$	
$\ell = 9.37 \mu \text{ sec}$	$\ell = 10.75 \mu \text{ sec}$	

5.6.3 Reactivity Measurements

The experimental procedure for determining reactivity from critical water height measurements is described in Appendix F for both single and multi-region configurations.

For the single-region cores, curves of installed reactivity available vs. just critical moderator height were developed as shown in Figure F-3. The reactivity worth of a core change was determined from the difference in available reactivity corresponding to the critical water height for the unperturbed and perturbed cores.

For the multi-region cores, data from a number of multi-region configurations were used in constructing the single $\delta\rho/\delta h$ curve shown in Figure F-4 and used in evaluating the reactivity change associated with a change in critical water height.

5.6.4 Fuel Substitution Experiments

5.6.4.1 Description

The fuel substitution experiments were of two types. First, insert experiments were carried out in which a relatively small number of fuel rods in the center of the core were replaced by fuel rods of a different type, as in a central 3 x 3 region. Second, changes in core configuration were made in small steps as in changing from a single-region core to a multi-region core by replacing one fuel type with the other fuel type. The reactivity change was measured for each step. Consequently, experimental information is available for a large number of different multi-region configurations. These data are included in Table F-2 in Appendix F.

5.6.4.2 Analytic Results

The fuel substitution experiments provide an excellent test of the

analysis methods and cross sections. The LEOPARD-PDQ-3 sequence was used to determine the reactivity effect of each core change. However, to determine a realistic value for the change, it was necessary to consider the effect of the difference in the size of the error associated with calculations involving each fuel type. For example, the PDQ analysis of the just-critical reference 19×19 PuO_2 - UO_2 core using the experimentally determined transverse buckling and 1964 Geneva Conference cross sections results in a calculated k_{eff} of 1.00867. For the just-critical reference UO_2 core using the same procedure, the calculated k_{eff} is 1.00379. Thus, the discrepancy between analysis and experiment is larger for the plutonium assembly than for the uranium assembly. To compensate for the difference in the inherent error in calculations involving the two fuels, the following procedure was followed:

1. The relative number of neutrons produced by fissioning in each fuel in the multi-region core was determined by means of a PDQ-3 calculation using the measured transverse buckling for the UO_2 reference case.
2. A calculated k_{eff} corresponding to that required for a just-critical experimental condition was determined for each multi-region assembly by weighting the calculated k_{eff} for each single-region 19×19 reference core by the relative number of neutrons produced by that fuel. The calculated k_{eff} for criticality is higher than 1.0 because of the error associated with each fuel.

3. The difference in the PDQ-3 calculated k_{eff} for the assembly and that required for a just-critical condition as determined in (2) represents the reactivity change associated with a change in configuration from the reference core.
4. The difference in reactivity between the 19 x 19 UO_2 reference core and the 19 x 19 PuO_2 - UO_2 reference core indicated by a small difference in just-critical water height was evaluated in PDQ-3 by varying the transverse buckling in accordance with the measured values.

Table 5.21 summarizes the PDQ-3 calculated k_{eff} , the k_{eff} required for criticality as determined by the outlined weighting procedure, and the resulting reactivity difference representing the effect of the configuration change from the reference core. Figure 5.24 compares the measured and calculated reactivity changes for a series of fuel substitution experiments. The analysis and experiment are in good agreement.

5.6.5 Perturbation Experiments

5.6.5.1 Water Slot and Aluminum Slab Experiments

The effect on reactivity and local power peaking resulting from a core perturbation such as a water slot or an aluminum slab were described in Section 5.4 and a comparison of analysis with experiment for the single-region configurations was included.

TABLE 5.21

A Comparison of Analysis and Experiment for
a Series of Fuel Substitution Experiments

Core Configuration	Calculated k_{eff} PDQ-03	Analytical		Experimental
		k_{eff} Required for Criticality PDQ Source Weighting	$\Delta k/k$	$\Delta k/k$
1. 19x19 UO_2 core	1.00379	1.00379	0	0
2. 19x19 core 17x17 UO_2 in center, PuO_2 - UO_2 outer region	1.01272	1.005196	+0.7774	+0.71
3. 19x19 core 15x15 UO_2 in center, PuO_2 - UO_2 outer region	1.01265	1.005861	+0.6789	+0.75
4. 19x19 core 7x7 UO_2 in center, PuO_2 - UO_2 outer region	1.01014	1.007982	+0.2158	+0.37
5. 19x19 core 3x3 UO_2 in center, PuO_2 - UO_2 outer region	1.00933	1.00855	+0.0780	+0.11
6. 19x19 PuO_2 - UO_2 core	1.00892*	1.00867**	+0.025	+0.06
1. 19x19 PuO_2 - UO_2 core	1.00892*	1.00867**	+0.025	+0.06
2. 19x19 core 17x17 PuO_2 - UO_2 , UO_2 outer region	1.00170	1.007619	-0.5919	-0.49
3. 19x19 core 15x15 PuO_2 - UO_2 , UO_2 outer region	1.00157	1.006913	-0.5343	-0.56
4. 19x19 core 11x11 PuO_2 - UO_2 , UO_2 outer region	1.00312	1.005683	-0.2563	-0.40
5. 19x19 core 7x7 PuO_2 - UO_2 , UO_2 outer region	1.00399	1.004643	-0.0653	-0.07
6. 19x19 core 3x3 PuO_2 - UO_2 , UO_2 outer region	1.00405	1.00397	+0.0080	-0.03
7. 19x19 UO_2 core	1.00379	1.00379	0	0

*Calculated k_{eff} at transverse buckling corresponding to UO_2 core.

**Calculated k_{eff} at transverse buckling corresponding to PuO_2 - UO_2 core.

Figure 5.24 A Comparison of Analysis with Experiment for Fuel Substitution Experiments in a Multi-Region 19×19 Core.

The scope of the perturbation experiments for both single-region and multi-region cores is listed in Table 5.15. A comparison of the measured and calculated reactivity effects for these experiments is included in Table 5.22. The method in which group constants for the perturbation were determined using an extra-region as described in Section 5.4 resulted in a favorable comparison of analysis with experiment for all configurations.

5.6.5.2 Control Rod and Soluble Poison Worth

The reactivity worth of control rods and soluble poison was determined in both single-region and multi-region cores. Tables 5.23 and 5.24 contain comparisons of the measurements with the appropriate PDQ-3 calculations. As shown in Table 5.23, the largest rod worth discrepancy, an underestimate of rod worth by 13.6% $\Delta k/k$, occurs in the single region $\text{PuO}_2\text{-UO}_2$ core. In the remaining configurations the comparison between analysis and experiment is good. Additional information concerning the method used is included in Appendix E.

5.7 Flux Measurements and Analysis

5.7.1 Experimental Scope

Relative fast, resonance, and thermal fluxes were measured by means of foils in single-region cores of each fuel and in multi-region cores composed of both fuels. The extent of the measurements is summarized in Table 5.25.

Table 5.22
Reactivity Effect of Perturbation Experiments - Analysis and Experiment

Single-Region Cores								
Core Size	Type of Fuel		Perturbation Position		Measured % $\Delta k/k$	PDQ03 k_{eff}	Calculated % $\Delta k/k$	
19 x 19	UO ₂		Reference Case		0.0	1.004224	+ 0.0	
19 x 19	UO ₂		Water Slot at the Center		+ 0.26	1.006567	+ 0.23	
19 x 19	UO ₂		Aluminum Slab at the Center		- 0.27	1.001655	- 0.26	
19 x 19	PuO ₂ -UO ₂		Reference Case		0.0	1.011479	0.0	
19 x 19	PuO ₂ -UO ₂		Water Slot at the Center		+ 0.49	1.016389	+ 0.49	
19 x 19	PuO ₂ -UO ₂		Aluminum Slab at the Center		- 0.04	1.01063	- 0.09	
Multi-Region Cores								
Core Size	Region 1		Region 2	Perturbation Position	Boron ppm	Measured % $\Delta k/k$	PDQ k_{eff}	Calculated % $\Delta k/k$
	Size	Fuel	Fuel					
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	Reference Case	0.0	0.0	1.008326	0.0
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	Water Slot at the Center	0	+ 0.46	1.01296	+ 0.46
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	Reference Case	0	0	1.00956	0
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	Water Slot at PuO ₂ -UO ₂ Interface	0	+ 0.27	1.012782	+ 0.32
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	Water Slot at UO ₂ Interface	0	+ 0.20	1.012147	+ 0.26
27 x 27	19 x 19	PuO ₂ -UO ₂	UO ₂	Reference Case	1453	0	1.013806	0
27 x 27	19 x 19	PuO ₂ -UO ₂	UO ₂	Water Slot at PuO ₂ -UO ₂ Interface	1453	- 0.25	1.011073	- 0.27
27 x 27	19 x 19	PuO ₂ -UO ₂	UO ₂	Aluminum Slab at PuO ₂ -UO ₂ Interface	1453	- 0.42	1.010917	- 0.29
21 x 21	11 x 11	UO ₂	PuO ₂ -UO ₂	Reference Case	0	0	1.00400	0
21 x 21	11 x 11	UO ₂	PuO ₂ -UO ₂	Water Slot at the Center	0	+ 0.20	1.00591	+ 0.19
21 x 21	11 x 11	UO ₂	PuO ₂ -UO ₂	Reference Case	0	0	1.00538	0
21 x 21	11 x 11	UO ₂	PuO ₂ -UO ₂	Water Slot at UO ₂ Interface	0	+ 0.28	1.00769	+ 0.23
21 x 21	11 x 11	UO ₂	PuO ₂ -UO ₂	Water Slot at PuO ₂ -UO ₂ Interface	0	+ 0.25	1.00775	+ 0.24

Table 5.23
Control Rod Worth - Analysis and Experiment

Single-Region Cores						
Core Size	Fuel Type	Control Rod Position	Measured % $\Delta k/k$	Calculated Rod Worth		Discrepancy %
				k_{eff}	% $\Delta k/\bar{k}$	
21 x 21	UO ₂	Reference Core	0	1.003477	0	0
21 x 21	UO ₂	5 rods-center	-4.50	0.95804	-4.63	+ 2.9
21 x 21	PuO ₂ -UO ₂	Reference Core	0	1.012429	0	0
21 x 21	PuO ₂ -UO ₂	5 rods-center	-3.92	0.978187	-3.45	-13.6

Multi-Region Cores								
Core Size	Region 1		Region 2 Fuel	Control Rod Position	Measured % $\Delta k/k$	Calculated Rod Worth		Discrepancy %
	Size	Fuel				k_{eff}	% $\Delta k/\bar{k}$	
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	Reference Core	0	1.008326	0	0
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	5 rods-center	-3.65	0.972443	-3.62	- 0.8
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	Reference Core	0	1.00956	0	0
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	5 rods-PuO ₂ -UO ₂ region 1 row from interface	-3.1			
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	5 rods-PuO ₂ -UO ₂ interface	-2.92	0.981933	-2.76	- 5.8
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	5 rods-UO ₂ inter- face (first row)	-2.60	0.983037	-2.65	+ 1.9
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	5 rods-UO ₂ region (second row)	-2.25			
21 x 21	11 x 11	PuO ₂ -UO ₂	UO ₂	5 rods-UO ₂ region (third row)	-1.92			

Table 5.24

Soluble Poison Worth - Analysis and Experiment

<u>Single Region Core</u>					
Core Size	Type Fuel	Δ Boron (ppm)	Measured % $\Delta k/k$	Poison Worth	
				Calculated	
				k_{eff}	% $\Delta k/\bar{k}$
19 x 19	PuO_2-UO_2	0 - 50 ppm	- 0.79	Δ Ref: = 1.011479 Boron: = 1.00374	- 0.77

<u>Multi-Region Core</u>							
Core Size	Region 1		Region 2	Δ Boron (ppm)	Measured % $\Delta k/k$	Poison Worth	
	Size	Fuel	Fuel			Calculated ¹	
						k_{eff}	% $\Delta k/\bar{k}$
27 x 27	19 x 19	PuO_2-UO_2	UO_2	1166 - 1388	- 1.49	Δ Ref: = 1.0400 Boron: = 1.0247	- 1.48
27 x 27	19 x 19	PuO_2-UO_2	UO_2	116 - 1453	- 1.95	Δ Boron: = 1.0206	- 1.88

¹PDQ Group Constants were Determined using Leonard Cross Sections

Table 5.25

Flux Measurements ScopeSingle-Region Cores

Fuel	Core Size	Type Foil Measurement
UO ₂	19 x 19	U-238, Np-239, Au
PuO ₂ -UO ₂	19 x 19	U-238, Np-239, Au, Dys

Multi-Region Cores

Core Size	<u>Region 1</u>		<u>Region 2</u>		Boron, ppm	Type Foil Measurement
	Size	Fuel	Fuel			
19 x 19	11 x 11	PuO ₂ -UO ₂	UO ₂	0		U-238, Np-239, Dys
27 x 27	19 x 19	PuO ₂ -UO ₂	UO ₂	1453		U-238, Np-239, Dys
19 x 19	11 x 11	UO ₂	PuO ₂ -UO ₂	0		U-238, Np-239, Dys
27 x 27	19 x 19	UO ₂	PuO ₂ -UO ₂	1252		U-238, Np-239, Dys

The fast flux was measured by counting the fission product decay gammas in bare irradiated U-238 foils. Since U-238 is fissionable by neutrons with energies greater than 1 Mev, the gamma emission is directly proportional to the greater than 1 Mev flux.

By counting the foils, three days later, the relative U-238 capture distribution was measured. The activity is due primarily to the resonance flux, however, it also contains a thermal flux component.

The 104 kev X-ray emitted during the information of ${}_{94}\text{Pu}^{239}$ was counted.

Thermal flux shapes were determined using bare dysprosium foils. The activity of the dysprosium foils is due to the following reaction:

The energy dependent absorption cross-section of the initial isotope follows a $\frac{1}{v}$ distribution. No cadmium covered measurements were made.

5.7.2 Analytic Results

Calculated flux distributions were determined using PDQ-3 for two of the multi-region configurations of Table 5.25, the 19 x 19 clean configuration containing a 11 x 11 $\text{PuO}_2\text{-UO}_2$ inner region and the 27 x 27

borated core with a 19×19 PuO_2 - UO_2 inner region. The PDQ-3 calculations were carried out using four energy groups and conventional thermal cross sections.

5.7.2.1 Fast Flux Comparisons

Figure 5.25 compares the measured and calculated fast flux distributions for the two configurations. The broken curves are plots of the fast flux, ϕ_1 , as taken directly from the four-group PDQ-3 calculation. However, the U-238 foils contain a small quantity of U-235 (180 ppm) as an impurity. Consequently, it is necessary to correct for the fissioning that occurs in the U-235. The corrected analytic distributions are shown by the solid curves and represent:

$$\Sigma \sigma_{fi} \phi_i$$

where σ_i = flux in group i from PDQ-3

$$\sigma_{fi} = \sigma_{fi_{\text{U-238}}} + 0.000180 \sigma_{fi_{\text{U-235}}}$$

The analysis shows a discontinuity at the region boundary because of the difference in the microscopic fission cross-section that results from the different spectrum in each region as determined in the LEOPARD calculation of group constants.

5.7.2.2 Resonance Flux

The Np-239 activity is due primarily to resonance capture in U-238. In the PDQ-3 analysis, the resonance flux is represented by the

Figure 5.25 - A Comparison of Calculated and Measured Fast Flux Distributions in Two Critical Configurations Containing a $\text{PuO}_2\text{-UO}_2$ Inner Region and a UO_2 Outer Region

third group flux, ϕ_3 . Figure 5.26 compares the foil measurement with the analysis. The broken curves represent ϕ_3 from PDQ-3. However, it is necessary to account for the thermal captures in U-238 and to correct for the difference between the homogenized flux of the calculation and the flux in the moderator where the measurement was actually made. It is also necessary to account for the effect of the fuel rods on the foil resonance flux. In an attempt to correct for these effects, the following equation was used:

$$(\sigma_{a3} + D \cdot \sigma_{a \text{ res}}) \phi_3 + \sigma_{a4} \cdot \phi_4 \left(\frac{\bar{\phi}_{\text{mod}}}{\bar{\phi}_{\text{cell}}} \right)$$

σ_{a3} = Smooth absorption cross-section for group 3 from LEOPARD.

D = Dancoff correction for fuel rod.

$\sigma_{a \text{ res}}$ = Resonance absorption cross-section assuming infinite dilution of U-238.

σ_{a4} = Microscopic absorption cross-section for group 4 from LEOPARD.

$\frac{\bar{\phi}_{\text{mod}}}{\bar{\phi}_{\text{cell}}}$ = Relative flux in moderator to average in cell-determined in LASER or LEOPARD.

The resulting corrected distributions are shown by the solid curves in Figure 5.26. The agreement between analysis and experiment is not close. While the corrections that were applied act in the right direction, additional work and time would have been desirable to improve the correlation.

Figure 5.26 - A Comparison of Calculated and Measured Np-239 Activity in Two Critical Configurations Containing a $\text{PuO}_2\text{-UO}_2$ Inner Region and a UO_2 Outer Region

5.7.2.3 Thermal Flux

The measured and calculated thermal flux distributions for the two configurations are compared in Figure 5.27. The broken curves represent the fourth group flux, ϕ_4 , from PDQ-3. However, it is necessary to correct for the difference in cell average to moderator flux as was done previously and for the effect of the difference in the thermal spectrum in each core region. The following correction was used:

$$\phi_4 \left(\frac{1}{V} \right) \times \frac{\phi_{\text{max}}}{\phi_{\text{cell}}}$$

The effect of the variation in the thermal spectrum across the cell in different fuel regions was found to be insignificant. The comparison was made using LASER in which the energy spectrum as a function of position is available and LEOPARD in which the region average to cell average flux is available.

The comparison of analysis with experiment shows a large discrepancy between the calculated and measured thermal flux distributions particularly near the reflector. This result is in agreement with the consistent over-prediction of fuel rod power at the reflector-fuel interface and points out the need for the use of reflector group constants that are determined with a more realistic spectrum.

5.8 Kinetics

5.8.1 Experimental Scope

Two different methods, pulse neutron techniques and noise analysis, were

Figure 5.27 - A Comparison of Calculated and Measured Thermal Flux in Two Critical Configurations Containing a $\text{PuO}_2\text{-UO}_2$ Inner Region and a UO_2 Outer Region

used during the WREC critical program in an effort to investigate the kinetic characteristics of UO_2 and $\text{PuO}_2\text{-UO}_2$ fueled cores. The scope of the measurements is shown in Table 2.1.

5.8.2 Results

The pulse neutron measurements did not provide meaningful results while satisfactory noise measurements were obtained in only one configuration, a 27×27 core with an inner 19×19 region of $\text{PuO}_2\text{-UO}_2$. In this experiment, described in Reference 14, a β/ℓ of 292 was determined.

To obtain useful kinetic information for comparison with the analysis, additional experimental work would be needed. In particular, it would be desirable to conduct both types of experiments using in-core detectors. It would also be desirable to carry out part of the experiments in unreflected critical configurations.

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Appendix A

APPENDIX A

Specifications

1. Fuel Specifications

The specifications for the five different types of fuel rods designed for Saxton Core II and available for the WREC critical experiments are listed in Table A-1.

Figure A-1 is a drawing of a typical $\text{PuO}_2\text{-UO}_2$ fuel rod. The removable and sub-assembly $\text{PuO}_2\text{-UO}_2$ fuel rods are shown in Figures A-2 and A-3.

2. Design Core Performance Specifications

Table A-2 summarizes the design and performance specifications for Saxton Core II.

3. WREC Critical Experiment

Figure A-4 is a side elevation of the core tank. The fuel rods project vertically through the fuel rod holes in the top aluminum core plate. Two additional aluminum grid plates below the top grid plate support the fuel rods in a vertical position.

The fuel rod holes in the aluminum grid plates are 0.397 ± 0.002 inches in diameter with additional holes of 0.193 ± 0.001 inches in diameter in the middle core plate, for water circulation. The hole pattern is also shown in Figure A-5. The three grid plates are supported at each corner with 7/16-inch stainless steel support rods which are covered with 5/8-inch aluminum pipe. The grid plates with their support rods make up the grid structure.

Table A-1
Fuel Rod Specifications

	UO ₂ Rods* (Pelletized)	PuO ₂ -UO ₂ Rods* (Pelletized)	PuO ₂ -UO ₂ Rods (Vibratory Compacted)	PuO ₂ -UO ₂ Rods (Pelletized)	PuO ₂ -UO ₂ Rods (Vibratory Compacted)
Enrichment	5.74 w/o U-235 in UO ₂	6.6 w/o PuO ₂ in UO ₂	6.6 w/o PuO ₂ in UC ₂	6.6 w/o PuO ₂ in UO ₂	6.6 w/o PuO ₂ in UO ₂
Pellet Diameter	0.357 inch	0.3374 inch	--	0.3558 inch	--
Clad O.D.	0.391 inch	0.391 inch	0.391 inch	0.391 inch	0.391 inch
Clad I. D.	0.351 inch	0.3445 inch	0.3445 inch	0.361 inch	0.361 inch
Clad Material	304 SS	Zircaloy-4	Zircaloy-4	304 SS	304 SS
% Theoretical Density	93.0	94.0	87.0	94.0	87.0
Fuel Length	36.6 inch	36.6 inch	36.6 inch	36.6 inch	35.6 inch

Theoretical Density of UO₂ = 10.96 grams/cc

Theoretical Density of PuO₂ = 11.46 grams/cc

* Utilized in the critical experiments.

UO₂

UO₂ Weight/Rod = 604.25 grams/rod

Weight Fraction of U Metal = 0.8813

Weight Percent of U-235 = 5.742

U Metal = 532.526 grams/rod

U-235 = 30.578 grams/rod

PuO₂-UO₂

Type Fuel (Clad-Fuel)	PuO ₂ -UO ₂ Weight/Rod (gms/rod)	PuO ₂ Weight/Rod (gms/rod)	Pu Metal (gms/rod)	Pu-239 (gms/rod)	Pu-240 (gms/rod)	Pu-241 (gms/rod)	Pu-242 (gms/rod)
Zirc-Pellets	545.576	36.074	31.815	28.789	2.727	0.283	0.013
SS-Pellets	603.707	40.175	35.432	32.062	3.037	0.315	0.014
Zirc-Vipac	535.059	35.314	31.145	28.183	2.669	0.277	0.012
SS-Vipac	587.534	38.777	34.199	30.947	2.931	0.304	0.014

Weight Percent of PuO₂ = 6.60

Weight Fraction of Pu Metal = 0.882

Weight Percent of Pu-239 = 90.49

Weight Percent of Pu-240 = 8.57

Weight Percent of Pu-241 = 0.89

Weight Percent of Pu-242 = 0.04

9 x 9 Assembly Removable Rods

Figure A-2

NOTES:

- 1-WELD PROCEDURE (INSPECTION TO BE APPROVED BY WAPD)
- 2-RECORD IDENT. NO., FUEL WT., (METAL WT.)
- 3-MOISTURE CONTENT OF FUEL PRIOR TO WELDING TO BE AS SPECIFIED IN NOTE "Y" TUBE MUST BE DRY.
- 4-USE ONE FILLER AT BOTTOM OF FUEL TUBE AND ADD FILLERS TO TOP END AS NECESSARY (MIN. OF ONE ADDITIONAL AT TOP END TO OBTAIN THIS DIMENSION.)
- 5-ROD ASSY MUST BE STRAIGHT WITHIN .010 PER FT. BETWEEN END PLUG WELDS WHEN LYING ON A SURFACE PLATE.
- 6-PRIOR TO ASSY, WIDE INSIDE (OUTSIDE SURFACES OF TUBE, SPACER, END PLUGS, (SPRING (USE ALCOHOL ONLY AS A SOLVENT) WITH SWABS TO REMOVE ALL FOREIGN MATTER. W.P.E. WITH DRY SWAB AFTER CLEANING.
- 7-END PLUGS TO BE CONCENTRIC WITH TUBE ENDS WITHIN .010 TOTAL AFTER WELDING.

BILL OF MATERIAL									
QTY	ITEM	SYMBOL	REVISIONS	NATURAL SPECIFICATION	UNIT	QTY	QTY	QTY	QTY
1	FUEL TUBE	1295							
2	FILLER	1295							
3	END PLUG	1295							
4	END PLUG	1295							
5	SPRING	1295							
6	SPRING	1295							
7	TUBE	1295							
8	SPRING	1295							
9	FUEL COLUMN	1295							
10	FILLER	1295							
11	END PLUG	1295							
12	END PLUG	1295							

*X: FUEL COLUMN TO CONSIST OF A MIXTURE OF URANIUM DIOXIDE & PLUTONIUM DIOXIDE POWDER VIBRATORY COMPACTED IN ACCORDANCE WITH WAPD APPROVED PROCEDURE TO A DENSITY OF 87.1% OF THEORETICAL. POWDER TO BE PREPARED FOR COMPACTION BY DYNAPAK OR EQUIVALENT PROCESS. ENRICHMENT OF UO₂ OXIDE COMPOSITION, PUO₂ CONTENT, AND PARTICLE SIZE & DENSITY FOR POWDER TO BE SPECIFIED ON D.O.

*Y: GAS (VAPOR) CONTENT OF FUEL TO BE LIMITED TO FOLLOWING:

- GROUP 1 - H₂O - 30 PPM
 N₂ - 75 PPM
 H₂ - 15 PPM
 TOTAL GAS - .05% (EXCLUSIVE OF H₂O)
- GROUP 2 - H₂O - 100 PPM
 N₂ - 100 PPM
 H₂ - 20 PPM (S.H.C.H BONDS)
 TOTAL GAS - .06% (EXCLUSIVE OF H₂O)

- 8-MAX. DIA. PERMITTED AFTER WELDING AS FOLLOWS: 1.397 DIA. (BEFORE PICKLING) ALL SURFACES IN WELD AREA TO HAVE SMOOTH TRANSITION. CAUTION! NO BONDING OF TUBE WALL PERMITTED ROLLING OF WELDS PERMITTED PROVIDED THE OPERATION DOES NOT EXTEND OVER THE UNSUPPORTED CLAD AREA. THE ROLLING PROCEDURE IS TO BE APPROVED BY WAPD ENGINEERING.
- 9-PRICK PUNCH END PLUGS AT 3 EQUALLY SPACED LOCATIONS TO OBTAIN .001/.003 INTERFERENCE FIT WITH TUBE. CAUTION! REMOVE ALL SHARP EDGES IN RAISED METAL CAUSED BY PRICK PUNCHING SO AS NOT TO SCORE I.D. OF TUBE. (TYP. BOTH ENDS)
- 10-FUEL ROD ASSY TO BE CORROSION TESTED AFTER WELDING IN ACCORDANCE WITH WAPD APPROVED PROCEDURE. MAX. TUBE & WELD DIA. AFTER PICKLING TO BE .397
- 11-THIS DIMENSION TO BE ESTABLISHED BY SPRING INSERTION TOOL / IS NOT AN INSPECTION REQUIREMENT.

BILL OF MATERIAL										NO. REQ.	
NOTE	ITEM	TITLE	DRAWING & Q.R. OR IT.	MATERIAL SPECIFICATION	REVISIONS	DATE	BY	CHKD	APP'D	1	2
	1	END PLUG	674C887 IT.4							1	-
	2	END PLUG	674C886 IT.3							-	2
	3	TUBE	499B772 IT.6							1	-
	4	TUBE	674C863 IT.1							-	1
	5	END PLUG	674C886 IT.1							-	2
	6	SPRING	500B086 IT.1							-	1
	7	FILLER	499B591 IT.2							1	-
	8	PELLET STACK	500B071 IT.1							-	1
	9	FUEL COLUMN								1	-
	10	END PLUG	674C887 IT.3							1	-
	11	FILLER	500B156 IT.2							-	1
	12	SPRING	500B157 IT.2							-	1
	13	END PLUG	674C887 IT.1							-	1
	14	END PLUG	674C887 IT.2							-	1

X- FUEL TO CONSIST OF URANIUM DIOXIDE AND PLUTONIUM DIOXIDE POWDER, PREPARED FOR COMPACTION, BY DYNAPAK OR EQUIV. PROCESS AND VIBRATORY COMPACTED PER WAPD APPROVED PROCEDURE TO A DENSITY OF $87.0 \pm 1\%$ OF THEORETICAL. ENRICHMENT OF UO_2 OXIDE COMPOSITION, $PLUO_2$ CONTENT, PARTICLE DENSITY & SIZE FOR POWDER TO BE SPECIFIED ON P.O.

Y- GAS & VAPOR CONTENT OF FUEL TO BE LIMITED TO FOLLOWING:

GROUPS 1 & 3 — H_2O — 30 PPM
 N_2 — 75 PPM
 H_2 — 15 PPM
 TOTAL GAS — $.05\% \text{ (EXCLUSIVE OF } H_2O)$

GROUP 2 & 4 — H_2O — 100 PPM
 N_2 — 100 PPM
 H_2 — 2.0 PPM (3% H_2 C-H BOND)
 TOTAL GAS — $.06\% \text{ (EXCLUSIVE OF } H_2O)$

NOTES:

- WELD PROCEDURE AND INSPECTION TO BE APPROVED BY WAPD.
- RECORD IDENT. NO., FUEL WT., & METAL WT.
- MOISTURE CONTENT OF FUELS PRIOR TO WELDING TO BE AS SPECIFIED IN NOTE 'Y'. TUBE MUST BE DRY.
- USE ONE FILLER AT BOTTOM OF FUEL TUBE AND ADD FILLERS TO TOP END AS NECESSARY (MIN OF ONE ADDITIONAL FILLER AT TOP END) TO OBTAIN THIS DIMENSION.
- ROD ASSEMBLY MUST BE STRAIGHT WITHIN .010 PER FT. BETWEEN END PLUG WELDS WHEN LYING ON A FLAT SURFACE AND END PLUGS MUST BE CONCENTRIC WITHIN .003 WITH RESPECT TO END OF TUBE.
- PRIOR TO ASS'Y, WIPE INSIDE & OUTSIDE SURFACES OF TUBE, SPACER, END PLUGS, & SPRING (USE ALCOHOL ONLY AS A SOLVENT) WITH SWAB TO REMOVE ALL FOREIGN MATTER WIPE WITH DRY SWAB AFTER CLEANING.
- ALL SURFACES IN WELD AREA TO HAVE SMOOTH TRANSITION. CAUTION! NO GRINDING OF TUBE WALL PERMITTED. ROLLING OF WELDS PERMITTED PROVIDED THE OPERATION DOES NOT EXTEND OVER THE UNSUPPORTED CLAD AREA. THE ROLLING PROCEDURE IS TO BE APPROVED BY WAPD ENGINEERING.
- FUEL ROD ASS'YS TO BE CORROSION TESTED AFTER WELDING IN ACCORDANCE WITH WAPD APPROVED PROCEDURE. MAX. TUBE WELD DIA. AFTER PICKLING TO BE .397.
- PICK PUNCH END PLUGS OF GROUPS 1 & 3 AT 5 EQUALLY SPACED LOCATIONS TO OBTAIN .001/.003 INTERFERENCE FIT WITH TUBE. CAUTION! REMOVE ALL SHARP EDGES IN RAISED METAL CAUSED BY PICK PUNCHING SO AS NOT TO SCORE I.D. OF TUBE (TOP BOTH ENDS).
- THIS DIMENSION TO BE ESTABLISHED BY SPRING INSERTION TOOL, IS NOT AN INSPECTION REQUIREMENT.

Figure A-3 Fuel Rods for Removable Fuel Assembly

Table A-2

Saxton Reactor Core II Design Data

Core Description

Average Core Diameter (cold)	28.07 inches
Active Core Length (cold)	36.6 inches
Number of Fuel Assemblies	21
PuO ₂ -UO ₂ Fuel Assemblies	9
UO ₂ Fuel Assemblies	11
Enrichment:	
w/o PuO ₂ in UO ₂	6.6 w/o
a/o U-235 in UO ₂	5.742 a/o
*Number of Fuel Rods:	
PuO ₂ -UO ₂ , Pelletized, Zr Clad	466
PuO ₂ -UO ₂ , Vibratory-Compacted, Zr Clad	134
PuO ₂ -UO ₂ , Pelletized, SS Clad	20
PuO ₂ -UO ₂ , Vibratory Compacted, SS Clad	10
UO ₂ , Pelletized, SS Clad	803
UO ₂ Fuel Rods in L-Sections	81
UO ₂ Fuel Rods in Control Rod Followers	54
Fuel Rod Specifications	Table A-1
Fuel Mass in the Core, kg.	911.2
UO ₂ Fuel, kg.	566.8
PuO ₂ -UO ₂ Fuel, kg.	344.5

*Does not include sub-assembly fuel rods.

Table A-2 (Continued)

Control Rod Data

Number of Movable Control Rods	6
Control Rod Material	80% Ag, 15% In, 5% Cd
Control Rod Shape	Offset Cruciform
Control Rod Design	18 Absorbant Rods Assembled Together
Absorbant Diameter, in.	0.378
Clad O.D., in.	0.414
Clad I.D., in.	0.380
Center to Center Distance, in.	0.4618

Thermal and Hydraulic Design Data

Total Core

Total Heat Output, Initial	22.1 MWt
Total Heat Output, initial	$75.4 \cdot 10^6$ Btu/hr
Heat Generated in Fuel	97.4%
System Pressure, Nominal	2000 psi
System Pressure, Minimum Steady State	1950 psi

Coolant Flow

Total Flow Rate	$2.94 \cdot 10^6$ lb/hr
Effective Flow Rate for Heat Transfer	$2.5 \cdot 10^6$ lb/hr
Flow Area for Heat Transfer Flow (Unit Cells)	2.51 ft^2
Average Velocity Along Fuel Rods	5.8 ft/sec

Table A-2 (Continued)

Coolant Temperatures

Nominal Inlet	520.0°F
Maximum Inlet, including instrumentation errors and deadband	525.0°F
Average Rise in Vessel	20.7°F
Average Rise in Core	24.8°F
Average in Vessel	530.7°F
Average in Core	532.5°F

Average Film Coefficient	2540 Btu/hr-ft ² -°F
Average Film Temperature Difference	58.0°F

Heat Transfer

Active Heat Transfer Surface Area of Fuel Rods	498 ft ²
Average Heat Flux	147200.0 Btu/hr-ft ²
Average Thermal Output	4.4 kw/ft
Maximum Clad Surface Temp. at Nominal Pressure	642°F

Pressure Drop

Across Core	4.1 psi
Across Vessel, including Nozzles	11.3 psi

Central Core Region (PuO₂-UO₂ Fuel)

F _q Heat Flux Hot Channel Factor	3.61
F _{ΔH} Enthalpy Rise Hot Channel Factor	2.81
Maximum Heat Flux	531,400 Btu/hr-ft ²
Maximum Thermal Output	16.0 kw/ft
Nominal Outlet Temperature of Hot Channel	586.7°F
Maximum Outlet Temperature of Hot Channel	591.7°F

Table A-2 (Continued)

Outer Core Region (UO_2 Fuel)

F_q Heat Flux Hot Channel Factor	2.04
$F_{\Delta H}$ Enthalpy Rise Hot Channel Factor	1.59
Maximum Heat Flux	301600 Btu/hr-ft ²
Maximum Thermal Output	9.05 kw/ft
Nominal Outlet Temperature of Hot Channel	558.9°F
Maximum Outlet Temperature of Hot Channel	563.9°F

Figure A-4 - Core Tank - Side Elevation

TOP AND BOTTOM CORE PLATE

$A = \text{PITCH}$

FUEL ROD HOLES = 0.397 ± 0.002 " DIA.

MIDDLE CORE PLATE

$A = \text{PITCH}$
 $B = \frac{1}{2}A$

FUEL ROD HOLES = 0.397 ± 0.002 " DIA.
 CIRCULATION HOLES = 0.193 ± 0.001 " DIA.

Figure A-5 - Core Plate Layout

Appendix B

APPENDIX B

Additional Parameter Study Information

1. Scope

The material in this section supplements the parameter study information presented in Section 3. Table B-1 summarizes the scope of this information.

Table B-1

Scope of Parameter Study Information

A. Characteristics of Saxton Core with Uniform Loading

Figure No.	Parameter Investigated	With Variations In	Remarks
B-1	1. K_{eff} (Hot, Clean, 23.5 MWt) 2. Fast to Thermal Flux Ratio 3. Wigner-Wilkins $\frac{1}{v}$ Factor	Boron Concentration a) 0 ppm b) 1000 ppm	SS-304 Clad LEOPARD Calculations
B-2	1. "g" Factor 2. Macroscopic Fission Cross Section Thermal Weighted	--	SS-304 Clad LEOPARD Calculation
B-3	1. Fast to Thermal Flux Ratio 2. Wigner-Wilkins $\frac{1}{v}$ Factor 3. Weighted Macroscopic Absorption Cross Section 4. Weighted Macroscopic Fission Cross Section	Boron Concentration with Time	Conventional Saxton Core 5.74 a/o U-235 SS-304 Clad LEOPARD Calculation
B-4	1. Fast to Thermal Flux Ratio 2. Wigner-Wilkins $\frac{1}{v}$ Factor 3. Weighted Macroscopic Absorption Cross Section 4. Weighted Macroscopic Fission Cross Section	Boron Concentration with Time	3.98 a/o Pu SS-304 Clad LEOPARD Calculations

B. Relative Power Distributions (LUX) For Two Region
Saxton Cores as a Function of Hours Operation at 23.5 MWt

Figure No.	Plutonium Enrichment	Hours Operation (23.5 MWt)
B-5	3.98 a/o PuO_2	Beginning of Life
B-6	5.04 a/o PuO_2	Beginning of Life
B-7	5.04 a/o PuO_2	1,000
B-8	5.04 a/o PuO_2	5,000
B-9	5.04 a/o PuO_2	10,000

C. PDQ Calculated Power Distributions for Variations in
Boron Concentration, Rod Pattern and Enrichment

Figure No.	Configuration	Variation
B-10	1 Highly Depleted UO_2 Center Assembly 8 PuO_2 - UO_2 (5.044 a/o Pu) Around Center	Boron - 1000 to 2000 ppm
B-11	1 Highly Depleted UO_2 Center Assembly 8 PuO_2 - UO_2 (5.044 a/o Pu) Around Center	Rod Pattern a) Rod 5 In b) Rods 2,5 In c) Rods 1,2,5,6 In
B-12	1 Highly Depleted UO_2 Center Assembly 8 PuO_2 - UO_2 (5.044 a/o Pu) Around Center	Pu Enrichment a) 4.0 a/o Pu b) 5.04 a/o Pu c) 8.0 a/o Pu

Figure B-1 - Variation of Core Characteristics with Plutonium Enrichment for a Saxton Core Uniformly Loaded with Natural Uranium Enriched with Plutonium (SS-304 Clad)

Figure B-2 - Variation of Core Characteristics with Plutonium Enrichment for a Saxton Core Uniformly Loaded with Natural Uranium Enriched with Plutonium (SS-304 Clad)

Figure B-3 - Variation of Core Characteristics During Lifetime for a Saxton Core Uniformly Loaded with 3.98 a/o Plutonium in Natural Uranium SS-304 Clad

Figure B-4 - Variation of Core Characteristics During Lifetime for a Conventional Saxton Core 5.74 a/o Enriched Uranium SS-304 Clad

Figure B-5 - Relative Power Distributions for a Two Region Saxton Core
Loaded in Part with 3.98 a/o PuO_2 & 96.02 a/o UO_2 at Beginning of Life

Figure B-6 - Relative Power Distributions for a Two Region Saxton Core
Loaded in Part with 5.04 a/o PuO_2 & 94.96 a/o UO_2 at Beginning of Life

Figure B-7 - Relative Power Distributions for a Two Region Saxton Core
Loaded in Part with 5.04 a/o PuO_2 & 94.96 a/o UO_2
After 1000 Hours at 23-1/2 MWt

Figure B-8 - Relative Power Distributions for a Two Region Saxton Core
Loaded in Part with 5.04 a/o PuO_2 & 94.96 a/o UO_2
After 5000 Hours at 23-1/2 MWt

Figure B-9 - Relative Power Distributions for a Two Region Saxton Core
Loaded in Part with 5.04 a/o PuO_2 & 94.96 a/o UO_2
After 10,000 Hours at 23-1/2 MWt

NOTE: The underlined value in each assembly is the average power in that assembly relative to the average power in the core. The relative power is also shown for individual fuel rods near the uncontrolled corners of assemblies (where hot spots usually occur). If the hot rod is elsewhere, the value and location are shown.

8 PuO₂-UO₂ assemblies (5.04 a/o Pu, Zr clad) around center highly depleted (18,000 MWD/MTU) UO₂ assembly. 1000 ppm Boron
Eigenvalue: 1.060

8 PuO₂-UO₂ assemblies (5.044 a/o Pu, Zr clad) around center highly depleted (18,000 MWD/MTU) UO₂ assembly. 2000 ppm Boron
Eigenvalue: 0.996

Figure B-10 - The Effect on PDQ-3 Calculated Power Distributions for a Variation in Boron Concentration

8 PuO₂-UO₂ Assemblies (5.0444 a/o Pu, Zr clad) around center
highly depleted (18,000 MWD/MTU) UO₂ assembly. 1000 ppm Boron,
Rods 2, 5 in. Eigenvalue: 0.980C

NOTE: The underlined value in each assembly is the average power in that assembly relative to the average power in the core. The relative power is also shown for individual fuel rods near the uncontrolled corners of assemblies (where hot spots usually occur). If the hot rod is elsewhere, the value and location are shown.

8 PuO₂-UO₂ assemblies (5.0444 a/o Pu, Zr clad) around center
highly depleted (18,000 MWD/MTU) UO₂ assembly. 1000 ppm Boron
Rod 5 in. Eigenvalue: 1.024

8 PuO₂-UO₂ Assemblies (5.0444 a/o Pu, Zr clad) around center
highly depleted (18,000 MWD/MTU) UO₂ Assembly 1000 ppm Boron,
Rods 1, 2, 5, 6 in Eigenvalue: 0.9285

Figure B-11 - The Effect on PDQ-3 Calculated Power Distributions for Three Different Rod Patterns

8 PuO₂-UO₂ assemblies (5.04 a/o Pu, Zr clad) in corner positions, 1000 ppm Boron Eigenvalue: 1.070

NOTE: The underlined value in each assembly is the average power in that assembly relative to the average power in the core. The relative power is also shown for individual fuel rods near the uncontrolled corners of assemblies (where hot spots usually occur). If the hot rod is elsewhere, the value and location are shown.

8 PuO₂-UO₂ Assemblies (4.000 A/o Pu, Zr Clad) in corner positions 1000 PPM Boron Eigenvalue: 1.067

8 PuO₂-UO₂ Assemblies (8.000 A/o Pu, Zr Clad) in corner positions 1000 PPM Boron Eigenvalue: 1.076

Figure B-12 - The Effect on PDQ-3 Calculated Power Distributions for Three Plutonium Enrichment Variations

Appendix C

APPENDIX C

Nuclear Design Calculation Details

1. Lifetime Calculations

Lifetime calculations were carried out using both the LUX and TURBO* depletion codes. For the LUX calculations, the following procedure was used:

- a) A boron dilution-burnup sequence was specified for the LEOPARD lifetime calculation. The following sequence was normally specified:

<u>Hours at</u> <u>23.5 MWt</u>	<u>Boron, ppm</u>
0	1,000
50	1,000
100	1,000
500	1,000
1,000	1,000
2,000	986
4,000	722
6,000	444
8,000	167
10,000	0

- b) The middle of life microscopic cross section library (4000 hours) from LEOPARD was used as input to a LUX radial depletion calculation. The LUX calculation was set up in four radial regions: 1) the $\text{PuO}_2\text{-UO}_2$ inner region; 2) a UO_2 outer region; 3) a stainless steel baffle region; 4) a H_2O stainless steel reflector region.
- c) Since LUX uses a single cross section library, it was necessary to adjust the thermal self-shielding factor, g-factor, to account

for the effect of differences in spectrum on the microscopic cross section for those elements that were present in more than one region. The thermal self-shielding factors calculated by LEOPARD were modified to preserve the correct reaction rate in each region.

- d) The epithermal cross sections of the fuel materials were also different due to the difference in spectrum. The epithermal microscopic library for plutonium isotopes in both fuel regions was taken from the LEOPARD calculation of the plutonium region while the library for U-235 used in both regions was taken from the LEOPARD calculation of the uranium region.

The results of the TURBO* x-y depletion calculation are summarized in Figure C-1. The boron dilution-burnup sequence that was followed is also shown in this figure. The same procedure that was used in the LUX calculations to account for a single library limitation was used for the TURBO* calculation.

The TURBO* calculation was carried out for an enrichment of 6.0 w/o $\text{PuO}_2\text{-UO}_2$ with the same isotopic composition as that used in the initial parameter studies (Section 4.1). This fuel has a slightly higher Pu-240 content than that of the final fuel specification. Each fuel assembly was treated as two characteristic compositions, an inner composition with the normal 0.58-inch pitch (cold) and an outer composition representing the outer fuel row. The outer fuel row had a slightly larger pitch to account for a small part of the water in the water slot. This procedure was adopted to conserve mesh lines in the calculation. An 87 x 87 mesh was used.

The reactivity-lifetime characteristics of the core obtained with LUX and TURBO* are compared in Figure C-2. In TURBO*, the installed reactivity was determined from the evaluation of boron worth before and after dilution. This procedure assumes the boron worth is linear with concentration and underestimates the installed reactivity at the higher boron concentrations early in life. Correcting for this effect would reduce the small difference in the slope of the reactivity-lifetime curves for the two analysis methods. Figure C-2 also includes a comparison of the effect on reactivity and lifetime for a change in isotopic content of the plutonium fuel.

Figure C-3 summarizes the initial hot, clean reactivity at power that is needed to achieve a specified lifetime as indicated by the LUX and TURBO* analysis. The effect on the reactivity requirement for a variation in plutonium enrichment is shown in Figure C-4.

Figure C-3 - Initial Reactivity Requirement versus
Available Lifetime at 23.5 MWt

Figure C-4

Initial Reactivity Requirement Versus
Available Lifetime at 23.5 MWt for
Two Different PuO_2 Weight Fractions

Appendix D

APPENDIX D

Pre-Analysis Summary

1. Experimental Program

The experimental program as originally planned is summarized in Table D.1. Also included in Table D.1 are the predicted reactivity requirements for the various configurations as to number of fuel rods and boron content. For cores containing the mixed-oxide fuel, an allowance for a possible discrepancy between analysis and experiment was included based on a calculation of Hanford mixed-oxide approach-to-critical experiments. No allowance was included for the cores containing UO_2 alone. The allowance for each configuration is listed in Table D.1 together with the fuel rod and boron content requirements.

2. Supporting Analysis

The analysis carried out before the start of the experimental program is divided into two parts. The first part was concerned with determining the criticality requirements for the specified experimental configuration. Also, it was desired to select the most efficient methods for carrying out the post critical comparison of analysis with experiment. The second part was concerned with transient studies for single-region critical configurations composed of the two different fuel types.

Table D-1
Measurements Program Outline

Configuration Number	General Description	Number of Regions	Fuel Type	Core Geometry	Lattice	Measurements	Remarks - Predicted Requirements
A(1)	UO ₂ , One-Region, Clean Core Experiments	One	UO ₂	Square	0.56	Criticality	A series of square cores at different water heights until all available conventional Saxton UO ₂ fuel rods are installed
A(2)		One	UO ₂	Square	0.792	Criticality & Buckling	Remove every other rod in A(1) to form a critical configuration in a loose lattice
A(3)		One	UO ₂	Square	0.56	Type A*	Predicted critical rods = 356 at full water height (calculated $k_{eff} = 1.0$)
A(4)		One	UO ₂	Square	0.56	Reactivity, Power, Flux	Special experiments including: Slot experiments - 1-5 slots in center Control rod experiments - 1-5 rods 3 x 3 experiment using PuO ₂ -UO ₂ rods
1(a)	One-Region, Clean Core Experiments	One	PuO ₂ -UO ₂	Square	0.56	Criticality	A series of square cores at different water heights until all available PuO ₂ -UO ₂ rods are installed. H/Pu = Saxton design, hgt.
1(b)		One	PuO ₂ -UO ₂	Square	0.792	Criticality & Buckling	Remove every other rod in 1(a) to form a critical configuration in a loose lattice
1(c)		One	PuO ₂ -UO ₂	Square	0.56	Type A	Predicted critical rods = 355 at full water height (calculated $k_{eff} = 1.025$)
1(d)		One	PuO ₂ -UO ₂	Square	0.56	Reactivity, Power, Flux	Special experiments including: Slot experiments - 1-5 slots in center Control rod experiments - 1-5 rods 3 x 3 experiment using UO ₂ rods
1(e)		One	PuO ₂ -UO ₂	Square	0.56	Criticality, $d\rho/dT$	Moderator temperature coefficient
2(a)	Two-Region, Clean Core Experiments	Two	PuO ₂ -UO ₂ Inside UO ₂ Outside	Square	0.56	Criticality, $d\rho/dT$	Using the heated water from 1(e) obtain a hot critical. While cooling, obtain $d\rho/dT$
2(b)		Two	As Above	Square	0.56	Type A	Dump hot water. Obtain cold critical. The predicted clean core critical configuration is 144 PuO ₂ -UO ₂ rods in the center of the core (12 x 12) with 217 UO ₂ rods installed on the outside forming a 19 x 19 rod array. (Calculated $k_{eff} \approx 1.025$)
2(c)		Two	As Above	Square	0.56	Reactivity, Power, Flux	Special experiments at region boundaries: Slot experiments Control rod experiments
3(a)	One-Region, Borated-Core Experiments	One	PuO ₂ -UO ₂	Square	0.56	Type A	Borate water. Use all PuO ₂ -UO ₂ rods except those needed for power measurements. For a core containing 400 PuO ₂ -UO ₂ rods, a boron concentration of 150 ppm is predicted. (Calculated $k_{eff} = 1.025$)
4(a)	Two-Region, Borated-Core Experiments	Two	PuO ₂ -UO ₂ Inside UO ₂ Outside	Square	0.56	Fuel Substitution Experiment	Remove PuO ₂ -UO ₂ rods and add UO ₂ rods. Obtain critical at same boron as 3(a)
4(b)		Two	As Above	Square	0.56	Type A	Increase boron content. Add PuO ₂ -UO ₂ rods and UO ₂ rods until critical. For a configuration consisting of 361 PuO ₂ -UO ₂ rods (19 x 19) in the center of the core with 360 UO ₂ rods installed on the outside forming a 27 x 27 rod array, a boron concentration of 1525 ppm is predicted (calculated $k_{eff} = 1.016$)
4(c)		Two	As Above	Square	0.56	Reactivity, Power, Flux	Slot experiment on boundary
5(a)	Two-Region, Inverted Core	Two	UO ₂ Inside PuO ₂ -UO ₂ Outside	Square	0.56	Type A	Load inverted core at $\approx$ the boron content of 4(b) above
5(b)		Two	As Above	Square	0.56	Critical Rods	Dilute to $\approx$ boron content of 3(a)
5(c)		Two	As Above	Square	0.56	Type A	Dump water. Clean core critical
6	One-Region Clean Core	One	PuO ₂ -UO ₂	Square	0.60	Type A	Predicted critical rods = 260 (calculated $k_{eff} = 1.025$)

* Type A Measurements Include:

1. Number of rods required for full water height critical
2. Critical buckling and savings from fuel rod scan and foil measurements
3. β/ρ measurement

3. Criticality Requirements

a. Single Region Cores

The LEOPARD-PDQ sequence was used to determine k_{eff} as a function of boron content in the moderator for a single region square core composed of 400 mixed-oxide ($\text{PuO}_2\text{-UO}_2$) fuel rods in a 0.56-inch lattice. The calculated k_{eff} from PDQ-3, shown in Figure D.1, was then used with the two-group constants from LEOPARD to determine the variation in reflector savings with boron content. Figure D.2 summarizes the results. A two-group formula using the calculated reflector savings was used to determine the number of fuel rods required for a specified k_{eff} . Figure D.1 also shows the number of fuel rods required for criticality for two assumed values of k_{eff} . Table D.2 summarizes the information contained in Figures D.1 and D.2.

The same type of analysis (LEOPARD - PDQ-3) was carried out for a square, clean-core containing UO_2 fuel rods at a 0.56-inch pitch. These results are also included in Table D.2.

b. Two-Region Cores

The criticality requirements for the clean, two-region cores were determined using both one and two-dimensional calculations. The two methods were compared with respect to the calculated reactivity and computer time requirements. Table D.3 summarizes the comparison for both one-region and two-region calculations. The calculated values

Appendix E

APPENDIX E

WREC Critical Experiment - Analysis Details

1. LEOPARD Calculations

The LEOPARD code was used extensively throughout the WREC critical program. To illustrate the type of information available from a LEOPARD calculation, the output for the $\text{PuO}_2\text{-UO}_2$ 0.56-inch critical lattice is included in Table E-1. Table E-2 contains a comparison of a number of LEOPARD calculated parameters for the different critical lattices using both types of fuel. In these calculations, the experimentally determined buckling was used as input and the aluminum mid-plate was neglected.

2. PDQ-Analysis Procedure

The PDQ-3 analysis was carried out in X-Y geometry. The following procedure was normally followed:

- a) Two-group constants were obtained from LEOPARD with MND cross sections used for the thermal group. (Table E-2 contains group constants for the critical configurations in which buckling measurements were made.)
- b) A mesh spacing corresponding to two spaces per fuel rod was used in the core except near region boundaries.
- c) Near the boundary between different fuels, at interfaces near perturbations, and at the core-reflector interface the mesh spacing was increased. The number of spaces was doubled over

Table E-1

LEOPARD Output for the 0.56-inch $\text{PuO}_2\text{-UO}_2$ Critical Lattice.

- UNIT CELL DESCRIPTION -

INDEX	NAME	PELLET	CLAD	MODERATOR	EXTRA
99	URAN'UM OXIDE	9.80000E-01	0.	0.	0.
3	ZIRCONIUM-2	0.	-1.72250E-01	0.	0.
100	LIGHT WATER	0.	0.	1.00000E 00	0.

TRACE ELEMENTS AND CORRESPONDENCE FACTORS

INDEX	NAME	FACTOR	FACTOR BY WEIGHT
18	URANIUM-235	71.40000E-04	65.84834E-04
21	PLUTONIUM239	63.66189E-03	59.71326E-03
22	PLUTONIUM240	60.52314E-04	57.00706E-04
23	PLUTONIUM241	63.33817E-05	59.90749E-05
24	PLUTONIUM242	28.15030E-06	26.73631E-06
2	OXYGEN	14.07515E-02	0.

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Table D-2

Calculated Reactivity, Buckling, Reflector Savings,
and Fuel Rod Requirements for Single-Region Cores

Core	Boron Content (ppm)	PDQ-3 k_{eff}	LEOPARD B_M^2 , cm^{-2}	Reflector Savings, cm	Number of rods for $k_{eff} = 1.0$	Number of rods for $k_{eff} = 1.025$
400 PuO_2 - UO_2 rods in a square array	0	1.04960	0.012914	7.662	312	355
	325	1.00688	0.012372	6.723	386	435
	750	0.97139	0.011322	6.249	472	539
	1500	0.92204	0.009876	5.738	615	711
324 UO_2 rods in a square array	0	0.98953	0.013522	6.929	356	-

Table D-3

Comparison of Reactivity and Computer Time Requirements
for Two Methods of Analysis

Geometry (Code)	One-Region Core			Two-Region Core		
	Radial (AIM-5)	Radial (X-Y, PDQ)	SQ (PDQ)	Radial (AIM-5)	Radial (X-Y, PDQ)	SQ (PDQ-3)
k_{eff}	1.026525	1.030828	1.025576	1.023077	1.025550	1.026433
PuO_2 - UO_2 Area, cm^2	706.1	706.1	706.1	277.2	277.2	277.2
No. PuO_2 - UO_2 rods	349	349	349	137	137	137
UO_2 Area, cm^2	-	-	-	461.3	461.3	461.3
No. UO_2 rods	-	-	-	228	228	228
Computer Time, min.	0.48	0.96	-	0.48	0.78	0.78

of k_{eff} are in good agreement and the difference in computer time was sufficiently small that it was not a factor in the selection of the most desirable method of analysis. Consequently, PDQ was selected as the basic analytic method to be used in the post-critical comparison of analysis with experiment.

To simulate the Saxton design as closely as possible with the available fuel rods, it was expected that all rods except those required for power measurements would be used in the borated, two-region cores. In the analysis, a configuration consisting of 361 mixed oxide ($\text{PuO}_2\text{-UO}_2$) fuel rods in the inner region (19 x 19) with 368 UO_2 fuel rods in the outer region (forming a core 27 x 27) was assumed. The LEOPARD - PDQ-3 results for three variations in boron content are shown in Figure D.3. An allowance for a possible discrepancy between analysis and experiment for the two-region core was included by weighting the k_{eff} allowance for each region (1.025 for $\text{PuO}_2\text{-UO}_2$, 1.0 for UO_2) by the neutron production for the region as determined in the PDQ-3 calculations containing 1750 ppm boron. For criticality (calculated $k_{\text{eff}} = 1.016$) a boron requirement of 1525 ppm was predicted.

4. Transient Analysis

a. Introduction

The kinetic characteristics of single-zone mixed-oxide ($\text{PuO}_2\text{-UO}_2$) critical assemblies were evaluated and a transient study was carried

out. The study consisted of two parts. In the first round of calculations, the expected fuel and moderator temperatures to be reached in a transient were assumed and reactivity coefficients were determined for these assumed temperatures. The reactivity coefficients and the calculated values for β_{eff} , neutron lifetime, and heat transfer coefficients were then used in a transient analysis code, WIT-5, for various ramp and step reactivity insertion values. (The WIT-5 code is basically a one-region neutron kinetics program that includes the effect of delayed neutrons and thermal feed-back). The code computes the fuel, clad, and moderator temperatures reached during the transient. The second round of calculations consisted of using the calculated temperatures in a second iteration. In this instance, however, a realistic assumption as to the amount of reactivity that could be added was made.

b. First Round Calculations

The Doppler coefficients for three mixed-oxide $\text{PuO}_2\text{-UO}_2$ critical lattices, 0.6-inch, 0.58-inch, and 0.55-inch, were calculated using the LEOPARD code. The code was run for variations in fuel pellet and effective resonance temperature between 500°F and 1500°F with a constant moderator and clad temperature of 212°F. The resulting values of k_{eff} were used to calculate the following Doppler coefficients:

	<u>0.6" Lattice</u> <u>131 H/Pu</u>	<u>0.58" Lattice</u> <u>-18 H/Pu</u>	<u>0.55" Lattice</u> <u>100 H/Pu</u>
α Doppler (0 ppm Boron Fuel at 1000°F)	$1.25 \times 10^{-5}/^{\circ}\text{F}$	$1.34 \times 10^{-5}/^{\circ}\text{F}$	$1.48 \times 10^{-5}/^{\circ}\text{F}$

The moderator temperature coefficient for the three lattices was calculated using the LEOPARD code by varying the moderator temperature from 68°F to 212°F while other temperatures were held constant. Figure D.4 summarizes the results for the 0.6-inch lattice. The moderator coefficient at 135°F ($-1.96 \times 10^{-4}/^{\circ}\text{F}$) was used in the first round transient study of the 0.6-inch lattice.

Values of β_{eff} were determined from the neutron production and delayed neutron characteristics of each of the fissionable isotopes. These values with the calculated prompt neutron lifetimes are shown below:

	<u>0.60" Lattice</u>	<u>0.58" Lattice</u>	<u>0.55" Lattice</u>
β_{eff}	0.00302	0.00306	0.00314
ℓ , microseconds	11.32	10.15	8.50

The first round hazards study using the WIT-5 code was made to determine if the fuel and clad temperatures would reach melting or if the moderator would reach boiling during a transient. The analysis was based on a slightly larger core radius than needed for criticality for the 0.6-inch lattice. The reference configuration is critical at

a core radius of 6.02 inches and a core height of 28.75 inches where the total fuel height is 36.6 inches. Thus, the addition of moderator to cover the core represents a positive reactivity insertion.

Calculations were made for ramp insertions of 6.45 ϕ /sec and 20 ϕ /sec and for a step insertion of \$2.00 for different volumes of water ranging from that contained within the core ($\approx 1.3 \text{ ft}^3$) to approximately 40% of the total contained within the core tank (144 ft^3). Because of the treatment of the heat sink in the code, the effect of a large moderator volume is to simulate the results one might expect from forced moderator circulation. In reality this condition could not occur since even convection is restricted by core support plates. The 6.45 ϕ /sec ramp is the maximum rate of reactivity change that can be obtained by adding moderator above the 28.75 inch level at the expected fill rate.

The results for representative cases are summarized in Table D.4. These cases show that the energy involved in the transient is proportional to the reactivity inserted and the assumed water volume. In all the calculations the fuel and clad reached temperatures that were far below melting. However, in most cases a clad temperature of 212°F was reached which would have produced local boiling of the moderator. Since the void coefficient is negative, the transient would have been terminated sooner than indicated.

Table D.4

Summary of Representative First-Round Transient Calculations

	Case I	Case II	Case III	Case IV
	Initial Power Peak	Final Power Peak		
Moderator Volume (cu. ft.)	1.33	1.33	144*	1.33
Reactivity Insertion	6.5 ϕ /sec	6.5 ϕ /sec	6.5 ϕ /sec	20 ϕ /sec \$2 (\$333/sec)
Length Reactivity Insertion (sec)	56	56	56	155 0.006
Duration of Analysis (sec)	350	350	350	355 150
Power Peak (Mw)	1.56	0.08	6	9 746
Time to Peak (sec)	15.5	55.6	55	5.5 0.083
Time to Subcritical (sec)	18.3	58.1	72	51 0.20
Total Energy Release (Mw-sec)	8	8	541	18 16
Peak Fuel Temperature °F	123	136	616	237 536
Time Fuel Temperature (sec) peak	17.3	230	56	51 0.10
Maximum Clad Temperature °F	118	136	480	237 380
Time Clad Temperature 212°F (sec)	-	-	20	45 0.13
Time of Maximum Clad Temperature (sec)	17.8	230	56	51 0.44
Bulk Moderator Temperature (Max) °F	-	136	125	212 200
Time of Maximum Moderator Temperature (sec)	-	240	300	51 85

* Corresponds to $\approx$ 40% of the total within the tank.

c. Second Round Calculations

The information obtained in the first-round transient calculations were used as a basis for a more realistic second-round analysis.

The temperature distributions determined in WIT-5 were used in the LEOPARD code to generate improved reactivity coefficients. In the second-round transient calculation, the Doppler coefficient was varied as a function of the fuel temperature as shown in Figure D.5. The moderator coefficient at a core average temperature of 100°F was used. In addition, revised heat transfer coefficient between fuel and clad and between clad and coolant were determined.

A 6.45 ϕ /sec ramp was inserted initially. Decreasing values of the ramp were then inserted to account for the decreasing reactivity worth of the moderator as the water level rose above the 28.75 inch starting level. Reactivity changes with water height in the core were determined by two-group hand calculations using LEOPARD group constants. The water was permitted to rise 5.9 inches above the full height of the fuel where the reactivity no longer changed with additional water. The reactivity worth of the water reflector added above the fuel was calculated using AIM-5. After the water height reached 5.9 inches above the fuel, it was assumed that the dump valve was activated and moderator removed in reactivity steps in accordance with the known dump rate. This assumption does not influence the transient.

The following input parameters were used in the analysis:

Water Volume*	2.15 ft ³
Heat transfer coefficient, fuel-to-clad	100 BTU/ft ² -hr-°F
Heat transfer coefficient, clad-to-water	280 BTU/ft ² -hr-°F
Specific heat of fuel	0.064 BTU/lb-°F
Specific heat of clad	0.073 BTU/lb-°F
Fuel weight	303 pounds
Clad weight	57.5 pounds

* Slightly more than that within the core

The results of the second round WIT-5 transient calculations are summarized in Figure D.6. From this figure it can be seen that the Doppler coefficient is the most important factor in terminating the transient. As the fuel temperature begins to rise near 15 seconds, the rate of reactivity removal due to Doppler exceeds the 6.45 ϕ /sec reactivity addition rate due to the rising moderator. Thus, the k_{eff} curve peaks and begins to descend as the fuel temperature increases. At approximately the 23 second mark, the system becomes subcritical with the Doppler effect compensating for $\approx$ 63% of the total reactivity added.

Figure D-6

Hazard Study - WIT 5

Max. Power = 0.91 MW at 15.7 Seconds
 Second Burst = 0.12 MW at 73.8 Seconds
 Energy Release = 17.1 MW-Seconds

Calculations were also made to determine the effect on the transient for an error in the size of the Doppler coefficient. The WIT-5 calculations were repeated for 0.8 and 1.2 times the variable Doppler coefficient used in second round calculations. Other input parameters were unchanged. The results of the two calculations are compared in Figure D.7. The difference in k_{eff} is small with both transients ending at about the same time. The smaller Doppler coefficient does not result in a significant change in the maximum fuel temperature reached in the transient.

d. Comparison of UO_2 and $\text{PuO}_2\text{-UO}_2$ Transients

To compare the kinetic response of a critical configuration composed of the mixed oxide fuel ($\text{PuO}_2\text{-UO}_2$) to one containing UO_2 alone, a duplicate kinetics study to that previously described was carried out for a single-region 0.60-inch lattice, UO_2 critical experiment. The Doppler and moderator coefficients were calculated using the LEOPARD code. The code was run for variations in fuel pellet and effective resonance temperatures between 68°F and 500°F with a constant moderator and clad temperature of 68°F. The resulting values of k_{eff} were used to calculate the Doppler coefficients shown in Figure D.8 and used in WIT-5. The moderator temperature coefficient was calculated using the LEOPARD code by varying the moderator temperature from 68°F to 212°F at a constant fuel temperature of 205°F and a constant clad temperature of 140°F. Figure D.8 summarizes the results. The moderator temperature coefficient at 95°F ($-9.5 \times 10^{-5}/^\circ\text{F}$) was used in WIT-5.

The prompt neutron lifetime (18.35 microseconds) and β_{eff} (0.008) were determined from LEOPARD. β_{eff} was determined by weighting the neutron production from each fissionable isotope by its delayed neutron characteristics.

All other input parameters to WIT-5 were the same as those used in the analysis of the mixed oxide ($\text{PuO}_2\text{-UO}_2$) critical. The results of the two calculations are compared in Figure D.9. Figure D.9 shows that for the same reactivity addition the maximum fuel temperature and installed system reactivity reach higher values in the UO_2 critical than in the mixed oxide ($\text{PuO}_2\text{-UO}_2$) critical. This result is attributed to the larger Doppler coefficient in the core containing plutonium. The study shows that in the comparison of transients terminated by an inherent shutdown mechanism such as the Doppler effect, the magnitude of the shutdown mechanism is significant whereas differences in prompt neutron lifetime and β_{eff} are not.

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TEMPERATURES, DEGREES CENTIGRADE

PELLET	CLAD AND VOID	MODERATOR	RESONANCE-EFFECTIVE
18.00	18.00	18.00	18.00

SQUARE CELL GEOMETRY

DISTANCES IN INCHES					UNIT CELL VOLUME FRACTIONS		
PELLET OR	CLAD IR	CLAD OR	MODERATOR OR	PITCH	PELLET	CLAD	MODERATOR
0.16870	0.17225	0.19550	0.315935	0.55998	0.2851	0.0978	0.61710
(0.16870 COLD) (0.17225 COLD) (0.19550 COLD)					(0.56000 COLD)		

MODERATING RATIOS

CELL WATER/OXIDE	CELL HOT-WATER/OXIDE	CELL WATER/METAL	CELL HOT-WATER/METAL	CELL H/U-235
0.216442E 01	0.216092E 01	0.453273E 01	0.452540E 01	0.960463E 03
CORE WATER/OXIDE	CORE HOT-WATER/OXIDE	CORE WATER/METAL	CORE HOT-WATER/METAL	CORE H/U-235
0.216442E 01	0.216092E 01	0.453273E 01	0.452540E 01	0.960463E 03

TWO-STEP OMEGA SEARCH PARAMETERS

DOPPLER	U-238 RES.INT.	CALC.OMEGA'	CONVERGED OMEGA'	MUFT OMEGA
0.230221E 01	0.222615E 02	0.166884E-00	0.166904E-00	0.213957E-00
DANCOFF	PU-240 L-FACTOR	L-U238	P-28	PR3-28 (CANDLE)
0.806673E 00	0.637917E 00	0.649442E 00	0.856984E 00	0.852938E 00

-MISCELLANEOUS-

H2-O DENSITY	D2-O DENSITY	U-O2 DENSITY	PUD2 DENSITY	THO2 DENSITY	MAXWELLIAN 1/V
0.998383E 00	0.100000E 01	0.880546E 00	0.960000E 00	0.960000E 00	0.889842E 00
H2O PRESSURE	D2O PRESSURE	BUCKLING	EFF. CHORD	XI-SCATTERING	WIGNER-WILKINS 1/V
0.147000E 02	0.100000E 01	0.122100E-01	0.106236E 01	0.860487E 00	0.633414E 00
4-GROUP K	K 'ONE'	K 'TWO'	K 'THREE'	K 'FOUR'	K INFINITY
1.0083887E 00	6.163246E-02	1.782058E-02	1.876338E-01	7.413018E-01	1.445974E 00
LOADING	POWER DENSITY	POWER DENSITY	SPECIFIC POWER	FAST FLUX	THERMAL FLUX
(G/CC)	(KW/LITER)	(KW/CU.FT.)	(KW/G)	(ABSOLUTE)	(ABSOLUTE)
2.5424E 00	0.	0.	0.	2.013365E 01	1.419304E 00

MATERIAL BUCKLING= 1.25026E-02

BUILT-IN K-BIAS IS UNITY

INDEX	ELEMENT NAME	NUMBER DENSITIES			DIFFERENTIAL REMOVAL CROSS SECTIONS		
		FLUX WTD.	VOLUME WTD.	RATIO	GROUP ONE	GROUP TWO	GROUP THREE
1	HYDROGEN	4.845606E-02	4.120478E-02	1.17598	1.545176E 00	1.990694E 00	1.881916E 00
2	OXYGEN	3.276217E-02	3.346518E-02	0.97899	2.572672E-01	1.229228E-01	6.912150E-02
3	ZIRCONIUM-2	3.226312E-03	3.705712E-03	0.87063	5.363814E-01	2.271111E-01	1.586161E-01
18	URANIUM-235	2.798271E-05	4.290096E-05	0.65226	-2.599395E 00	1.336863E 01	-8.292709E-02
20	URANIUM-238	3.958052E-03	5.965638E-03	0.66348	1.652542E 00	7.126783E-02	-3.534448E-01
21	PLUTONIUM239	2.192724E-04	3.825149E-04	0.57324	1.039169E 00	1.304458E 00	-1.427984E 01
22	PLUTONIUM240	2.475374E-05	3.636556E-05	0.68069	-3.320595E 00	1.632254E 01	-7.042627E 01
23	PLUTONIUM241	2.378942E-06	3.805698E-06	0.62510	-4.328081E 01	1.580437E 02	1.208760E 02
24	PLUTONIUM242	1.127666E-07	1.691422E-07	0.66670	-1.004873E 03	3.573989E 03	3.379732E 03
100	LIGHT WATER				3.347620E 00	4.104312E 00	3.832953E 00
101	HEAVY WATER				2.572672E-01	1.229228E-01	6.912150E-02
304	304STAINLESS				0.	0.	0.
316	316STAINLESS				0.	0.	0.
348	348STAINLESS				0.	0.	0.

REMOVAL CROSS SECTIONS ARE FITTED FOR HYDROGEN

ELEMENT		MICROSCOPIC TRANSPORT CROSS SECTIONS									
INDEX	NAME	GROUP ONE		GROUP TWO		GROUP THREE		THERMAL		THERMAL MND	
1	HYDROGEN	2.0119278E	00	3.1838612E	00	5.4347886E	00	2.2802493E	01	2.3610444E	01
2	OXYGEN	1.3992433E	00	3.6417867E	00	4.1007680E	00	4.0249867E	00	5.9747515E	00
3	ZIRCONIUM-2	3.7705939E	00	7.5178806E	00	6.5416254E	00	8.0548861E	00	1.2025167E	01
4	CARBON	1.8531145E	00	3.7668716E	00	4.9070193E	00	4.2537541E	00	6.1725348E	00
6	IRON	2.2959152E	00	3.2936089E	00	1.0031892E	01	1.2392264E	01	1.9354778E	01
7	NICKEL	2.6205399E	00	5.4170653E	00	1.7578575E	01	2.0055280E	01	3.1516070E	01
9	ALUMINUM	2.0288884E	00	2.8675360E	00	1.5360256E	00	1.5054792E	00	2.3219334E	00
718	INCONEL-718	2.6378831E	00	5.1431203E	00	1.3978680E	01	1.5499694E	01	2.4476281E	01
15	MANGANESE	2.6932401E	00	3.8236550E	00	6.3631878E	00	8.7631712E	00	1.8549701E	01
18	URANIUM-235	5.6805478E	00	9.6441497E	00	3.6999914E	01	2.6622586E	02	7.2580532E	02
19	URANIUM-236	5.2137647E	00	8.3842365E	00	1.7345096E	01	1.3416142E	01	2.2214067E	01
20	URANIUM-238	4.8186225E	00	8.3566254E	00	1.2868809E	01	1.1604636E	01	1.8262127E	01
21	PLUTONIUM239	5.1374744E	00	7.7010074E	00	4.3578650E	01	7.1865838E	02	1.3334078E	03
22	PLUTONIUM240	6.5020856E	00	1.0484824E	01	1.9716786E	02	1.9268059E	02	3.3109441E	02
23	PLUTONIUM241	6.5986307E	00	1.1408942E	01	5.8777230E	01	4.5499164E	02	1.6682481E	03
24	PLUTONIUM242	6.0305346E	00	9.3265123E	00	1.1079426E	02	1.7553700E	01	3.2450293E	01
26	SAMARIUM-149	4.1355214E	00	7.5519768E	00	3.1240553E	01	3.9772825E	03	9.2918129E	04
27	XENON-135	4.1355261E	00	7.8965418E	00	2.1202112E	01	2.4699563E	05	4.5939705E	06
28	FISSION PRD.	0.		0.		2.8509989E	01	5.5934820E	01	1.3819191E	02
29	BORON-10	2.5125185E	00	5.9775355E	00	4.0067015E	01	1.7488627E	03	4.3149829E	03
38	DEUTERIUM	1.8243835E	00	2.3876834E	00	4.1085289E	00	3.6184807E	00	6.2338676E	00
62	THORIUM-232	5.2273544E	00	8.4279946E	00	2.0723580E	01	1.6503495E	01	2.7305566E	01
65	PROTIUM-233	5.3480915E	00	8.3825694E	00	5.5841482E	01	3.3020015E	01	6.4474212E	01
50	URANIUM-233	6.2417147E	00	8.8376441E	00	8.5667919E	01	2.8669607E	02	6.6085280E	02
51	URANIUM-234	5.3017737E	00	8.6543287E	00	1.9022848E	01	6.0154916E	01	1.3440761E	02
100	LIGHT WATER	5.4230989E	00	1.0009509E	01	1.5090345E	01	4.9629972E	01	5.3195639E	01
101	HEAVY WATER	5.0480103E	00	8.4171535E	00	1.2317826E	01	1.1261948E	01	1.8442487E	01
304	304STAINLESS	2.3941652E	00	3.5912426E	00	1.0944778E	01	1.3368628E	01	2.0995287E	01
316	316STAINLESS	2.3361745E	00	3.5577848E	00	1.0878238E	01	1.3232947E	01	2.0789160E	01
348	348STAINLESS	2.3928356E	00	3.6142017E	00	1.1032393E	01	1.3450114E	01	2.1125449E	01

ELEMENT		MICROSCOPIC ABSORPTION CROSS SECTIONS					
INDEX	NAME	GROUP ONE	GROUP TWO	3-SMOOTH	RESONANCE	THERMAL	THERMAL MND
1	HYDROGEN	3.8101373E-05	1.8648501E-04	1.1677135E-02	0.	2.1028857E-01	3.3199248E-01
2	OXYGEN	3.5459640E-02	0.	0.	0.	1.2668273E-04	1.9999999E-04
3	ZIRCONIUM-2	6.3721139E-02	2.1289823E-02	1.1043982E-02	8.2510232E-02	1.1402242E-01	1.8001256E-01
4	CARBON	0.	0.	0.	0.	2.9142604E-03	4.6008800E-03
6	IRON	2.3440361E-03	0.	9.1357780E-02	0.	1.6024451E 00	2.5298556E 00
7	NICKEL	0.	0.	1.7401434E-01	0.	2.9138622E 00	4.6002514E 00
9	ALUMINUM	9.1537141E-03	3.4422117E-03	2.5042810E-02	0.	1.4567320E-01	2.2998114E-01
718	INCONEL-718	4.4114759E-04	0.	1.9171520E-01	0.	2.4495150E 00	3.8671646E 00
15	MANGANESE	2.4694053E-02	0.	9.2459455E-01	0.	8.3605825E 00	1.3199245E 01
18	URANIUM-235	1.2461508E 00	2.0760188E 00	1.5791968E 01	2.2457395E 01	4.0714408E 02	6.4277753E 02
19	URANIUM-236	8.2319216E-01	2.9733379E-01	2.4164003E 01	0.	3.8004820E 00	6.0000000E 00
20	URANIUM-238	4.4557199E-01	2.6733068E-01	6.3558789E-02	2.0728525E 00	1.7290481E 00	2.7297297E 00
21	PLUTONIUM239	1.9311236E 00	2.2951927E 00	2.4694365E 01	1.8736517E 01	1.1508165E 03	1.8168483E 03
22	PLUTONIUM240	2.0369662E 00	2.1775690E 00	1.5507771E 01	1.8527566E 02	2.1241566E 02	3.3535061E 02
23	PLUTONIUM241	2.0984989E 00	3.3445933E 00	4.1107553E 01	2.6000224E 01	1.0612291E 03	1.6754123E 03
24	PLUTONIUM242	1.5666875E 00	7.8587634E-01	8.3106402E 00	9.9243045E 01	1.2412221E 01	1.9595758E 01
26	SAMARIUM-149	0.	2.8725669E-01	2.5617279E 02	0.	5.5027844E 04	8.6875049E 04
27	XENON-135	0.	0.	2.8362103E 02	0.	1.9465660E 06	3.0731354E 06
28	FISSION PRD.	0.	0.	2.8429999E 01	0.	7.7992227E 01	1.2313000E 02
29	BORON-10	4.3975716E-01	2.1594667E 00	1.3511187E 02	0.	2.4317353E 03	3.8390950E 03
38	DEUTERIUM	2.5307848E-02	0.	0.	0.	2.9138622E-04	4.6002515E-04
62	THORIUM-232	1.6332512E-01	2.9074572E-01	1.3608313E-01	8.4381411E 00	4.6421141E 00	7.3287241E 00
65	PROTIUM-233	8.2215145E-01	1.4530211E 00	8.5382644E 01	0.	2.8322713E 01	4.4714401E 01
50	URANIUM-233	1.9461193E 00	3.1251345E 00	3.6554816E 01	5.1661885E 01	3.6629083E 02	5.7828057E 02
51	URANIUM-234	1.4787276E 00	5.3099657E-01	5.8314849E 01	0.	6.6526352E 01	1.0502829E 02
100	LIGHT WATER	3.5535843E-02	3.7297002E-04	2.3354269E-02	0.	4.2070383E-01	6.6418495E-01
101	HEAVY WATER	8.6075336E-02	0.	0.	0.	7.0945517E-04	1.1200503E-03
304	304STAINLESS	2.6290635E-03	0.	1.1824105E-01	0.	1.9036958E 00	3.0054542E 00
316	316STAINLESS	2.4895934E-03	0.	1.1802570E-01	0.	1.8957662E 00	2.9929354E 00
348	348STAINLESS	2.5875741E-03	0.	1.1923424E-01	0.	1.9190404E 00	3.0296795E 00

ELEMENT		MICROSCOPIC FISSION CROSS SECTIONS				
INDEX	NAME	GROUP ONE	GROUP TWO	GROUP THREE	THERMAL	THERMAL MND
18	URANIUM-235	1.1288585E 00	1.6268337E 00	2.4894781E 01	3.4680772E 02	5.4752168E 02
19	URANIUM-236	7.6426552E-01	8.0756545E-03	0.	0.	0.
20	URANIUM-238	3.8407383E-01	3.8049905E-04	0.	0.	0.
21	PLUTONIUM239	1.9057997E 00	1.8651300E 00	2.6203145E 01	7.7509515E 02	1.2236792E 03
22	PLUTONIUM240	1.5899357E 00	2.3712835E-01	6.9954962E-02	2.3358923E-02	3.6877831E-02
23	PLUTONIUM241	1.7200804E 00	2.7414690E 00	5.3375857E 01	7.9076430E 02	1.2484169E 03
24	PLUTONIUM242	1.4197929E 00	1.4823941E-01	3.0346519E-02	0.	0.
38	DEUTERIUM	2.5307848E-02	0.	0.	0.	0.
62	THORIUM-232	9.2809383E-02	0.	-0.	0.	0.
65	PROTIUM-233	7.6322482E-01	8.0756545E-03	0.	0.	0.
50	URANIUM-233	1.8765723E 00	2.9449511E 00	7.5273775E 01	3.4204246E 02	5.3999854E 02
51	URANIUM-234	1.4198010E 00	2.4173844E-01	0.	0.	0.

ELEMENT		MICROSCOPIC NEUTRON FISSION CROSS SECTIONS				
INDEX	NAME	GROUP ONE	GROUP TWO	GROUP THREE	THERMAL	THERMAL MND
18	URANIUM-235	3.0903150E 00	3.9848788E 00	6.0445797E 01	8.4704318E 02	1.3372670E 03
19	URANIUM-236	2.1302373E 00	1.9721444E-02	0.	0.	0.
20	URANIUM-238	1.0922316E 00	9.1635848E-04	0.	0.	0.
21	PLUTONIUM239	6.1289390E 00	5.4330404E 00	7.5465944E 01	2.2290961E 03	3.5191790E 03
22	PLUTONIUM240	4.9831403E 00	6.9711734E-01	2.0196719E-01	6.7437210E-02	1.0646630E-01
23	PLUTONIUM241	5.7584112E 00	8.3190157E 00	1.6067177E 02	2.3548170E 03	3.7176605E 03
24	PLUTONIUM242	4.4538534E 00	4.3672407E-01	8.7612561E-02	0.	0.
38	DEUTERIUM	5.0616300E-02	0.	0.	0.	0.
62	THORIUM-232	2.6123402E-01	0.	-0.	0.	0.
65	PROTIUM-233	2.1353095E 00	1.9767268E-02	0.	0.	0.
50	URANIUM-233	5.2206895E 00	7.4408165E 00	1.8818684E 02	8.5760305E 02	1.3539383E 03
51	URANIUM-234	3.8884925E 00	5.9144353E-01	0.	0.	0.

ONE-FAST-GROUP MICROSCOPIC CROSS SECTIONS

	ELEMENT	ABSORPTION	TRANSPORT	FISSION	REMOVAL	NU-FISSION
1	HYDROGEN	3.4552119E-03	3.0840011E 00	0.	5.4338406E-01	0.
2	OXYGEN	1.1729063E-02	2.4314149E 00	0.	1.9958130E-02	0.
3	ZIRCONIUM-2	5.6190529E-02	5.4802303E 00	0.	4.5798782E-02	0.
4	CARBON	0.	2.9555500E 00	0.	0.	0.
6	IRON	2.7153971E-02	3.4677048E 00	0.	0.	0.
7	NICKEL	5.0244870E-02	4.6993709E 00	0.	0.	0.
9	ALUMINUM	1.1568371E-02	2.0674062E 00	0.	0.	0.
718	INCONEL-718	5.5501732E-02	4.5448605E 00	0.	0.	0.
15	MANGANESE	2.7513537E-01	3.7355070E 00	0.	0.	0.
18	URANIUM-235	1.2246206E 01	9.4799807E 00	8.1805005E 00	-2.3944354E-02	1.9991498E 01
19	URANIUM-236	7.3625310E 00	7.9700185E 00	2.5587047E-01	0.	7.1212705E-01
20	URANIUM-238	8.6596595E-01	7.3199458E 00	1.2718572E-01	-1.0205359E-01	3.6162849E-01
21	PLUTONIUM239	1.4052280E 01	8.3044422E 00	8.9059367E 00	-4.1231595E 00	2.5884513E 01
22	PLUTONIUM240	5.9476482E 01	1.1283428E 01	6.3632977E-01	-2.0334868E 01	1.9718450E 00
23	PLUTONIUM241	2.1343390E 01	1.1313520E 01	1.7023786E 01	3.4901706E 01	5.1462347E 01
24	PLUTONIUM242	3.1872266E 01	1.0177906E 01	5.3479356E-01	9.7586313E 02	1.6646763E 00
26	SAMARIUM-149	7.4076584E 01	7.1629016E 00	0.	0.	0.
27	XENON-135	8.1892687E 01	7.0528987E 00	0.	0.	0.
28	FISSION PRD.	8.2088732E 00	9.8739370E 01	0.	0.	0.
29	BORON-10	3.9979287E 01	4.9380506E 00	0.	0.	0.
38	DEUTERIUM	8.3711323E-03	2.4334518E 00	8.3711323E-03	0.	1.6742465E-02
62	THORIUM-232	2.6403687E 00	8.1728849E 00	3.0698763E-02	0.	8.6408949E-02
65	PROTIUM-233	2.5478172E 01	8.8960277E 00	2.5552624E-01	0.	7.1382222E-01
50	URANIUM-233	2.7304473E 01	1.0058606E 01	2.3475773E 01	0.	5.8895034E 01
51	URANIUM-234	1.7528979E 01	8.2282451E 00	5.6160925E-01	0.	1.5112423E 00
100	LIGHT WATER	1.8639487E-02	8.5994170E 00	0.	1.1067262E 00	0.
101	HEAVY WATER	2.8471327E-02	7.2983185E 00	0.	1.9958130E-02	0.
304	304STAINLESS	3.5010522E-02	3.6798826E 00	0.	0.	0.
316	316STAINLESS	3.4902207E-02	3.6145354E 00	0.	0.	0.
348	348STAINLESS	3.5283570E-02	3.6889949E 00	0.	0.	0.

NEUTRON BALANCE, CONTROLLED SPECTRUM

ELEMENT	FAST CAPTURE	FAST FISSION	SLOW CAPTURE	SLOW FISSION	TOTAL
1 HYDROGEN	2.864607E-03	0.	1.445305E-02	0.	1.731766E-02
2 OXYGEN	7.897675E-03	0.	5.8868E7E-06	0.	7.903562E-03
3 ZIRCONIUM-2	4.189650E-03	0.	5.217859E-04	0.	4.711435E-03
18 URANIUM-235	3.509505E-03	7.061383E-03	2.394774E-03	1.376494E-02	2.673060E-02
20 URANIUM-238	8.867782E-02	1.526645E-02	9.7069E2E-03	0.	1.136512E-01
21 PLUTONIUM239	3.960865E-02	6.854422E-02	1.168546E-01	2.410653E-01	4.660728E-01
22 PLUTONIUM240	4.305335E-02	4.656026E-04	7.457151E-03	8.201425E-07	5.097696E-02
23 PLUTONIUM241	3.307662E-04	1.303567E-03	9.126206E-04	2.668250E-03	5.215204E-03
24 PLUTONIUM242	1.066495E-04	1.820040E-06	1.985258E-06	0.	1.104548E-04
GEOMETRIC LEAKAGE	2.962176E-01		3.901655E-03		3.001193E-01
CONTROL LEAKAGE	7.097382E-03		9.348381E-05		7.190866E-03

MACROSCOPIC EDIT

	ONE/THREE	TWO/THREE	THREE/THREE	CNE/ONE	THERMAL	THERMAL MND
DIFFUSION	1.905674E 00	9.963359E-01	6.798460E-01	1.205736E 00	2.252877E-01	1.877143E-01
ABSORPTION	4.956957E-03	2.740435E-03	3.932288E-02	1.403641E-02	2.889249E-01	4.561393E-01
REMOVAL	8.395482E-02	9.027856E-02	7.134110E-02	2.059902E-02		
NU-FISSION	9.196728E-03	2.311725E-03	3.207430E-02	1.318273E-02	5.180856E-01	8.179261E-01
K-FISSION	6.884278E-01	2.186329E-01	2.206749E 00	8.990809E-01	3.614546E 01	5.706454E 01
FISSION	3.133276E-03	8.045843E-04	1.129520E-02	4.603908E-03	1.815434E-01	2.866111E-01
FLUXES	6.701563E 00	7.708823E 00	5.849973E 00	2.026036E 01	1.430848E 00	9.103731E-01
AGE	2.143331E 01	1.071110E 01	6.143336E 00	3.481223E 01		
P	9.442486E-01	9.705390E-01	6.446641E-01	5.947384E-01		

FOR CUTOFF AT .625, RHO-28 IS 10.6886, AND DELTA-25 IS 0.5121

RATIO OF FERTILE CAPTURES TO FISSION ABSORPTIONS	0.2990	'FERTILE' MEANS TH-PA+U234+U238+PU240.
RATIO OF FERTILE FISSIONS TO FISSION ABSORPTIONS	0.0316	'FISSION' MEANS U233+U235+PU239+PU241.
RATIO OF U-238 CAPTURES TO FISSION ABSORPTIONS	0.1976	
RATIO OF U-238 CAPTURES TO U-235 FISSIONS	4.7241	

PROMPT NEUTRON LIFETIME IS 5.04 MICROSECONDS.

ONE VELOCITY	D	SIGMA A	NU-SIGMA F	KAPPA FISSION
	1.14106E 00	3.21693E-02	4.64884E-02	5.31981E-13

TABLE E-2

A Summary of LEOPARD Calculated Parameters for a Number of Different Critical Lattices Using $\text{PuO}_2\text{-UO}_2$ and UO_2 Fuels

<u>Configuration Description</u>									
Fuel	$\text{PuO}_2\text{-UO}_2$	$\text{PuO}_2\text{-UO}_2$	$\text{PuO}_2\text{-UO}_2$	$\text{PuO}_2\text{-UO}_2$	$\text{PuO}_2\text{-UO}_2$	$\text{PuO}_2\text{-UO}_2$	UO_2	UO_2	UO_2
Pitch, inches	0.52	0.56	0.56	0.735	0.792	1.04	0.52	0.56	0.792
Temperature, °C	25.8	16.4	18.0	24.1	16.1	19.9	19.2	18.0	17.3
Experimental Buckling	0.01088	0.01215*	0.01123	0.01596	0.01593	0.01284	0.01175	0.01271	0.01368
Boron, ppm	1.6	0	337	1.6	0	1.6	0	0	0
<u>Calculated Parameters</u>									
k_{eff}	0.98903	1.01026	1.01482	1.01283	1.01749	1.01668	0.99358	1.00589	1.00205
k_{∞}	1.39147	1.44622	1.41778	1.53362	1.53015	1.41941	1.42349	1.45553	1.43628
Material Buckling	0.01052	0.01251	0.01174	0.01649	0.01667	0.01350	0.01154	0.01292	0.01374
g Hydrogen	1.1989	1.1762	1.1758	1.1101	1.0975	1.0606	1.1296	1.1204	1.0756
g Pu-239	0.5911	0.5731	0.5720	0.5240	0.5097	0.4654			
g U-235	0.6838	0.6519	0.6525	0.5607	0.5363	0.4699	0.8113	0.7790	0.6539
Wigner Wilkins 1/v	0.5881	0.6344	0.6287	0.7382	0.76637	0.81330	0.6288	0.6679	0.78349
Pu-240 L-factor	0.6937	0.6377	0.6379	0.4758	0.4389	0.3306	(0.6991)	(0.6461)	(0.4496)
Fast Flux	22.020	20.126	20.411	16.907	16.542	16.616	21.881	20.225	17.464
Thermal Flux	1.097	1.419	1.414	3.201	3.920	8.021	2.247	2.702	6.183
Fertile Captures/Fissile Absorptions	0.3539	0.2989	0.3031	0.1825	0.1633	0.1200	0.3076	0.2645	0.1584
Fertile Fissions/Fissile Absorptions	0.0391	0.0316	0.0319	0.0164	0.0140	0.0083	0.0430	0.0354	0.0165
U-238 Captures/Fissile Absorptions	0.2367	0.1975	0.2003	0.1183	0.1060	0.0776	0.3076	0.2645	0.1584
<u>Group Constants</u>									
D_1	1.20656	1.20575	1.20942	1.20421	1.20500	1.22575	1.14335	1.15723	1.19109
D_2 (MND)	0.19886	0.18674	0.18703	0.17830	0.17155	0.16948	0.20367	0.19515	0.17587
Σ_{a1}	0.01557	0.01404	0.01418	0.00907	0.00801	0.00510	0.01278	0.01143	0.00652
Σ_{a2} (MND)	0.58052	0.45560	0.46961	0.21186	0.17404	0.09532	0.29981	0.25340	0.12061
Σ_{r1}	0.01710	0.02062	0.02069	0.03029	0.03246	0.03863	0.01964	0.02302	0.03438
Σ_{r2}	0.01478	0.01319	0.01322	0.00822	0.00719	0.00435	0.01189	0.01052	0.00564
Σ_{f2} (MND)	1.04362	0.81703	0.82302	0.36457	0.29331	0.14237	0.52333	0.43627	0.18620
τ	36.931	34.791	34.686	30.598	29.778	28.029	35.425	33.593	29.123
ρ	0.52345	0.59488	0.59330	0.76953	0.80207	0.88328	0.60585	0.66816	0.84067

* Average of two buckling measurements.

a region equivalent to one fuel rod on each side of the boundary.

On the PDQ-3 analysis to determine the relative flux distributions, four group calculations were made with conventional constants used for the thermal group. Comparisons were made between the normal procedure, two-group calculations with MND constants, and the four-group calculation with conventional thermal constants. The comparison of the two methods from the standpoint of flux distributions is shown in Figure E-1. A comparison from the standpoint of the effect on power distributions is shown in Figure 5.6.

3. Control Rod Calculations

The procedure outlined in Reference 18 was used to determine control rod group constants for use in PDQ-3 calculations of the reactivity and power distribution effects of control rods in the WREC single region and multi-region experiments. These steps were followed:

- a) The current-to-flux ratio, α_1 , at the rod surface was found for each of four energy groups.
- b) A four-group cylindrical AIM-5 calculation was then carried out. The configuration consisted of two regions. The inner boundary corresponded to the control rod surface with the boundary condition for each energy group determined from the values calculated in a). The regions were: 1) a water region surrounding the rod and 2) a homogenized core region ≈ 6 cm wide. A zero-current condition was set at the outer boundary.

- c) The number of absorptions in the rod-cell, control rod and surrounding water, was determined from the flux at the surface of the rod and the current-to-flux ratio and from the absorption cross section and average flux in the water. The total number of absorptions was set equal to and solved for an absorption cross section times an average rod cell flux as inferred from AIM-5. Then the four group absorption cross sections were reduced to two groups by flux weighting.
- d) The rod-cell absorption cross section determined in c) was used in a duplicate PDQ-3 calculation of the rod-cell and surrounding homogenized core region. An iterative procedure was followed adjusting the absorption cross section until the calculated k_{eff} in PDQ-3 was the same as that obtained with AIM-5.
- e) The rod-cell group constants were then used in PDQ-3 calculations for specific rodged experimental configurations.

The following list contains the group constants used for the two fuels for the 0.56 inch lattice.

<u>UO₂ Fuel</u>		<u>PuO₂-UO₂ Fuel</u>	
$D_1 = 1.0$	$D_2 = 0.124$	$D_1 = 1.0$	$D_2 = 0.185$
$\Sigma_{a1} = 0.049495$	$\Sigma_{a2} = 2.17698$	$\Sigma_{a1} = 0.048504$	$\Sigma_{a2} = 1.02843$
$\Sigma_{\text{rem}} = 0.02778$		$\Sigma_{\text{rem}} = 0.027688$	

4. Power Peaking

The procedure used in calculating power peaking effects was described in Section 5.4. A summary of the reactivity and power distribution results for the various methods is contained in Tables E-3 through E-6.

5. Power Sharing

The power sharing experiments and the associated analysis were described in Section 5.5. Core diagrams and tabulated results are shown in Figures E-2 through E-7 and Tables E-7 through E-17.

Table E-3

A Summary of Reactivity and Power Distribution Comparisons
for Water Slot Experiments in a UO_2 Single-Region Core

Ref. Case	Experimental Ref. Case	Analysis Method		Experimental Results
		Soft Spectrum	Extra Region	
k_{eff} : 1.004224 PDQ07		1.007415	1.006567	
$\Delta k = 0$	0	+0.319%	+0.234%	+0.26%
Boundary				
1. 1.4617	-	1.4607	1.4613	
2. 1.0073	-	1.0068	1.0070	-
3. 1.0	1.0	1.0	1.0	1.0
4. 1.0728	-	1.0739	1.0737	-
5. 1.1515	1.1551	1.1548	1.1541	1.1500
6. 1.2201	-	1.2294	1.2278	-
7. 1.2750	1.3052	1.3076	1.3037	1.3071
8. 1.3147	1.3255	1.4571	1.4441	1.4554
9. 1.3387	-	2.0334	1.9750	1.9257
10. 1.3458	-	0	0	0
LEOPARD II, Leonard Cross Sections				

Table E-4

A Summary of Reactivity and Power Distribution Comparisons for
Aluminum Slab Experiments in a UO_2 Single Region Core

Ref. Case PDQ Analysis	Experimental Ref. Case	Analysis Method		Experimental Results
		Soft Spectrum	Extra Region	
k_{eff} : 1.004224		1.001888	1.001655	
$\Delta k = 0$	0	-0.234%	-0.257%	-0.27%
Boundary				
1. 1.4617	-	1.4645	1.4646	-
2. 1.0073	-	1.0084	1.0084	-
3. 1.0	1.0	1.0	1.0	1.0
4. 1.0728	-	1.0719	1.0718	-
5. 1.1515	1.1551	1.1496	1.1494	1.1587
6. 1.2201	-	1.2186	1.2181	-
7. 1.2750	1.3052	1.2816	1.2805	1.2995
8. 1.3147	1.3255	1.3723	1.3690	1.4138
9. 1.3387	-	1.6696	1.6566	1.6550
10. 1.3458				

Table E-5

A Summary of Reactivity and Power Distribution Comparisons
for Water-Slot Experiments in a $\text{PuO}_2\text{-UO}_2$ Single Region Core

Ref. Case PDQ Analysis	Experimental Ref. Case	Analysis Method			Experimental* Results
		Soft Spectrum	Soft Spectrum Spec. Dancoff Correction	Extra Region Spec. Dancoff Correction	
		Reactivity			
$k_{\text{eff}}=1.011479$		1.017449	1.017246	1.016389	
$\Delta k = 0$	0	+0.597%	+0.577%	+0.491%	+0.490%
Rod Position from outer boundary		Power Distributions			
1. 1.7385	-	1.7344	1.7346	1.7353	-
2. 0.9969	-	0.9953	0.9954	0.9956	-
3. 1.0	1.0	1.0	1.0	1.0	1.0
4. 1.0782	1.0786	1.0803	1.0802	1.0799	1.0792
5. 1.1545	1.1616	1.1597	1.1595	1.1588	-
6. 1.2448	1.2295	1.2294	1.2292	1.2278	1.2167
7. 1.2704	1.2786	1.2958	1.2953	1.2924	1.3004
8. 1.3073	1.2945	1.4249	1.4236	1.4138	1.4655
9. 1.3297	1.3358	2.2158	2.2117	2.1469	2.0604
10. 1.3363	-	0	0	0	0

* Power distributions are averages of two experiments

Table E-6

A Summary of Reactivity and Power Distribution Comparisons for
Aluminum Slab Experiments in a $\text{PuO}_2\text{-UO}_2$ Single Region Core

Ref. Case PDQ Analysis	Experimental Ref. Case	Analysis Method		Experimental** Result
		Soft Spectrum Alum. Slab	Extra Region Alum. Slab	
		Reactivity		
k_{eff} : 1.011479	0	1.021657*	1.01063	-0.04%
Δk : 0	0	-0.049%	-0.0851%	
		Power Distributions		
Rod Position from Outer Boundary				
1. 1.7385	-	1.7516	1.7397	
2. 0.9969	1.0	0.9989	0.9972	-
3. 1.0	-	1.0	1.0	1.0076
4. 1.0782	1.0786	1.0775	1.0778	1.0698
5. 1.1545	1.1616	1.1534	1.1526	1.1553
6. 1.2448	1.2295	1.2182	1.2181	-
7. 1.2704	1.2786	1.2739	1.2731	1.2938
8. 1.3073	1.2945	1.3543	1.3498	-
9. 1.3297	1.3358	1.7678	1.7408	1.6879
10. 1.3363	-	0	0	0

* LEOPARD II, Leonard Cross Section
Ref Case: $k_{\text{eff}} = 1.021269$

** Power distributions are averages of two experiments

Figure E-1 - A Comparison of Calculated and Measured Flux Distributions Using a Two-Group and Four-Group PDQ Analysis

Configuration 1(a)
 19x19 Core
 11x11 PuO₂-UO₂ Center Region
 UO₂ Outer Region
 Date: 4/29/65

○ UO₂
 ● PuO₂-UO₂
 S Standard
 M Monitor

Configuration 1 (b)
 19x19 Core
 11x11 PuO₂-UO₂ Center Region
 UO₂ Outer Region
 Date: 6/3/65

○ UO₂
 ● PuO₂-UO₂
 S Standard
 M Monitor

Figure E-2 - Core Diagrams for Power Maps in Configurations 1(a) and 1(b)

Configuration 2
 27 x 27 Core
 19 x 19 PuO₂-UO₂ in Center Region
 UO₂ Outer Region
 1453 ppm Boron
 Date: 5/14/65

● PuO₂-UO₂
 ○ UO₂
 M Monitor
 S Standard

Configuration 3
 Core 27x27
 3x3 UO₂ Insert
 19x19 PuO₂-UO₂ Inner Region

UO₂ Outer Region
 1425 ppm Boron
 Date: 5/21/65

○ UO₂
 ● PuO₂-UO₂
 S Standard
 M Monitor

Figure E-3 - Core Diagrams for Power Maps in Configurations 2 and 3

Configuration 4
 19x19 Core
 11x11 UO_2 Center Region
 PuO_2 - UO_2 Outer Region
 Date: 5/6/65

○ UO_2
 ● PuO_2 - UO_2
 S Standard
 M Monitor

Configuration 5
 27x27 Core
 19x19 UO_2 Center Region
 PuO_2 - UO_2 Outer Region
 1252 ppm Boron
 Date: 5/28/65

○ UO_2
 ● PuO_2 - UO_2
 S Standard
 M Monitor

Figure E-4 - Core Diagrams for Power Maps in Configurations 4 and 5

Configuration 6
 19x19 UO_2
 3x3 $\text{PuO}_2\text{-UO}_2$ Insert
 Date: 4/9/65

○ UO_2
 ● $\text{PuO}_2\text{-UO}_2$
 S Standard
 M Monitor

Configuration 7

19x19 $\text{PuO}_2\text{-UO}_2$
 3x3 UO_2 Insert
 Date: 4/26/65

○ UO_2
 ● Pelletized $\text{PuO}_2\text{-UO}_2$
 ■ Vipac $\text{PuO}_2\text{-UO}_2$

Figure E-5 - Core Diagrams for Power Maps in Configurations 6 and 7

Configuration 9
 19x19 Core
 11x11 $\text{PuO}_2\text{-UO}_2$ Center Region
 UO_2 Outer Region
 Aluminum Slab at Region Boundary
 Date: 4/30/65

○ UO_2
 ● $\text{PuO}_2\text{-UO}_2$
 S Standard
 M Monitor

Configuration 10
 Core 21x21
 11x11 $\text{PuO}_2\text{-UO}_2$ Center Region
 UO_2 Outer Region
 Date: 5/4/65

○ UO_2
 ● $\text{PuO}_2\text{-UO}_2$
 □ Ag-In-Cd Rods
 S Standard
 M Monitor

Figure E-6 - Core Diagrams for Power Maps in Configurations 9 and 10

Configuration 11

27x27 Core
 19x19 $\text{PuO}_2\text{-UO}_2$ Center Region
 UO_2 Outer Region

1453 ppm Boron
 Water Slot at Region Boundary
 Date: 5/14/65

○ UO_2
 ● $\text{PuO}_2\text{-UO}_2$
 S Standard
 M Monitor

Configuration 12

27x27 Core
 19x19 $\text{PuO}_2\text{-UO}_2$ Center Region
 UO_2 Outer Region
 Aluminum Slab at Region Boundary

Date: 5/17/65

○ UO_2
 ● $\text{PuO}_2\text{-UO}_2$
 S Standard
 M Monitor

Figure E-7 - Core Diagrams for Power Maps in Configurations 11 and 12

Table E-7 - A Summary of the Calculated and Measured Power
in Configurations 1(a) and 1(b)

CONFIGURATION 1

Core Size: 19x19, 0.56-inch pitch

Analysis

Inner Region: 11x11 $\text{PuO}_2\text{-UO}_2$

$$B_H^2: 9.58764 \times 10^{-4}$$

Outer Region: UO_2

Temp: 20°C

Critical Water Height, cm: 1 a) 88.57

 $k_{eff}: 1.007135$

Experiment Temp:

2-Group PDQ

1 a) 14.9°C

Wescott Cross Sections

1 b) 16.2°C

Flat Side

Type of Fuel Rods	Rod Position From Center	Configuration 1(a)			Configuration 1(b)			Average Relative Power		PDQ-03 Analysis 2-Group MND Cross Sections	PDQ-03 Analysis 4-Group
		Rod Number	Relative Power		Rod Number	Relative Power					
			Thermal Method	Foil Method		Thermal Method	Foil Method	Thermal Method	Foil Method		
PuO ₂ -UO ₂	1	69	1.0		15	1.0		1.0		1.0	1.0
	2	71,80	1.000977		12	0.992694		0.996836		0.99421	0.99427
	3	91	0.982318		10	0.973521		0.977920		0.97727	0.97781
	4	83	0.973143		7	0.965661		0.969402		0.95025	0.95317
	5	85,74	0.955504		4	0.925034		0.940269		0.92153	0.93226
	6	70,72,80	0.978816		1,14,17,18	0.959169		0.968993		0.96227	0.97163
UO ₂	7	5,4,7,8	0.7413	0.7859	32,K,J,L	0.7264	0.7701	0.7339	0.7780	0.78160	0.76934
	8	3	0.7536	0.7990	28	0.7589	0.8045	0.7563	0.8017	0.80597	0.80489
	9	2	0.7918	0.8395	23	0.7919	0.8395	0.7919	0.8395	0.82913	0.85997
	10	1,6	1.1006	1.1669	10	1.1039	1.1703	1.1023	1.1686	1.21203	1.16533

Diagonal

Type of Fuel Rods	Rod Position From Center	Configuration 1(a)			Configuration 1(b)			Average Relative Power		PDQ-03 Analysis 2-Group MND Cross Sections
		Rod Number	Relative Power		Rod Number	Relative Power		Thermal Method	Foil Method	
			Thermal Method	Foil Method		Thermal Method	Foil Method			
PuO ₂ -UO ₂	1	69	1.0		15	1.0		1.0		1.0
	2	73	1.000782		13	0.988110		0.994446		0.98845
	3	81	0.975149		11	0.943276		0.959213		0.954999
	4	84	0.918709		9	0.912593		0.915651		0.902691
	5	89	0.879472		6	0.859423		0.869448		0.847764
	6	97	0.906933		3	0.894029		0.900481		0.899540
UO ₂	7	12	0.6470	0.6860	10	0.6316	0.6697	0.6393	0.6779	0.681077
	8	11	0.5846	0.6198	31	0.5728	0.6073	0.5787	0.6136	0.614188
	9	10	0.5987	0.6348	27	0.5855	0.6207	0.5921	0.6278	0.620541
	10	9	0.9550	1.0125	22	0.9395	0.9960	0.9473	1.0043	1.056116

Table E-8 - A Summary of the Calculated and Measured Power in Configuration 2

CONFIGURATION 2

Core Size: 27x27, 0.56-inch pitch

Analysis

Inner Region: 19x19 PuO₂-UO₂

$$B_H^2 = 8.1242 \times 10^{-4}$$

Outer Region: UO₂

Temp: 20°C

Critical Water Height, cm: 91.72

k_{eff}: 1.013806 (2-Group PDQ-3)

Experiment Temp: 18.2°C

1.009328 (4-Group PDQ-3)

Westcott Cross Sections

Flat Traverse

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)	PDQ-3 Analysis (4-Groups)
			Foil Method	Thermal Method		
PuO ₂ -UO ₂	1	87	1.0		1.0	1.0
	2	7	0.984450		0.9957	0.9957
	3	90	0.989254		0.9833	0.9830
	4	8	0.967851		0.9624	0.9620
	5	92	0.946697		0.9335	0.9329
	6	9,81	0.903751		0.8968	0.8961
	7	95	0.860293		0.8527	0.8522
	8	67	0.801034		0.8019	0.8033
	9	98	0.753496		0.7500	0.7577
	10	68,85,82,83	0.759427		0.7482	0.7555
UO ₂	11	67,79	0.5450	0.5718	0.5753	0.5613
	12	68	0.5324	0.5586	0.5579	0.5501
	13	69	0.5011	0.5258	0.5104	0.5237
	14	70	0.5483	0.5753	0.5785	0.5752

Diagonal

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
PuO ₂ -UO ₂	1	87	1.0		1.0
	2				0.9915
	3				0.9667
	4	78	0.942415		0.9262
	5				0.8715
	6	75	0.805276		0.8044
	7				0.7271
	8	73	0.648009		0.6433
	9				0.5626
	10	72,84	0.560440		0.5485
UO ₂	11	76,80	0.3650	0.3830	0.3804
	12	75	0.2999	0.3146	0.3081
	13	74	0.2633	0.2736	0.2524
	14	73	0.2833	0.2969	0.2966

Table E-9 - A Summary of the Calculated and Measured Power
in Configuration 3

CONFIGURATION 3

Core Size: 27x27, 0.56-inch pitch

Inner Region: 19x19 PuO_2 - UO_2

Outer Region: UO_2

Critical Water Height, cm: 87.5

Experiment Temp: 18.5°C

3x3 UO_2 Insert

1425 ppm Boron

Analysis

$$B_H^2: 9.96 \times 10^{-4}$$

Temp: 18°C

k_{eff} : 1.009672

2-Group PDQ-3

Westcott Cross Sections

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO_2	1	I	0.905039	0.850332	0.91859
	2	L	0.853348	0.801766	0.85550
PuO_2 - UO_2	3	5,2	1.0		1.0
	4	6	0.916649		0.89557
	5	7	0.883166		0.86186
	6	8	0.838671		0.82796
	7				0.78750
	8	9	0.756036		0.74102
	9				0.69346
UO_2	10	10	0.705512		0.69218
	11	0	0.547580	0.514480	0.53267
	12				0.51703
	13				0.47362
	14				0.53829

Table E-10 - A Summary of the Calculated and Measured Power in Configuration 4

CONFIGURATION 4

Core Size: 19x19, 0.56-inch pitch

Analysis

Inner Region: 11x11 UO_2

B_H^2 : 1.3203×10^{-3}

Outer Region: $\text{PuO}_2\text{-UO}_2$

Temp: 20°C

Critical Water Height, cm: 74.43

k_{eff} : 1.006606

Experiment Temp: 15.6°C

2-Group PDQ-3

Westcott Cross Sections

Flat Side

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO_2	1	69	1.0		1.0
	2	70	0.99156		0.99403
	3	71,78	0.98492		0.97623
	4	72	0.94333		0.94486
	5	73	0.87577		0.89124
	6	74,67	0.77519		0.77068
$\text{PuO}_2\text{-UO}_2$	7	11,95,96	0.8325	0.8804	0.83020
	8	12	0.6829	0.7222	0.68822
	9	13	0.6716	0.7102	0.67540
	10	17	1.0317	1.0911	1.17393

Diagonal

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO_2	1	69	1.0		1.0
	2	75	0.98068		0.98811
	3	79	0.95174		0.95309
	4	81	0.88712		0.89309
	5	83	0.77971		0.79637
	6	84	0.60269		0.61092
$\text{PuO}_2\text{-UO}_2$	7	90	0.58640	0.62010	0.60561
	8	92	0.49170	0.52000	0.50523
	9	93	0.50590	0.5350	0.49943
	10	94	0.95096	1.00570	1.13192

Table E-11 - A Summary of the Calculated and Measured Power
in Configuration 5

CONFIGURATION 5

Core Size: 27x27, 0.56-inch pitch Analysis
 Inner Region: 19x19 UO_2 B_H^2 : 1.042×10^{-3}
 Outer Region: PuO_2 - UO_2 Temp: 18°C
 Critical Water Height, cm: 85.05 k_{eff} : 1.003622
 Experiment Temp: 18°C 2-Group PDQ-3
 Flat Side Westcott Cross Sections

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO_2	1	0	1.0		0.94359
	2	F	0.94788		0.93991
	3	E	0.92505		0.92915
	4	B	0.90473		0.91113
	5	A	0.88863		0.88622
	6	98	0.84060		0.85433
	7	97	0.81573		0.81573
	8	96	0.76398		0.76940
	9	95	0.69703		0.70977
	10	P, 94, I, 84	0.61105		0.60333
PuO_2 - UO_2	11	81, 79, 80, 72	0.65578	0.68724	0.63899
	12	78	0.52597	0.55120	0.51393
	13	77	0.47712	0.50001	0.46363
	14	76, 82	0.59591	0.62450	0.61932

Diagonal

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO_2	1		1.0		0.94359
	2				0.93632
	3				0.91481
	4		0.88144		0.87971
	5				0.83215
	6		0.76466		0.77346
	7				0.70514
	8		0.60530		0.62749
	9				0.53464
	10		0.38327		0.39461
PuO_2 - UO_2	11		0.37890	0.39708	0.37302
	12		0.28956	0.30345	0.29004
	13		0.25367	0.26584	0.24199
	14		0.34316	0.35962	0.37224

Table E-12 - A Summary of the Calculated and Measured Power
in Configuration 6

CONFIGURATION 6

Core Size: 19x19, 0.56-inch pitch Analysis
 Inner Region: 3x3 PuO_2 - UO_2 B_H^2 : 1.13×10^{-3}
 Outer Region: UO_2 Temp: 20°C
 Critical Water Height, cm: 82.17 k_{eff} : 1.005199
 Experiment Temp: 21.2°C Flat Traverse 2-Group PDQ-3
 Leonard Cross Sections

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
PuO_2 - UO_2 UO_2	1	15	1.31816	1.38839	1.2959
	2	12,14,16,18	1.40361	1.47850	1.4073
	3	30,23	1.19854		1.1811
	4	31,22	1.22875		1.2520
	5				1.2167
	6	32,21	1.14396		1.1511
	7				1.0729
	8	33,20	1.0		1.0
	9				1.0080
	10				1.4667

Diagonal

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
PuO_2 - UO_2 UO_2	1	15	1.31816	1.38839	1.2959
	2	13,17,19	1.47719	1.55589	1.5003
	3	25	1.22337		1.2416
	4	26	1.18774		1.2044
	5				1.1054
	6	27	0.96074		0.9844
	7				0.8546
	8	28	0.73998		0.7433
	9				0.7501
	10				1.2766

Table E-13 - A Summary of the Calculated and Measured Power
in Configuration 7

CONFIGURATION 7

Core Size: 19x19, 0.56-inch pitch

Analysis

Inner Region: 3x3 UO_2

B_H^2 : 1.14×10^{-3}

Outer Region: PuO_2-UO_2

Temp: 20°C

Critical Water Height, cm: 79.96

k_{eff} : 1.021561

Experiment Temp: 14.7°C

2-Group PDQ-3

Leonard Cross Sections

Flat Side

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO_2	1	5	1.15218	1.08050	1.2605
	2	2,6,8	1.08520	1.0177	1.1680
PuO_2-UO_2	3	10,15,32,35	1.35173		1.3521
	4	11,16	1.21503		1.1926
	5	12,17,33,37	1.13967		1.1320
	6				1.0707
	7	13,18	1.0		1.0
	8				0.9277
	9	14,19	0.97209		0.9265
	10				1.6248

Diagonal

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO_2	1	5	1.15218	1.08050	1.2604
	2	1,3,9,7	1.0400	0.97520	1.0950
PuO_2-UO_2	3	20,34	1.23216		1.2187
	4	21	1.11683		1.1200
	5	22,36	1.03651		1.0306
	6				0.9241
	7	23	0.79233		0.8059
	8				0.6935
	9	24	0.71942		0.6896
	10				1.5759

Table E-14
A Summary of the Calculated and Measured Power
in Configuration 9

CONFIGURATION 9

Core Size: 19x19, 0.56-inch pitch	Analysis
Inner Region: 11x11 $\text{PuO}_2\text{-UO}_2$	B_H^2 : 1.13×10^{-3}
Outer Region: UO_2	Temp: 18°C
Critical Water Height, cm: 90.44	k_{eff} : 1.00190
Experiment Temp: 15°C	2-Group PDQ-3
Aluminum Slab at Interface	Westcott Cross Sections
	Extra-region used for Slab

Flat Side

Type of Fuel	Rod Position	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO ₂	1				1.33958
	2				0.91891
	3				0.89484
PuO ₂ -UO ₂	4	21	0.86302 0.81401		0.86861
	5	77	1.07488		1.07010
	6				1.02509
	7				1.05702
	8				1.08676
	9				1.10499
	10	80	1.11080		1.11039
	11	79	1.09698		1.10308
	12	82	1.09713		1.08634
	13	84	1.14046		1.08919
	14	96	1.35787		1.37557
Aluminum	15		0	0	0
UO ₂	16	22	1.11727	1.05381	1.14910
	17	23	0.95674	0.90240	0.946170
	18	24	0.93003	0.87721	0.92677
	19	25	1.29439	1.22088	1.33835

Other Rods

Type of Fuel	Rod Position From Center	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
PuO_2	Around Plate	89	1.34371		1.35270
		88	1.28935		1.24600
		99	1.06969		1.04130
		98	1.20363		1.20870
		86	0.97631		0.94935
		94	1.02654		0.99839
	Other Rod	76	1.0		1.0
UO_2	Around Plate	26	1.09541	1.03320	1.12934
		27	1.02767	0.96931	1.04243
		29	0.88308	0.83293	0.88485
		28	0.90278	0.85150	0.90846
	Other Rods	30	0.87898	0.82906	0.87278
		31	0.81009	0.76408	0.81612
		32	0.78008	0.73578	0.78726
		33	0.74892	0.70638	0.75492
		20	0.75040	0.70778	0.75615

Table E-15

A Summary of the Calculated and Measured Power
in Configuration 10

CONFIGURATION 10

Core Size: 21x21, 0.56-inch pitch

AnalysisInner Region: 11x11 $\text{PuO}_2\text{-UO}_2$

$$B_H^2: 2.354 \times 10^{-3}$$

Outer Region: UO_2

Temp: 18°C

Critical Water Height, cm: 71.86

 k_{eff} : 0.981933

Experiment Temp: 15.5°C

2-Group PDQ-3

5 Control Rods at Interface

Westcott Cross Sections

Flat Side

Type of Fuel	Rod Position	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO_2	1				1.18959
	2				0.81926
	3				0.80993
	4				0.85054
$\text{PuO}_2\text{-UO}_2$	5	F	0.81833	0.77206	0.82487
	6	83	1.0		1.0
	7				0.93933
	8				0.95059
	9				0.95894
	10				0.95551
	11	82	0.93537		0.93884
	12	81	0.86971		0.90823
	13	78	0.85192		0.86254
	14	75	0.77731		0.79719
	15	68	0.62629		0.68064
Cont. Rod	16	O			
UO_2	17	E	0.53087	0.50085	0.54956
	18	D	0.61046	0.57594	0.62384
	19	C	0.61958	0.58454	0.61391
	20	B	0.65519	0.61814	0.63288
	21	A	0.94673	0.89320	0.93184

Table E-16 - A Summary of the Calculated and Measured Power
in Configuration 11

CONFIGURATION 11

Core Size: 27x27, 0.56-inch pitch

Inner Region: 19x19 $\text{PuO}_2\text{-UO}_2$

Outer Region: UO_2

Critical Water Height, cm: 98.19

Experiment Temp: 18°C

Water-slot at Interface

1453 ppm Boron

Analysis

$$B_H^2: 8.1242 \times 10^{-4}$$

Temp: 18°C

k_{eff} : 1.011073

2-Group PDQ-3

Westcott Cross Sections

Extra-region Used for Water-slot

Flat Side

Type of Fuel	Rod Position	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO_2	1	97	0.78106	0.73835	0.76640
	2	96	0.69938	0.66113	0.67915
	3	95	0.74516	0.70440	0.74402
	4	94	0.75971	0.71816	0.76838
$\text{PuO}_2\text{-UO}_2$	5	77	1.0		1.0
	6				1.00330
	7				1.07323
	8				1.14151
	9				1.20080
	10	79	1.24040		1.24993
	11				
	12				
	13				
	14	80	1.30456		
	15				
	16	82	1.29876		
	17				1.28234
	18	83	1.22572		1.24167
	19	84	1.16638		1.19052
	20	85	1.13137		1.13252
	21	86	1.11210		1.10107
	22	87	1.37129		1.44017
Waterslot	23		0		
UO_2	24	88	1.11680	1.05572	1.11958
	25	85	0.82333	0.77830	0.78827
	26	84	0.70638	0.66775	0.67417
	27	83	0.76210	0.72042	0.75188

Table E-17 - A Summary of the Calculated and Measured Power
in Configuration 12

CONFIGURATION 12

Core Size: 27x27, 0.56-inch pitch Analysis
 Inner Region: 19x19 PuO₂-UO₂ B_H²: 8.1242 x 10⁻⁴
 Outer Region: UO₂ Temp: 18°C
 Critical Water Height, cm: 104.76 k_{eff}: 1.010917
 Experiment Temp: 17.8°C (on period) 2-Group PDQ-3
 Aluminum Slab at Interface Westcott Cross Sections
 1453 ppm Boron Flat Side Extra Region Used for Slab

Type of Fuel	Rod Position	Rod Number	Experimental Relative Power		PDQ-3 Analysis (2-Groups)
			Foil Method	Thermal Method	
UO ₂	1	68	0.63120	0.59929	0.61336
	2	83	0.56787	0.53916	0.54352
	3	82	0.59800	0.56777	0.59543
	4	81	0.61987	0.58853	0.61490
PuO ₂ -UO ₂	5	98	0.80741		0.80027
	6				0.80284
	7				0.85880
	8				0.91339
	9				0.96076
	10	95	1.0		1.0
	11				
	12				
	13				
	14	90	1.0495		
	15				
	16	85	1.04519		
	17				1.02467
	18	84	0.99035		0.99176
	19	83	0.94830		0.95021
	20	82	0.90372		0.90206
	21	81	0.88772		0.86547
	22	75	0.96783		1.02943
Aluminum Plate	23	0	0		0
UO ₂	24	74	0.78328	0.74369	0.79401
	25	70	0.63142	0.59950	0.61740
	26	69	0.56526	0.53669	0.54041
	27	67	0.62399	0.59244	0.60523

Appendix F

APPENDIX F

WREC Critical Experiment Procedure and Results

1. Introduction

Reference 14 contains a complete description of the results obtained during the course of the WREC critical experiments program. However, sufficient experimental information is also included in this report to show how the information was used in confirming the nuclear design of Saxton Core II and to provide a basis for a comparison of the analysis with the WREC experiments. In addition, the measured reactivity values are based on an interpretation of the associated analysis. For this reason, it is desirable to include sufficient experimental data to permit a further evaluation as the methods of analysis are improved.

2. Experimental Procedure - Buckling

Buckling data were obtained by gamma scanning irradiated fuel rods. For the initial single-region UO_2 experiments the irradiated fuel rods were gamma counted at four positions along their length. In all subsequent experiments, the fuel rods were counted at six positions along their length. The data at each height from all the rods counted provided a radial power shape for radial buckling calculations. For axial buckling measurements, two irradiated rods in each core were gamma counted every two centimeters along their length.

The gamma counting included all gamma ray emission above one-half mev. The total counts obtained were corrected automatically for decay and wait time

by the rod scanning apparatus. This was accomplished by using a fuel rod from the same core in a second monitor channel. This monitor channel was programmed to stop the counting equipment after counting a certain number of preset counts from the monitored fuel rod. Since, this fuel rod decays at the same rate as those being scanned, the scanned rods yield total counts corrected for decay, wait time and count time. The total counts are then corrected for dead-time and background.

Axial and radial bucklings were calculated, from the data, by means of the DARED^{/19} code. The code normalizes and fits the data to either a cosine or Bessel function. DARED makes use of either a COFIT or JOFIT subroutine for this purpose. From the fitting, the code obtains an extrapolated power length. The extrapolated lengths were corrected by hand for dead-time with the formula shown below. The extrapolated lengths were then divided into π or J_0 to obtain the buckling.

$$H_C = H_D \left[1 - \frac{(\text{Peak Count/sec.})}{1000} \times 0.00025 \right]$$

where, H_C = extrapolated length corrected for dead-time

H_D = extrapolated length from DARED

3. Experimental Procedure and Results - Reactivity

a) Single-Region Configurations

During the initial single-region criticality experiments, $\delta\rho/\delta H$ measurements at the just critical moderator height were made using an on-line

analogue computer. The kinetic parameters used as input to the computer were those discussed in Section 5.6.2. A linear plot of moderator height vs. $\delta\rho/\delta H^{1/3}$ was made for each type of fuel, and differential moderator height curves were drawn for the active fuel region. One dimensional diffusion calculations, LEOPARD and AIM-5, were used to calculate $\delta\rho/\delta H$ in the reflector region. The resulting differential worth curves for each fuel are shown in Figure F-1. Subsequent $\delta\rho/\delta H$ data are also plotted on these curves. Total worth curves representing the integral of the differential worth are shown in Figure F-2. From these curves the fully inundated reactivity was determined for each fuel type using the calculated values of β_{eff} of 0.00796 for the UO_2 and 0.00338 for the $\text{PuO}_2\text{-UO}_2$. For various critical moderator heights, the reactivity was determined and subtracted from that available fully flooded. The resulting curve of installed reactivity available vs. just critical moderator height is shown in Figure F-3. The reactivity worth of a core change was determined from the difference in available reactivity corresponding to the critical water height for the unperturbed and perturbed cores. The installed reactivity curves show the available reactivity for the UO_2 and $\text{PuO}_2\text{-UO}_2$ cores is approximately the same for equal water heights.

The single-region measured reactivity results are summarized in Table F-1.

b) Multi-Region Configurations

For the multi-region configurations, $\delta\rho/\delta H$ was determined for each core from period measurements and reactivity vs. period curves. While each

Figure F-1 - Differential Moderator Worth Curves for the WREC
Single-Region UO_2 and PuO_2 - UO_2 Critical Experiments

Figure F-2 - Integral Moderator Worth Curves for the WREC
Single-Region UO_2 and $\text{PuO}_2\text{-UO}_2$ Critical Experiments

Table F-1

Reactivity Worths of Perturbations to Uniform Critical Lattices

Perturbation	5.74 w/o U-235			6.6 w/o PuO ₂			Boron*
	Aluminum Plate	Water Slot	Control Rods	Aluminum Plate	Water Slot	Control Rods	
Core Configuration	19x19	19x19	21x21	19x19	19x19	21x21	
Unperturbed Critical Moderator Height (cm)	81.94	81.94	51.0	80.75	80.75	50.38	81.34 (19x19) 50.38 (21x21)
Unperturbed Water Temperature (°C)	18.4	18.4	16.1	17.0	17.0	15.4	16.9 (19x19) 15.4 (21x21)
Perturbed Critical Moderator Height (cm)	85.73	78.33	87.38	81.36	74.22	77.34	87.77 (25 ppm, 19x19) 97.83 (50 ppm, 19x19) 70.78 (228 ppm, 21x21) 83.00 (309 ppm, 21x21) 88.06 (337 ppm, 21x21)
Perturbed Water Temperature (°C)	18.0	18.4	18.0	16.0	15.4	15.4	16.9 (19x19) 18.0 (21x21)
Reactivity Worth ($\Delta k/k$, %)	-0.27	+0.26	-4.50	-0.04	+0.49	-3.92	-0.39 (25 ppm, 19x19) -0.79 (50 ppm, 19x19) -3.38 (228 ppm, 21x21) -4.36 (309 ppm, 21x21) -4.64 (337 ppm, 21x21)

*Boron used only in those configurations where ppm values are listed

configuration has its own characteristic $\delta\rho/\delta H$ curve, the curve shown in Figure F-4 using data from a number of multi-region configurations was constructed and used in evaluating the reactivity change associated with a change in critical water height. The error resulting from this procedure was determined and found to be negligible.

The multi-region reactivity measurements are summarized in Table F-2.

Figure F-4 - Differential Moderator Worth and Installed Reactivity Above
Critical for the WREC Multi-Region Critical Experiments

Table F-2
Summary of Reactivity Measurements for the WREC
Multi-Region Critical Experiments

Type of Experiment	Core Size	Region 1		Region 2	Critical Height (cm)	Period		Perturbation	$\Delta k/k$	Remarks	Experiment Date
		Size	Fuel	Fuel		sec	Height (cm)				
Fuel Substitution	19x19	19x19	UO ₂		81.73			None	0	21°C, Reference Case	4/09/65
	19x19	3x3	PuO ₂ -UO ₂	UO ₂	82.17			3x3 PuO ₂ -UO ₂ Inserted in Center	-0.028	21°C	
	19x19	19x19	UO ₂		81.95			Detectors Removed From Core Boundaries	0	18°C, New Reference	4/12/65
	19x19	7x7	PuO ₂ -UO ₂	UO ₂	84.22			Insertion of 7x7 PuO ₂ -UO ₂ in Center	-0.13	18°C	
	19x19	19x19	PuO ₂ -UO ₂		80.73			None	0	14.7°C Reference Case	4/26/65
	19x19	3x3	UO ₂	PuO ₂ -UO ₂	79.95			3x3 UO ₂ Insert in the Center	+0.053		
	19x19	19x19	PuO ₂ -UO ₂		80.78			None	0	15°C, Reference Case	4/28/65
	19x19	17x17	PuO ₂ -UO ₂	UO ₂	91.31			Substitution, 1 Outer Row of UO ₂	+0.55	15°C	
	19x19	15x15	PuO ₂ -UO ₂	UO ₂	92.76			Substitution, 2 Outer Rows of UO ₂	-0.62	15°C	
	19x19	15x15	PuO ₂ -UO ₂	UO ₂	92.02			Repeat Last Exp. - 4/28/65	0	14.8°C, New Reference	4/29/65
	19x19	13x13	PuO ₂ -UO ₂	UO ₂	90.87			Substitution, 3 Outer Rows of UO ₂	+0.06		
	19x19	11x11	PuO ₂ -UO ₂	UO ₂	88.40			Substitution, 4 Outer Rows of UO ₂	+0.16		
	19x19	19x19	UO ₂		81.73			None	0	16°C, New Reference	5/05/65
	19x19	17x17	UO ₂	PuO ₂ -UO ₂	71.90			Substitution, 1 Outer Row PuO ₂	+0.71	16°C	
	19x19	15x15	UO ₂	PuO ₂ -UO ₂	71.67			Substitution, 2 Outer Rows PuO ₂	+0.75	16°C	
	19x19	13x13	UO ₂	PuO ₂ -UO ₂	72.95			Substitution, 3 Outer Rows PuO ₂	+0.62	15°C	
	19x19	11x11	UO ₂	PuO ₂ -UO ₂	74.45			Substitution, 4 Outer Rows PuO ₂	+0.51	15°C	
	19x19	7x7	UO ₂	PuO ₂ -UO ₂	77.55			Substitution, 6 Outer Rows PuO ₂	+0.27	15.5°C	
	19x19	19x19	PuO ₂ -UO ₂		80.78			All PuO ₂	+0.05	15°C	
Water Slot	21x21	11x11	PuO ₂ -UO ₂	UO ₂	51.38	52.0	51.90	None	0	Reference Core, T° = 15.5°C	5/04/65
	21x21	11x11	PuO ₂ -UO ₂	UO ₂	49.33	29.5	49.95	Water Slot in the Center	+0.46	T° = 15.5°C	
	21x21	11x11	PuO ₂ -UO ₂	UO ₂	50.11	43.0	50.65	Water Slot at PuO ₂ Interface	+0.27	T° = 15.5°C	
	21x21	11x11	PuO ₂ -UO ₂	UO ₂	50.35	30.0	50.95	Water Slot at UO ₂ Interface	+0.20	T° = 15.5°C	
Control Rods	21x21	11x11	PuO ₂ -UO ₂	UO ₂	51.38	52.0	51.9	None	0	Reference Core	5/04/65
	21x21	11x11	PuO ₂ -UO ₂	UO ₂	81.84	23.0	84.31	5 Control Rods in the Center	-3.65	T° = 15.5°C	
	21x21	11x11	PuO ₂ -UO ₂	UO ₂	73.88	28.6	75.39	5 Control Rods, 1 Row in From PuO ₂ -UO ₂ Interface	-3.10	T° = 15.5°C	
	21x21	11x11	PuO ₂ -UO ₂	UO ₂	71.86	37.0	73.09	5 Control Rods at PuO ₂ -UO ₂ Interface	-2.92	T° = 15.5°C	
	21x21	11x11	PuO ₂ -UO ₂	UO ₂	68.49	36.0	69.69	5 Control Rods at UO ₂ Interface	-2.60	T° = 15.8°C	5/05/65
	21x21	11x11	PuO ₂ -UO ₂	UO ₂	65.68	35.5	66.71	5 Control Rods, 1 Row Out From UO ₂ Interface	-2.25	T° = 15.8°C	
	21x21	11x11	PuO ₂ -UO ₂	UO ₂	62.90	52.5	63.63	5 Control Rods, 2 Rows Out From UO ₂ Interface	-1.92	T° = 15.8°C	
	21x21	11x11	PuO ₂ -UO ₂	UO ₂							
Water Slot Inverted Core	21x21	11x11	UO ₂	PuO ₂ -UO ₂	48.85	34.0	48.98	None	0	Reference Core, T° = 16.0°C	5/07/65
	21x21	11x11	UO ₂	PuO ₂ -UO ₂	47.62	41.4	48.05	Water Slot in the Center	+0.20	T° = 16.0°C	
	21x21	11x11	UO ₂	PuO ₂ -UO ₂	47.46	81.2	47.75	Water Slot at UO ₂ Interface	+0.28	T° = 16.0°C	
	21x21	11x11	UO ₂	PuO ₂ -UO ₂	47.49	49.4	47.87	Water Slot at PuO ₂ -UO ₂ Interface	+0.25	T° = 16.0°C	
Control Rods Inverted Core	21x21	11x11	UO ₂	PuO ₂ -UO ₂	48.85	34.0	48.98	None	0	Reference Core, T° = 16.0°C	5/07/65
	21x21	11x11	UO ₂	PuO ₂ -UO ₂	80.32	46.5	81.46	5 Control Rods in the Center	-4.35	T° = 16.0°C	
	21x21	11x11	UO ₂	PuO ₂ -UO ₂	62.0	36.7	62.84	5 Control Rods at UO ₂ Interface	-3.06	T° = 16.0°C	
	21x21	11x11	UO ₂	PuO ₂ -UO ₂	65.65	36.0	66.52	5 Control Rods at PuO ₂ -UO ₂ Interface	-2.55	T° = 16.0°C	
Boron Worth	27x27	19x19	PuO ₂ -UO ₂	UO ₂	65.46	74.0	65.97	Boron Content: 1166 ppm	0	Reference Core, T° = 18.4°C	5/12/65
	27x27	19x19	PuO ₂ -UO ₂	UO ₂	82.89	35.0	84.33	Boron Content: 1388 ppm	-1.49	T° = 18.5°C	5/13/65
	27x27	19x19	PuO ₂ -UO ₂	UO ₂	91.19	32.0	93.62	Boron Content: 1453 ppm	-1.95	T° = 18.5°C	5/13/65
Borated Perturbation	27x27	19x19	PuO ₂ -UO ₂	UO ₂	91.72	26.0	94.61	Boron Content: 1453 ppm	0	Reference Core, T° = 18.2°C	5/14/65
	27x27	19x19	PuO ₂ -UO ₂	UO ₂	98.19	87.0	102.36	Water Slot at PuO ₂ -UO ₂ Interface	-0.25	T° = 18.0°C	5/15/65
	27x27	19x19	PuO ₂ -UO ₂	UO ₂		140.0	104.76	Aluminum Slab at PuO ₂ -UO ₂ Interface	-0.42	T° = 17.8°C	5/17/65
Borated Insert	27x27	19x19	PuO ₂ -UO ₂	UO ₂	86.56	43.0	87.97	Boron Content: 1425 ppm	0	Reference Core, T° = 18.0°C	5/21/65
	27x27	19x19	PuO ₂ -UO ₂	UO ₂	87.85	27.0	89.77	3x3 UO ₂ Insert in Core Center	-0.07	T° = 18.5°C	

Appendix G

APPENDIX G

Saxton Core I

1. Introduction

The nuclear design of Saxton Core I is fully described in Reference 1. Information from this report and experimental data from the operation of the core provided a base point for the nuclear design calculations for the Saxton plutonium core. However, a number of changes had occurred in the methods of analysis since the completion of the original design. Consequently, a limited number of calculations were carried out to test the methods that were to be used in the design of the second core. The purpose of this section is to briefly summarize the comparisons of analysis with experiment that were made and to show how they influenced the design of Saxton Core II.

2. Saxton Core I Reactivity

The reactivity of Saxton Core I was determined for a number of different control rod configurations by means of a series of PDQ calculations. Table G-1 summarizes the results from data listed in Reference 1. The measured hot, clean, at full power reactivity available in Saxton Core I was 14.4% $\Delta k/k$. This value was extrapolated from boron worth and power coefficient measurements (1804 ppm boron = 15.4%, power defect = 1% $\Delta k/k$). The calculated value at the same conditions from Table G-1 is 13.5% $\Delta k/k$ or 14.5% $\Delta k/k$ depending on the method of defining installed reactivity.

TABLE G-1

Multiplication Factors for Saxton Core I for
Various Core Conditions (X-Y PDQ Analysis)

	k_{eff}	$\% \frac{\Delta k}{k_{Initial}}$	$\% \frac{\Delta k}{k} (\% \ln \frac{k_1}{k_2})$	k_{eff}	$\frac{\Delta k}{k_{Initial}}$	$\frac{\Delta k}{k} (\% \ln \frac{k_1}{k_2})$
No Control Rods	1.241	$\rho=19.4\%$	$\rho=21.5\%$	1.156	$\rho=13.5\%$	$\rho=14.5\%$
Fully Rodded	1.042	-16.0	-17.5	0.929	-19.6	-21.9
One Outer Rod Withdrawn	1.086	4.2	4.1	0.979	5.4	5.2
One Center Rod Withdrawn	1.108	6.3	6.1	1.005	8.1	7.9
Two Symmetric Outer Rods Withdrawn	1.100	5.5	5.4	0.997	7.2	7.1
Four Outer Rods Withdrawn	1.150	10.4	10.0	1.055	13.5	12.7
Two Center Rods Withdrawn	1.155	10.8	10.3	1.061	14.2	13.3
Three Rod Pattern Withdrawn	1.146	10.0	9.5	1.049	12.7	12.2
Four Outer + One Center Rod Withdrawn	1.200	15.2	14.1	1.109	19.2	17.7

During the Saxton plutonium design study, calculations were made for two operating conditions of Saxton Core I for which experimental information was available. Table G-2 summarizes the comparison of analysis with experiment for a number of different calculation methods.

A calculated k_{eff} of 0.9997 (PDQ-3) for the hot, clean, just-critical conditions containing 1804 ppm is in excellent agreement with the experiment. (This configuration is one in which the experimental reactivity can be accurately determined). The hot, full-power PDQ-3 calculated reactivity values, as defined in Table G-2, bracket the experimental result. In PDQ-3 calculations, the fuel element spacer grids were neglected. The spacer grids lie in four parallel planes perpendicular to the length of the fuel assembly. Based on these calculations, it was concluded that the PDQ-3 analysis was adequate and that the longitudinal grid structure could be neglected. To be consistent with the previous analysis, it was decided to quote the expected clean core installed reactivity in Saxton Core II on the basis of $\% \Delta k/k$ initial. (The available lifetime is the same irrespective of the definition used.)

3. Rod Worths

Figure G-1 compares the analytic and experimental differential control rod worth and boron concentration required for criticality as a function of the total rod bank insertion length. The integral of the calculated differential control rod worth curve is approximately 21.1% $\Delta k/k$. The measured bank worth was $\approx 19\% \Delta k/k$ or approximately

TABLE G-2

Analysis Experiment Reactivity
Comparison for Saxton Core I

	<u>Calculated Value</u>		<u>Experimental Value</u>
	<u>% $\Delta k/k_{\text{Initial}}$</u>	<u>% $\Delta k/k$ (% $\ln \frac{1}{k}$)</u>	<u>% $\Delta \rho$ ⁴</u>
¹ Original Analysis - hot, full power	13.5	14.5	14.4
² PDQ (Sax Pu Study) hot, full power	13.8	14.8	14.4
² PDQ (Sax Pu Study) 1804 ppm, hot, full power	0.9997 ³	0.9997 ³	$k_{\text{eff}} = 1.0$
¹ LEOPARD (Sax Pu Study) hot, full power	13.6	14.6	14.4

1 Contains homogenized spacer grid structure

2 Does not contain grid structure

3 Calculated k_{eff}

$$4 \quad \Delta \rho = \int_{C_{B1}}^{C_{B2}} \frac{\partial \rho}{\partial C_B} \cdot dC_B$$

Figure G-1: Saxton Core I Differential Control Rod Worth and Boron Requirement-All Rods Banked

10% less than that calculated from the differential rod worth curve. Figure G-1 also shows that approximately 2000 ppm boron was expected to be required for the hot, clean, zero-power just critical condition based on results from a one-dimensional diffusion calculation. The actual requirement was 1804 ppm.

These comparisons show that the methods of analysis available at that time result in an overprediction of rod worth. However, defining rod worth as $\% \Delta k/k_{\text{initial}}$ reduced the calculated value to about the value measured. This procedure was followed in quoting rod worths for the Saxton plutonium core until additional comparisons of analysis with control rod experiments were available from the WREC critical experiments.

Figures G-2 through G-5 contain the same type of information for different rod configurations and are included to permit a later comparison with Saxton Core II measurements.

4. Lifetime

Lifetime calculations for Saxton Core I were made using the same methods to be used in the design of the plutonium loading. Figure G-6 shows the installed reactivity as a function of hours operation at 23.5 MWt as determined by means of a LUX one-dimensional burnup calculation. Based on an extrapolation of experimental data, the projected lifetime capability of Saxton Core I was $\approx 11,400$ MWD or 11,650 hours at a power level of 23.5 MWt. Thus, the lifetime capability of the core was ≈ 1000 hours longer than the analysis would indicate. However, there are small differences in the fission product cross sections resulting from

Figure G-2: Saxton Core I Differential Control Rod Worth and Boron Requirement-Rods 2 and 5 Banked, Rods 1,3,4,6 In

Figure G-3: Saxton Core I Differential Control Rod Worth and Boron Requirement-Rods 1,3,4,6 Banked, Rods 2 and 5 In

Figure G-4: Saxton Core I Differential Control Rod Worth and Boron Requirement-
Rods 3 and 4 Banked, Rods 1 and 6 Out, Rods 2 and 5 In

Figure G-5: Saxton Core I Differential Control Rod Worth and Boron Requirement-Rods 1 and 6 Banked, Rods 2,3,4,5 In

Figure G-6: Calculated Lifetime for Saxton Core I

fissioning in plutonium and uranium fuels that are not treated correctly in the analysis. In addition, the initial comparisons of analysis and experiment for a number of Hanford mixed-oxide experiments tended to indicate that the number of absorptions per fission in plutonium was actually higher than that of the available cross section set. Consequently, the overprediction in Core I lifetime was not used as a basis for increasing the calculated lifetime expected for Saxton Core II. However, the lifetime prediction is believed to be conservative.

5. Reactivity Coefficients Data

Experimental data concerning Saxton Core I Doppler, pressure, and temperature coefficients were used in the interpretation of the calculations of the same quantities for the plutonium core. Measured values are included in the summary of the various calculations. The experimental data is included in Figures G-7 through G-10.

6. Calculated and Measured Power Peaking

Figure G-11 shows the PDQ-3 calculated power distribution for Saxton Core I. As shown in this figure, the maximum calculated peak-to-average radial power is 1.91. A value of 2.03 was inferred from experimental (flux wire) data for comparable conditions. However, the experimental value includes a 10% safety factor to allow for a possible error in translating the flux wire measurements to local rod power. Throughout the design calculations for Saxton Core II, the

Figure G-9: A Comparison of the Calculated and Measured Moderator Temperature Coefficients for Saxton Core I

Figure G-10: Saxton Core I Pressure Coefficient at Different Boron Concentrations

PDQ Power Distribution: Case 1

Configuration: Normal Saxton core, 1000 ppm Boron
Eigenvalue: 1.067

NOTES: The underlined value in each assembly is the average power in that assembly relative to the average power in the core. The relative power is also shown for individual fuel rods near the uncontrolled corners of assemblies (where hot spots usually occur). If the hot rod is elsewhere, the value and location are shown.

Figure G-11: PDQ-3 Radial Power Distribution for Saxton Core I

maximum local radial power value was increased by the ratio of 2.03 to 1.91 to account for this possible error.

7. Saxton Core I Design Data and Performance Characteristics

A summary of Saxton Core I design data and performance characteristics are included in tables G-3 and G-4. The information in these tables was taken directly from Reference 1 and is included for comparison with similar information for Saxton Core II included in Appendix A.

TABLE G-3

Saxton Core I Design DataCore Description

Average core diameter (cold)	28.07 inches
Active core length (cold)	36.6 inches
Number of fuel assemblies	21
UO ₂ in the core	2230 lbs ^{**}
Uranium enrichment	5.69

Fuel Design Data (cold dimensions)

Pellet diameter	.357 inch
Pellet length	0.732 inch
Average UO ₂ density	10.18 gms/cm ³
Fuel tube O.D.	.391 inch
Fuel tube I.D.	.361 inch
Fuel tube center-to-center pitch	.580 inch
Total number of fuel rods	1676 [*]

* The total number of fuel rods is made up as follows:

21 assemblies at 72 rods each:	1512
6 followers at 18 rods each:	108
9 inserts at 9 rods each:	81

deducting

23 instrumentation thimbles	- 23
2 secondary source tubes	- 2

1676

** Measured weight of UO₂ = 2256 lbs.

Control Rod Data

Number of moveable control rods	6
Control rod material	80% Ag, 15% In, 5% Cd
Control rod shape	Offset Cruciform
Control rod span (cold)	5.605 inches
Control rod thickness (cold)	.390 inch

Thermal and Hydraulic Design Data

Coolant flow, total rate, lb/hr	2.8 x 10 ⁶
Operating pressure, normal, psi	2000
Pressure drop across core, psi	6.5
Average heat flux, Btu/hr-ft ²	137,000
"Active" heat transfer surface, ft ²	498
Temperature, °F	
Coolant in the main coolant system, average	530
Coolant rise in core, average	22
Film drop, average	53.5
Thermal power level, MW	20
Mean linear coefficients of thermal expansion, inch per inch per °F	
Fuel, UO ₂ pellets, zero power 530 F	4.0 x 10 ⁻⁶
full power 1400 F	4.85 x 10 ⁻⁶
Clad, SS, zero power 530 F	9.44 x 10 ⁻⁶
full power 595 F	9.54 x 10 ⁻⁶
Structure, SS	9.44 x 10 ⁻⁶

TABLE G-4

Nuclear Design Data Sheet for Saxton Core IA. Structural Characteristics

1.	Core Material at Power (Core plus Followers)	Density (lb/in ³)	Volume (in ³)	Atomic** Density	Weight (lbs)
	Fuel (UO ₂)	0.361	6,060	0.00598	2230
	Water	0.028	14,900	0.0166	413
	Stainless Steel (304)	0.285	1,570	0.00594	448
	(348)	0.285	103	0.00039	29
	Zirconium	--	--	--	--
	TOTAL		22,633		3120
1.1	Core Diameter, in.				28.1
	Core Height, in.				36.6
	Reflector Thickness and Composition				
	Top	SS + H ₂ O		c. 10 in.	
	Bottom	SS + H ₂ O		c. 10 in.	
	Side	SS + H ₂ O		c. 10 in.	
2.	H ₂ O/U - Cold				5.04
	At power*				3.87
3.	Number of Fuel Assemblies				21
4.	UO ₂ Rods Per Assembly				71***

* Volume ratio if $\rho_{H_2O} = 1$ and $\rho_U = 18.7$

** (No./barns - cm)

*** Total rods per assembly = 72, less 1 instrumentation thimble.

B. Performance Characteristics

1.	Heat output, MW	20
2.	Loading technique	Batch
3.	Full power lifetime, hrs.	7800
4.	Average discharge burnup, MWD/MTU	7290
5.	Initial weight of U per charge, kg	891
6.	Discharge weight of U per charge, kg	882
7.	U-235 feed enrichment, w/o	5.7
8.	U-235 discharge enrichment, w/o	4.6
9.	Pu produced per charge, kg	2.2
	% Pu-240	7
	% Pu-241	2
10.	Initial conversion ratio, clean fuel	0.3
11.	Fractional energy production	
	U-235	0.89
	U-238	0.05
	Pu	0.06

C. Control Characteristics

1.	Effective Multiplication (beginning of life)	
	Hot, power, equilibrium poison	1.127
	Hot, power, unpoisoned	1.156
	Hot, no power, unpoisoned	1.171
	Cold, no power, unpoisoned	1.241
2.	Number of potential control rod locations	6
3.	Control rod span, in.	5.605

4.	Control rod thickness, in.	0.3925
5.	Control rod material	Ag-In-Cd
6.	Number of control rods specified	6
7.	Estimates control rod worth (average per rod)	3.8%
8.	Boron concentrations, ppm, natural boron	
a.	To shut reactor core down to $k = 0.97$ with no rods inserted.	
	Cold	2200
	Hot	2330
b.	To control at power with no rods inserted	
	Unpoisoned	1835
	Poisoned (Xe and Sm)	1495
c.	Approximate concentration for worth of one (1) per cent	
	Hot	110
	Cold	80

D. Nuclear Characteristics at Power, Clean Core

1.	Neutrons produced per thermal absorption (ηf)	1.669
2.	Resonance escape probability (p)	0.620
3.	Non-thermal fission factor ($\beta \epsilon$)	1.359
4.	Infinite multiplication factor ($\eta f p \beta \epsilon$)	1.406
5.	Neutron age (τ), cm^2	50.5
6.	Thermal diffusion area (L^2), cm^2	2.7
7.	Geometric buckling (B^2), cm^{-2}	0.00405
8.	Average flux levels, neutrons/ $\text{cm}^2\text{-sec}$	
	Thermal	1.5×10^{13}
	Fast	1.0×10^{14}

E. <u>Neutron Balance</u> (including control)	<u>Fraction</u>
Fast Capture	0.243
Fast Leakage	0.172
Epithermal Fission	0.112
Thermal Capture	0.177
Thermal Fission	0.291
Thermal Leakage	0.005
TOTAL	1.000

F. Kinetic Characteristics (rodded core)

1. Temperature Coefficients

Fuel ($^{\circ}\text{F}^{-1}$)	$-1.6 \times 10^{-5} \leq \alpha \leq -0.4 \times 10^{-5}$
Coolant	
Moderator ($^{\circ}\text{F}^{-1}$)	$-5.3 \times 10^{-4} \leq \alpha \leq -2.0 \times 10^{-4}$
Pressure (psig) $^{-1}$	$+1.9 \times 10^{-6} \leq \alpha \leq +5.1 \times 10^{-6}$
Void	$-0.39 \leq \alpha \leq -0.14\% \text{ k}/\% \text{ Void}$

2. Delayed Neutron Fraction

0.63%

3. Prompt Neutron Lifetime

1.6×10^{-5} sec.