WCAP-3385-1

SAXTON PLUTONIUM PROGRAM

QUARTERLY PROGRESS REPORT

FOR THE PERIOD

ENDING SEPTEMBER 30, 1964

Westinghouse Atomic Power Division

WCAP-3385-1

(Category UC-80)
(Reactor Technology)
(Joint US-Euratom Program)

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N. R. Nelson
Project Manager

Prepared for the New York Operations Office
U. S. Atomic Energy Commission
Under AEC Contract AT(30-1)-3385

October 1964

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SUMMARY

This is the first quarterly report issued and covers work accomplished on a preliminary basis prior to formal contract signing on June 30, 1964 as well as work done during the first quarter of fiscal 1965.

Objectives

The purpose of the Saxton Plutonium Project is to develop information concerning the utilization of plutonium enriched fuel in pressurized water reactor systems, through design, fabrication and operation of a partial core of PuO_2-UO_2 fuel in the Saxton Reactor. In-pile performance of this fuel will be evaluated and post-irradiation examination of fuel samples will be made. Performance will be analyzed and compared with analytical predictions.

Prior to loading, critical experiments will be conducted to evaluate nuclear design methods and to predict reactivity and power distribution of the PuO_2-UO_2 fuel rods in the Saxton core.

Prior to startup, a series of zero and low power physics measurements will be made to verify reactivity, control rod and boron worth, flux and power distribution, and temperature, pressure and power coefficients. These characteristics, as well as core thermal and hydraulic performance, will also be measured at intermediate and full power levels. Similar data will be obtained during a minimum of four reactor shutdowns to determine characteristic changes with irradiation.

Schedule

Operation of the fuel in the Saxton Reactor is scheduled from July 1965 to July 1967 and post-irradiation examination and analysis from January 1968 to February 1969. A copy of a PERT-type summary project schedule is included at the end of this report.

Status

Currently, portions of the project are running about four weeks behind schedule. These delays, however, can be recovered and the key date of July 1, 1965 for loading fuel into the Saxton Reactor remains firm.

No topical reports have been issued to date.

SAX-100 Project Administration

N. R. Nelson

Prior to official signing of Contract AT(30-1)-3385 on June 30, 1964, preliminary work and planning authorized by the AEC was undertaken. This work included literature search reviews on plutonium cross sections, on materials behavior, on plutonium critical experiment data and on hazards analyses. In addition the Saxton core design was analyzed to determine the number and location of UO_2 fuel assemblies which could be replaced by PuO_2-UO_2 assemblies during the next Saxton refueling.

After contract signing, emphasis was placed on expediting designs, specifications and invitations to bid on the PuO_2-UO_2 fuel rods. Of six likely vendors contacted, only NUMEC and Hanford agreed to quote. The other four indicated long range interest but were not yet ready to undertake a fabrication job of this size.

An analysis of fuel fabrication methods which would be likely to survive as economic competitors in future plutonium fuel factories indicated that both automated vibratory compaction and automated pelletization showed potential. Consequently, it was decided that fuel made by both processes needed to be tested and evaluated.

An investigation was made of the reliability of vibratory compacted fuel and discussions were held with the Saxton Nuclear Experimental Corporation (SNEC) to verify that both types of fuel would be acceptable in the second Saxton core.

Firm price bids from NUMEC and cost estimates from Hanford were received for fabrication of several package combinations of pelletized and vibratory compacted fuel rods. Reviews were made to evaluate prices, schedules and exceptions taken to the preliminary drawings and specifications. Recommendations for contract awards were made to the Commission. Authority was received from the Commission to proceed, pending satisfactory resolution of the final contact negotiations and prices. This action is expected to be completed early in October.

A PERT-type summary project schedule and work programs for all planned subtasks were prepared and submitted to the AEC. This planning work highlighted the extremely tight timing necessary to meet the mid-summer 1965 date for loading the plutonium fuel into the Saxton Reactor. In spite of temporary delays, this date still appears realistic.

A request for license modifications was submitted to the Division of Reactor Licensing to permit receipt and use of PuO_2-UO_2 fuel rods at the Westinghouse Evaluation Center (WREC) for confirmatory critical experiments. Subsequent meetings with the Division have been held. Work on the safeguards analysis for use of PuO_2-UO_2 fuel in Saxton is underway on an accelerated basis.

Plutonium for use on this project will be obtained from inventory at Hanford. Release approval by the AEC has been obtained. Since the

isotopic concentration of this fuel varies from the concentration utilized in the preliminary calculations, final adjustments are now being made in the exact amount of PuO_2 to be mixed with natural UO_2 and in other related design parameters.

Negotiations with the AEC are underway to determine a method for calculating and verifying the extra costs to Westinghouse of supplying PuO_2-UO_2 fuel instead of UO_2 fuel in nine assemblies of the second Saxton core.

SAX-210 Nuclear Fuel Design

F. L. Langford, W. L. Orr, A. J. Impink,
R. H. Chastain, H. A. Risti, H. I. Sternberg

A. Introduction and Scope of Work

1. Introduction

The objective of the nuclear design task is to develop the nuclear specifications for the partial loading of the mixed-oxide (PuO_2-UO_2) fuel assemblies to be used in the Saxton reactor during a two-year irradiation period. The basic requirement is to determine a plutonium loading that will provide the desired lifetime within the power limitations set by the plant, thermal, and hydraulic design. In addition, it is desirable to achieve as high a burnup in the plutonium fuel as possible.

2. Scope of Work

During the quarter, studies were carried out to establish the nuclear configuration that most nearly meets the objectives of the program. The following statements briefly describe the scope and results of these studies.

a. A comparison was made of the relative advantages of Zircaloy and stainless steel clad. Since Zircaloy provides a higher initial reactivity and a longer lifetime than stainless steel for the same plutonium enrichment, it was selected as the fuel clad material.

b. Parameter studies were carried out to evaluate the effect of plutonium enrichment and location on core lifetime, initial multiplication, power peaking and power sharing. As a result, a reference configuration was established in which nine plutonium fuel assemblies are installed in the center of the core with twelve conventional uranium fuel assemblies installed on the periphery. A second purpose of the parameter studies was to narrow the range of possible plutonium enrichments to permit a more detailed investigation of burnup and power distribution effects.

c. One-dimensional and two-dimensional burnup calculations were carried out to set the plutonium enrichment specifications. These studies show that a plutonium enrichment that provides sufficient lifetime when installed in the center of the core results in an increase in the radial nuclear hot channel factor over that of a 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 fuel. Thus the enrichment specification requires a compromise between irradiation, power peaking, and lifetime objectives.

d. Kinetic characteristics, reactivity coefficients, and controls and boron worth were investigated. The following qualitative statements briefly summarize some of the reactivity and kinetic effects that occur with nine plutonium fuel assemblies installed in the center of the core.

(1) The Saxton-Plutonium core has a more negative moderator temperature coefficient than with a conventional uranium loading.

(2) The negative Doppler coefficient is larger with PuO_2-UO_2 fuel.

(3) The part-plutonium core has a larger positive pressure coefficient and a more negative void coefficient than the conventional uranium core.

(4) Identical boron concentrations and control rods are worth less in the core containing plutonium.

(5) The β_{eff} and prompt neutron lifetime at the beginning-of-life is less with a partial plutonium loading than with a full loading of uranium fuel and remains constant throughout life. (The kinetic characteristics of the plutonium configuration are similar to that of a uranium power reactor at the end-of-life).

B. General Descriptions

1. Core Configuration

The plutonium core configuration used as a basis for the analysis consists of nine plutonium fuel assemblies installed in the center of the core with twelve uranium fuel assemblies installed on the periphery. Seven of the nine plutonium fuel assemblies are assumed to contain pelletized fuel while two contain vibratory compacted (VIPAC) fuel. The L-sections and fuel followers are partially burned in that they were also used in the first core loading.

2. Fuel Elements

The specifications for the fuel rods and fuel used in the analysis for each region are compared in the following list.

Fuel Rods

	Plutonium	Uranium
Clad Material	Zircaloy	304 SS
Clad Thickness, inches	0.023	0.015
Clad O.D., inches	0.391	0.391
Diametral Gap, inches	0.005 - pelletized 0.0 VIPAC	0.004

Fuel

Material	6 PuO_2-94 UO_2	6 PuO_2-94 UO_2	UO_2
Type	Pelletized	VIPAC	Pelletized
Fuel Diameter, inches	0.340	0.345	0.357
Density, % Theoretical	94	88	93
Fuel Enrichment	6 w/o Pu	6 w/o Pu	5.7 w/o U-235

The following isotopic content of plutonium was used:

		%
Pu-239	=	90.46
Pu-240	=	8.60
Pu-241	=	0.90
Pu-242	=	0.04

The plutonium isotopic content of the fuel available is expected to be approximately the same as that used in the analysis. However, other design parameters such as fuel column length or nuclear hot channel factors may require a small variation in the enrichment. The final enrichment specification will be included in the next quarterly report.

3. Description of Analysis Methods

The analysis to establish the nuclear characteristics of the plutonium loading in the Saxton reactor has consisted of the use of standard WAPD analytic methods that have been verified in the design and in the operational analysis of the Saxton reactor and other pressurized water reactors. In addition these methods have been compared to six mixed-oxide (PuO_2-UO_2) critical and approach-to-critical experiments with variable lattice pitches conducted at Hanford. Thus, the extrapolation to a plutonium configuration is based in so far as possible on available experimental information.

The following paragraphs contain a brief description of the computer programs used in the analysis.

LEOPARD[1]

Determines fast and thermal spectra based on a modified MUFT - SOFOCATE model. The code also optionally computes fuel depletion effects for a dimensionless reactor. An additional description of the program is included in description of the SAX-250 task.

---

[1] R. F. Barry, "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094", WCAP-3741 (1963)

TURBO*[1]

Solves the few-group, two-dimensional (X-Y geometry) neutron diffusion equations in combination with a point-wise 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*.

PDQ-3[2]

Solves the few-group, time independent, neutron diffusion equations in X-Y geometry.

AIM-5[3]

Solves the few-group, time independent, neutron diffusion equation in one dimension.

---

[1] S. M. Hendley, R. A. Mangan, "TURBO* - A Two-Dimensional Few-Group Depletion Code for the IBM-7090, WCAP-6059 (1964).

[2] W. R. Cadwell, et. al., "PDQ-3, A Program for the Solution of the Neutron Diffusion Equations in Two-Dimension on the IBM-704", WAPD-TM-179 (May 1960).

[3] H. P. Flatt and D. C. Baller, "AIM-5 - A Multi-Group, One-Dimensional Diffusion Equation Code", NAA-SR-4694 (March 1960)

LUX

LUX is a modification of the CANDLE[1] one-dimensional few-group depletion code to determine reactivity as a function of lifetime. The LEOPARD microscopic library is used.

THERMOS[2]

THERMOS is a cell transport theory code in both space and energy. This program was used as a check on the thermal group calculation contained in LEOPARD. The results from the two calculations were in excellent agreement.

---

[1] O. J. Marlowe, P. A. Ombrellaro, "CANDLE - A One-Dimensional Few-Group Depletion Code for the IBM-704", WAPD-TM-53 (1957).

[2] H. Honeck, "THERMOS, A Thermalization Transport Theory Code for Reactor Lattice Calculations", BNL-5826 (1961)

C. Analytic Results

1. Available Reactivity

The hot, clean reactivity of the reference design at power is expected to be 9.8 ± 1.3% Δk/k as shown in Table 1. The listed range in the available reactivity value is based on estimates as to the magnitude of possible error for each item in the table.

Table 1

Hot, Clean Reactivity for the Reference Design

		% Δk/k
1.	Calculated reactivity, AIM-5 analysis	13.16
2.	Depleted followers and L-sections	-1.06 ± 0.4
3.	Allowance for analysis-to-experiment discrepancy	-2.00 ± 1.0
4.	Increased discrepancy - Higher Pu loading than in available experiments	-0.00 ± 0.7
5.	Power effect - depleted followers	-0.10 ± 0.1
6.	Possible error in temperature extrapolation	0.20 ± 0.2
	Available Reactivity	9.8 ± 1.3

The following paragraph explains each item listed in Table 1.

1. The calculated reactivity of the system was determined by means of a LEOPARD - AIM-5 calculation. As a check on the method, the regular Saxton core composition containing the measured boron concentration (1804 ppm boron) for the hot, clean, rods-out critical condition was used in a similar calculation. A k_{eff} of 1.005 was determined as compared to the experimental k_{eff} = 1.0. Consequently the analysis and experiment are in good agreement when applied to the Saxton reactor for UO_2 fuel.

2. The one-dimensional AIM-5 calculation does not include the presence of partially burned L-sections and followers. This effect was evaluated in a PDQ-3 calculation in X-Y geometry where the followers and L-sections were entered explicitly. (The one-dimensional and two-dimensional calculations are in good agreement when corrected for minor differences in composition).

3. A comparison of analysis with experiment for six mixed oxide (1.5 PuO_2-UO_2) critical and approach-to-critical experiments performed at Hanford results in a mean discrepancy of 2.6% $\Delta k/k$. This comparison is described more fully in the SAX-250 summary. Installation of the plutonium in only part of the core should reduce the discrepancy since the uranium region reactivity can be accurately determined. A weighting factor for the

plutonium region of ≈ 0.7 was determined from calculations of the reactivity effect for changes in each region separately. The listed discrepancy corresponds to an equivalent full core error of 3% $\Delta k/k$, the maximum error found in the analysis of the Hanford experiments.

4. The Hanford experiments were carried out for a lower plutonium concentration (1.5 w/o PuO_2) than the enrichment proposed for the Saxton reactor (6 w/o PuO_2). Previous calculations for more heavily loaded Pu-Al fueled experiments do not indicate that the reactivity discrepancy is dependent on the loading. They do tend to indicate that the size of the error is reduced in the tighter lattices when the reference cross-section set is used (Leonard cross-sections). No discrepancy for this item is listed in the table. However, a reactivity uncertainty is included because no mixed oxide experiments near the proposed loading are available for a more realistic evaluation. The uncertainty value corresponds to a full core error of about 1% $\Delta k/k$.

5. In addition to the reduction in reactivity resulting from the use of partially burned fuel followers and L-sections, a second order effect on the reactivity-lifetime relationship occurs. The power output from the fuel followers and L-section will be reduced from that indicated by the analysis. Consequently, for the same total power output, the power density in the rest of the core will be slightly increased. The total

change in lifetime is about 100 hours. This effect has been translated into an initial reactivity loss for convenience.

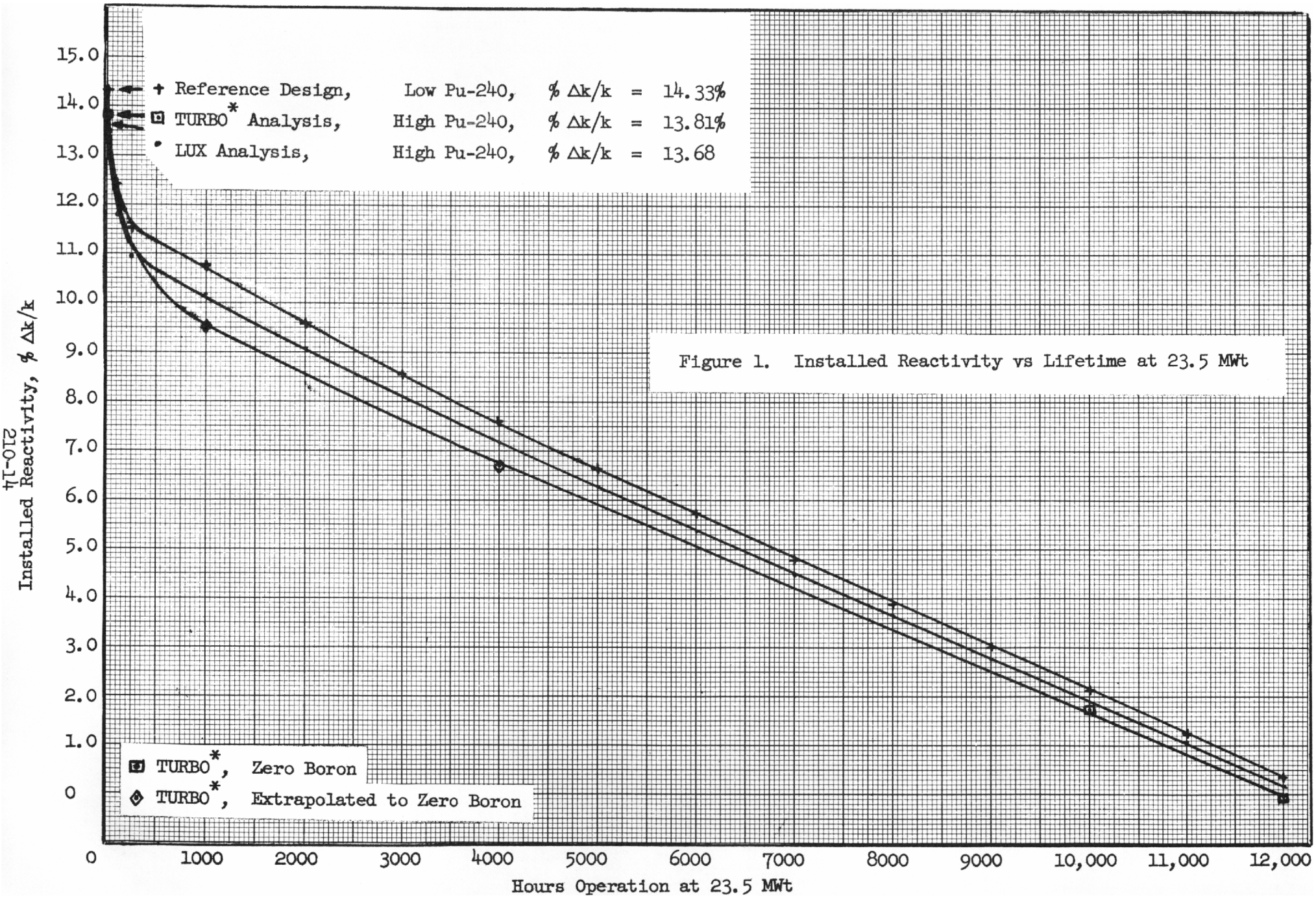
6. The Doppler broadening of the Pu-240 resonances is based on an L factor determined from a U-238 correlation as described in the SAX-250 summary. Initial calculations indicate this is a good approximation at the design temperature. However, differences in thermal conductivity for the VIPAC fuel in two of the nine assemblies has not been included in the analysis. It is felt that these two considerations will tend to reduce reactivity.

An examination of Table 1 shows that the significant items contributing to the reactivity uncertainty are the allowances for possible discrepancies between analysis and experiment indicated by a comparison of the more lightly-loaded Hanford experiments. Consequently, reactivity measurements in the WREC critical experiment with the design mixed oxide fuel rods are necessary before the available reactivity can be established with confidence. The large range in reactivity values that could occur leads to the conclusion that the planned critical experiments are essential for the successful execution of the program.

2. Reactivity - Lifetime Relationship

Reactivity as a function of operating time at 23.5 MWt for two different plutonium compositions is shown in Figure 1 as determined from a TURBO* calculation in X-Y geometry and from two one-dimensional LUX calculations. This figure shows that there is a small difference in lifetime in the one-dimensional and two-dimensional analysis. The one-dimensional analysis indicates a 400-hour longer available life than the TURBO* calculation for the same reactivity change after 6000 hours operation and a 200-hour longer available life after 12,000 hours.

The TURBO* lifetime calculation sequence assumed chemical shim operation. In the first step, sufficient boron was added to reduce the 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. At those points in the analysis where the reactivity was evaluated both before and after boron dilution, the worth of boron was calculated and a clean core reactivity was determined. 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 slight difference in the slope of the reactivity-lifetime curves for the two analysis methods. Since the


+ Reference Design, Low Pu-240, % Δk/k = 14.33%
TURBO* Analysis, High Pu-240, % Δk/k = 13.81%
• LUX Analysis, High Pu-240, % Δk/k = 13.68
Figure 1. Installed Reactivity vs Lifetime at 23.5 MWt
TURBO*, Zero Boron
TURBO*, Extrapolated to Zero Boron
Installed Reactivity, % Δk/k
Hours Operation at 23.5 MWt
15.0
14.0
13.0
12.0
11.0
10.0
9.0
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0
1000
2000
3000
4000
5000
6000
7000
8000
9000
10,000
11,000
12,000

difference in the two methods is small, the lifetime prediction for the reference design is based on the LUX one-dimensional analysis. A comparison of the one-dimensional lifetime calculations carried out during the quarter for the uranium fueled Saxton reactor using current design methods with the projected lifetime from operations data shows that the available lifetime is approximately 9% larger (equivalent to ≈ 1000 hours at 23.5 MWt) than was calculated. However, no allowance for a possible underestimate of the lifetime that may be introduced by the analytic method has been included in the lifetime predictions for the core containing plutonium.

Figure 1 also includes a comparison of the effect on reactivity and lifetime for a change in the relative isotopic content of the plutonium for a constant PuO_2 weight fraction. The isotopic content of the two fuels used in the analysis is listed in Table 2.

Table 2

Plutonium Isotopic Composition

Isotope	Reference Composition, % (Low Pu-240)	High Pu-240, %
Pu-239	90.46	88.2
Pu-240	8.60	10.2
Pu-241	0.90	1.6
Pu-242	0.04	-

The initial reactivity for the reference composition of Figure 1 is higher than that of the comparable composition used in the AIM-5 analysis of Table 1. This difference is due to a small difference in the amount of structure included.

Figure 2 summarizes the initial hot, clean reactivity at power that is needed to achieve a specified lifetime for each of the three conditions. This figure shows that if the nominal value of the available reactivity listed in Table 1, 9.8% $\Delta k/k$, is realized, the available operating life at 23.5 MWt is expected to be 7300 hours. For the reactivity range listed, $\pm$ 1.3% $\Delta k/k$, the corresponding available lifetime is 6000 hours minimum to 8700 hours maximum. These results illustrate the importance of reducing the reactivity uncertainties by measurements with fuel rods of the proper loading in the critical experiment. Even though it is believed that the analysis provides a conservative estimate as to the lifetime available, at the present time the feasibility of obtaining a lifetime of 8400 hours at 23.5 MWt (8250 MWD) is marginal. Should the experiment show that the available reactivity is lower than that needed to provide the desired lifetime, an alternate procedure such as redistribution of the fuel will be required. An increase in plutonium enrichment is not desirable because of the resulting increase in radial nuclear hot channel factors.

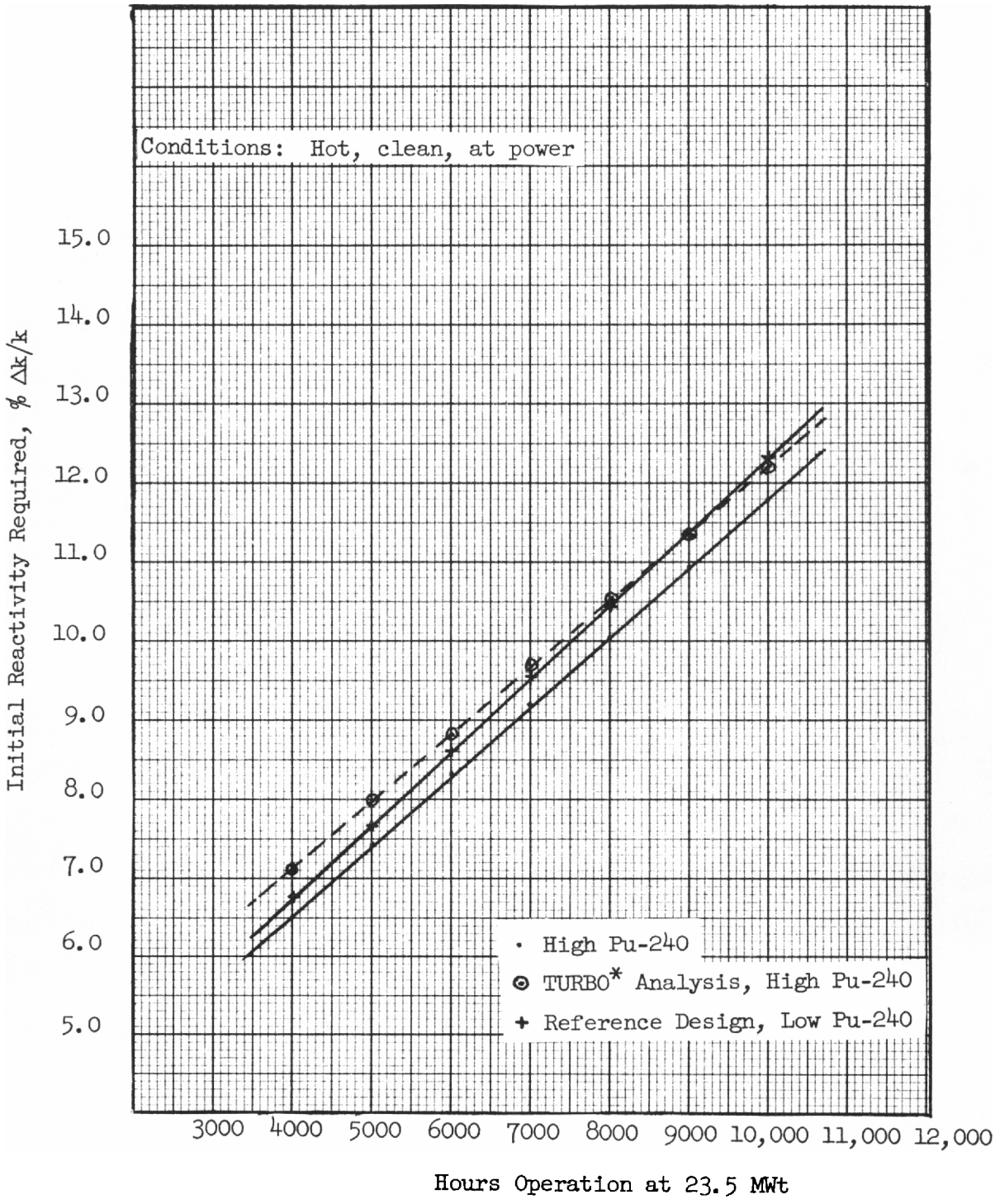


Figure 2. Initial Reactivity Requirement versus Available Lifetime at 23.5 MWt

3. Power Distribution Summary

The peak power in the core occurs in the plutonium region at the beginning-of-life. Figures 3 and 4 show the core power distribution at the beginning-of-life and after 6000 hours operation. The peak power in each region is also shown. These power distributions were determined from TURBO* calculations.

The following list summarizes the maximum value of peak rod power to core average power for each region as a function of operating time at 23.5 MWt.

Peak Rod/Core Average Power

Time, hours	Pu Region	U Region
0	2.25	1.22
6,000	1.80	1.26
12,000	1.56	1.23
Average	1.85	1.24

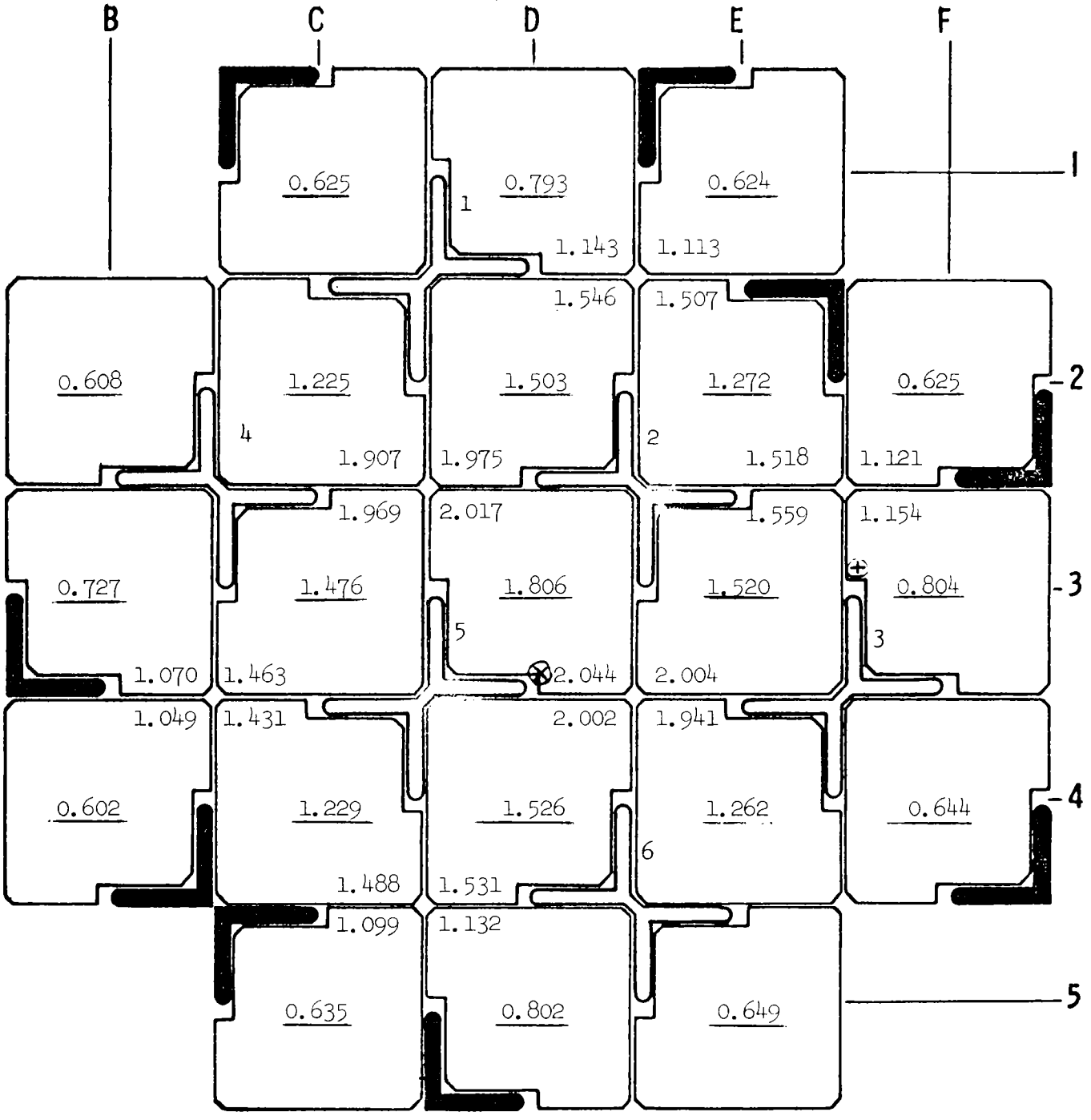
A design lifetime objective of 8250 MWD results in a core average burnup of 9600 MWD/tonne. Using the listed average value for the peak rod power in each region results in the following predicted values for maximum burnup under chem shim operation.

Maximum burnup, Pu fuel rod = 18,700 MWD/tonne

Maximum burnup, U fuel rod = 11,500 MWD/tonne

Conditions: 2400 ppm Boron, All rods out

Note: The underlined value in each assembly is the average power in the assembly relative to the average value in the core. The relative power for individual fuel rods is not underlined.



⊗ Maximum Power in a Plutonium Rod = 2.25

⊕ Maximum Power in a Uranium Rod = 1.22

Figure 3. TURBO* Power Distribution at the Beginning-of-Life

Conditions: 800 ppm Boron, All rods out.

Note: The underlined value in each assembly is the average power in the assembly relative to the average value in the core. The relative value for individual fuel rods is not underlined.

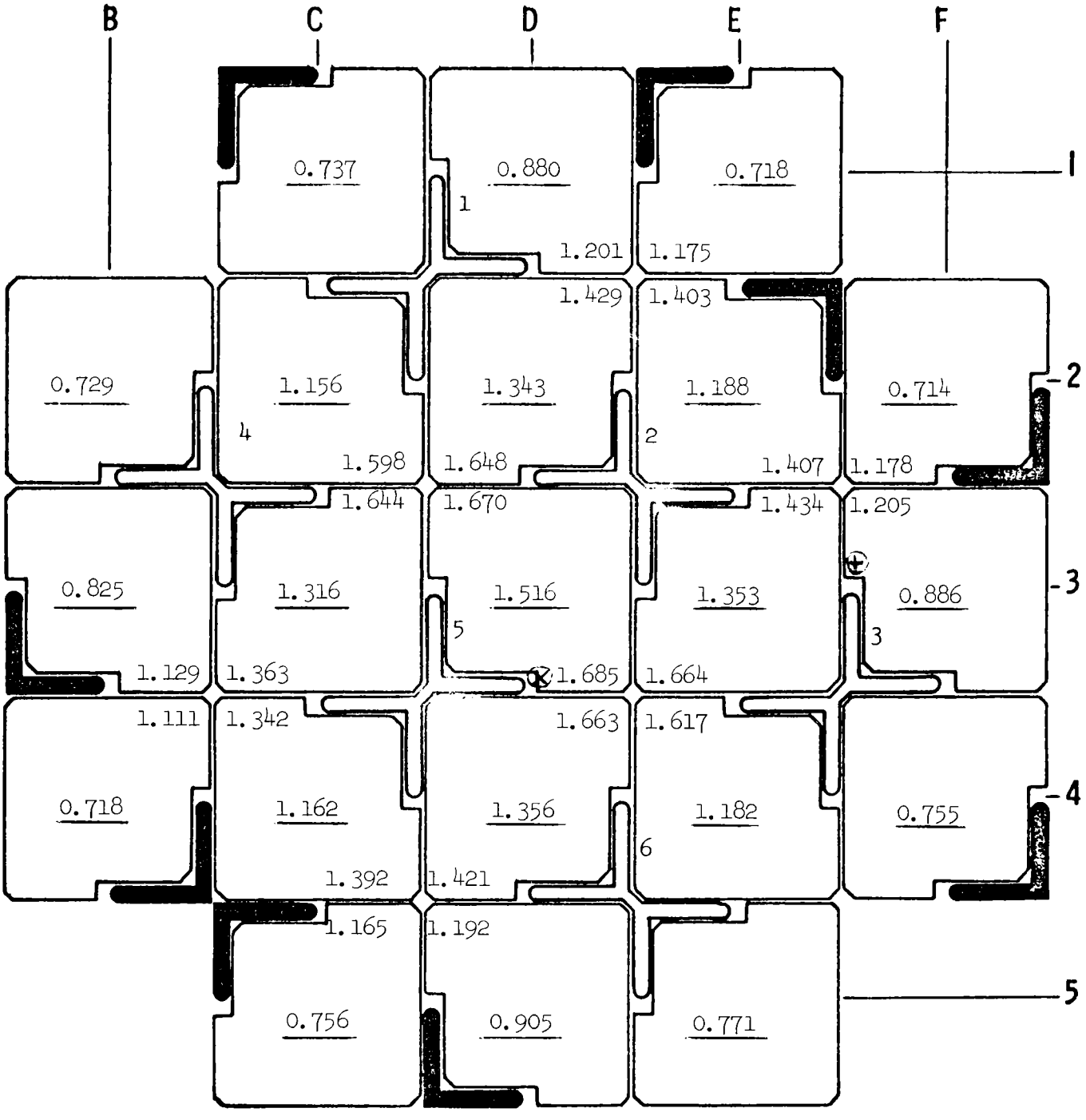


Figure 4. TURBO* Power Distribution after 6000 Hours at 23.5 MWt

4. Controls Summary

The nominal mode of control for the Saxton Plutonium Program is expected to be chemical shim. However a basic ground rule has been that the control of the reactor by either control rods alone, or chemical shim alone, or by any combination of the two shall not be restricted by the presence of the plutonium fuel. However, in either mode of control, soluble poison is required for cold, clean core shutdown.

Control rod worths were determined by means of a series of PDQ calculations. The calculated total bank worth is 16.9% $\Delta k/k$. With a maximum predicted excess reactivity of 11.1% $\Delta k/k$, the available shutdown margin is 5.8% $\Delta k/k$. Stuck rod calculations show that the total bank worth with the most effective rod stuck in the fully withdrawn position is 11.7% $\Delta k/k$. The shutdown margin for this condition is therefore 0.6% $\Delta k/k$. Since it is necessary that the capability exist for shutting the reactor down by at least 2% $\Delta k/k$ with the most reactive rod stuck in a pre-set withdrawn position, it may be necessary to do one of the following:

1. Restrict rod withdrawal initially by setting the presently installed limit switches.

2. Use chemical shim for xenon as well as temperature defect.

5. Reactivity Coefficients and Kinetics

During the quarter, the reactivity coefficients and kinetic characteristics of the Saxton plutonium core were investigated. Additional work is being carried out in the evaluation of the Doppler coefficient. Consequently, the reporting of the work in this area will be deferred until the next quarter. However, a qualitative summary is included in the opening paragraphs of this section.

SAX-220 Fuel Design - Mechanical, Thermal and Hydraulic

H. N. Andrews

A. Thermal Design

E. A. McCabe, D. Fischer

Saxton was originally designed to operate at 20 MWt using rodded control and at 23.5 MWt with chemical shim control. A heat rating of 16.0 KW/ft was was the maximum allowed in the first core and this was set in the spiked assembly. Future operation of the core at 28 MWt has been contemplated.

In the second Saxton core, the central nine fuel assemblies will contain PuO_2-UO_2 fuel clad in Zircaloy-4 instead of UO_2 fuel in stainless steel cladding. The fuel rods in seven of the PuO_2-UO_2 assemblies will contain pelletized fuel while the rods in the remaining two assemblies will contain vibratory compacted fuel. Fuel rod design is proceeding on the basis of equal heat ratings for both types.

Preliminary thermal calculations for a 28 MWt core were based on existing hot channel factors and fuel specifications. Under these conditions, high clad temperatures and temperature drops were found for the Zircaloy fuel rods and relatively large end gaps became necessary to accommodate the maximum allowable internal gas pressure. Considering the scarcity of high temperature stress-corrosion data available at the present time, an allowable fuel rod heat rate of 16 KW/ft was established for the design of the PuO_2-UO_2 fuel rods.

A second set of thermal calculations were made based on 16 KW/ft, on revised fuel specifications, and on new time-dependent hot channel factors established by physics. New fission gas release estimates were made and, using allowable pressures established by Mechanical Analysis, the required void volumes and end gaps for the pelletized plutonium rods were established. The axial length of these void spaces are .750" for Zircaloy rods and 2.625" for stainless steel rods. The required end gaps for vibratory compacted fuel are being calculated.

B. Mechanical Analysis

G. H. Eng, D. G. Frank

A preliminary investigation of clad stresses, maximum pellet temperature and diametral gaps between the pellets and the cladding was made based on a power rating of 18.6 KW/ft. As a result of this investigation, and considering possible variations in hot channel factors and in cladding temperatures, it was decided that the design ground rules should not be raised above the present 16 KW/ft at this time.

Based on 16 KW/ft, a new analysis of fuel temperature, clad stresses and diametral gaps has been initiated. This analysis is being carried out for both stainless steel and Zircaloy-4 cladding. For design purposes, the maximum allowable fission gas pressure has been established so as to limit the total clad stresses to 50% of the 0.2% yield stress at end of life.

C. Mechanical Design

N. J. Georges, E. A. Bassler

Preliminary mechanical designs of the Zircaloy-4 clad PuO_2-UO_2 fuel rods have been completed and issued for bidding purposes. Final drawings are nearing completion and will be issued in early October for fabrication.

The design of the fuel rod end plug weld is being revised from the original Saxton design to conform to a design proposed by Hanford for use in the vibratory compaction manufacturing process. To avoid possible contact of fine PuO_2-UO_2 powder with the weld area and therefore possible alloying of the Pu in the end plug weld, the loaded rod is welded in the vertical position. The Hanford weld is of the convex fillet type at the tube end and is readily accomplished in the vertical rod position. Experience at Hanford has established that this weld design yields a negligible rejection rate and gives reliable in-pile performance. It is planned to use this weld design for both vibratory compacted and pelletized fuel rods.

The grid design for the PuO_2-UO_2 fuel assembly enclosures will be the same as for the UO_2 fuel in the Saxton second core. The grid lattice is formed by interlocking sheet metal straps brazed together in an "egg crate" assembly.

SAX-230 Fuel Design - Materials R. J. Allio

A. Biancheria

Fuel and fuel rod loading specifications for vibrationally compacted and pelletized fuel were prepared and inquiries released to vendors for bids. Hanford was recommended to the AEC for fabrication of the vibrationally compacted fuel and NUMEC for the pelletized fuel. The specifications and proposed exceptions were reviewed in detail with each fabricator. Most exceptions were accepted and in the case of Hanford all differences have been resolved. Areas requiring additional discussion with NUMEC are: (1) welding procedures, (2) sampling plan, (3) inspection procedures, (4) hydrogen impurity level and (5) price adjustments. No problems are anticipated in resolving the remaining differences.

Zircaloy cladding will be used for the bulk of the fuel rods. Some stainless steel cladding will be included for comparison. The pelletized fuel will be prepared by the mechanically mixed process and the vibrationally compacted fuel will be prepared by Hanford's dynapak process.

A summary of performance with vibrationally compacted fuel was prepared for review by the Safety Analysis section and is included in this report at the end of this section starting with page 230-2 . The hydrogen pickup by the Zircaloy clad and the thickness of the ZrO_2 coating formed in-pile were calculated for linear rod powers of 16 kw/ft and assuming

40% down time for the reactor. The results showed that the hydrogen level at the end of life, 130 ppm, will be considerably below the level at which hydride problems occur. The ZrO_2 coating thickness and the temperature drop across the coating were found to be negligible.

A preliminary thermal conductivity-temperature curve for vibrationally compacted fuel was established from the available data. Flux depression factor-enrichment curves were calculated for plutonium of "Saxton" cross sections. The curves will be employed for estimating temperature profiles and temperature-reactivity relationships in the fuel.

Use of Vibrationally Compacted Fuel in Saxton Plutonium Program

A. Safety Considerations

1. Evaluation of Defect Potential

Technical feasibility of the vibrational compaction process has been demonstrated by satisfactory in-pile performance of a number of test samples and by irradiation of bulk quantities of fuel rods in the PRTR.*(1,2,3,4) While a number of defects occurred during the early stages of these tests, the causes were identified in almost every case and the net result has been increased confidence in the use of vibrationally compacted fuels. This increased confidence is reflected in the choice of such fuel for the EBWR program. Under this program, 1,296 Zircaloy clad fuel rods containing vibrationally compacted UO_2 - 1.5 w/o PuO_2 will be irradiated to exposures of 15,000 to 20,000 MWD/T.

*(1,2,3,4) - See pages 230-9 and 230-10 for all references on this subject.

Thirty-two of the thirty-three defects which occurred in PRTR have been attributed to fluoride contamination, excess moisture content, and traces of oil introduced by faulty powder attrition apparatus.[2,5,6,7,8,9] The defects occurred in both vibrationally compacted and swaged PuO_2-UO_2 fuel rods. The only defect which has not been explained to date occurred in a swaged rod. Investigation of this defect is continuing. The impurities cited are now being controlled and the results have been considered in developing specifications for the Saxton Plutonium Program. Since control of these impurities was initiated, one hundred and fifty seven fuel rods containing vibrationally compacted PuO_2-UO_2 fuel have been irradiated in PRTR to exposures of over 1,000 MWD/T (peak at 1600 MWD/T). These exposures are significant since all defects of vibrationally compacted rods prepared under old technology occurred at less than 400 MWD/T. Also, 560 rods from this latter group are still in pile and have attained exposures of 6,000 MWD/T without a defect.

Chloride, moisture, bad welds, and poor spacer design (resulting in fretting) were the main causes of the five defects which occurred at Savannah River during the early stages of the program.[10,11] Two defects which occurred in swaged rods[12,13] have not been explained to date and investigation is continuing. Since corrective measures were taken, they have had no defects in vibrationally compacted rods.[10,14] The rods have been irradiated to exposures of 9,000 MWD/T. While these rods contained only UO_2, there is no reason to suspect different behavior for PuO_2-UO_2 mixtures.

B. Evaluation of Water Logging and Washout Potential

During the early stages of development of loose powder fuels, fuel washout and waterlogging in the event of a defect were considered as possible performance limitations. These conjectures were based on (1) preliminary results of purposely defected fuel rods containing low density fuel (less than 85% T.D.) which was more susceptible to washout,[15,16] and (2) the reporting of a possible waterlogging failure of a swaged UO_2 fuel rod by Savannah River (details of this experiment are classified).[17,18] Apprehensions were greatly relieved when the former investigators reported the results of an unintentional defect in a swaged fuel rod containing UO_2 at 88% T.D.[19] Although the longitudinal split was 1.5 inches long and the reactor allowed to run at full power for 15 hours after the defect was detected, only a small amount of UO_2 (a maximum of 10% of the fuel in the area of the split) was eroded out of the rod. Apprehensions were further relieved when out-of-pile tests at the General Electric Laboratories indicated that vibrationally compacted, swaged and rolled UO_2 fuels had sufficient erosion resistance to prevent substantial losses to the coolant.[20,21]

More recent results, however, have shown clearly that the potential of these problems were greatly exaggerated during the early stages of development. No significant fuel washout and no waterlogging failures were observed for the 33 defects which occurred in PRTR during the last two years.[2,5,6,7,8] Although cladding losses

occurred in some cases, no severe reactor operating difficulties were reported. In some cases, the reactor underwent several pressure and power cycles after the defects were detected yet no waterlogging failures occurred.

The Hanford PRTR results are confirmed by the experience at the Savannah River Laboratories. They have never observed erosion of UO_2 from vibratory compacted or swaged fuel rods although they experienced seven defects including the previously mentioned possible waterlogging failure which had a 9 inch crack.[10,11,12,13,17,18]

The erosion resistance of loose powder fuels results from high density packing coupled with in-pile sintering. Evidence exists which indicates that in-pile sintering occurs at temperatures as low as 300°C.[22] Other evidence of enhanced sintering in a radiation field has also been reported,[23,24] These results may be explained by the following mechanism: (1) localized high temperatures resulting in increased rates for all sintering mechanisms,(2) increased bulk and surface diffusion rates resulting from increased vacancy concentrations, (3) enhancement of the vaporization-condensation mechanism through recoil processes.

Two waterlogging - washout type failures reported in 1962 should be mentioned since they caused some unnecessary anxiety. Neither failure can be considered applicable to the present situation. The first case reported was failure of a swaged, $MgO-PuO_2$ fuel rod

exposed for 8 MWD/T in the PRTR.[25] Failure resulted from interaction between the MgO and water resulting in swelling and loosening of the fuel compact. No such reaction occurs in PuO_2-UO_2 fuels. The initial defect apparently resulted from fluoride contamination and release of absorbed water from the MgO. The second case reported was failure of a swaged UO_2 fuel rod undergoing transient tests in SPERT.[26] The failure occurred during a 7.5 sec-period power excursion test in which fuel temperatures rose by 300°C within 0.02 sec. It is certain that even a waterlogged rod containing pellets would burst under these conditions. The initial defects apparently resulted from broken epoxy resin seals used to insert eleven thermocouples into the center regions of the rod. The rod underwent several power excursion tests and remained in the reactor water for two days prior to the last test. Fuel washout occurred because of the large opening, 12 inches long and up to 0.5 inches wide, and because the fuel was not in-pile long enough to sinter.

The two cases cited cannot be employed to evaluate failure probabilities in loose powder fuels. Similarly one cannot employ the waterlogging failure reported by the Bettis Laboratories to evaluate failure potential in pelletized fuel.[27] The failure in this case was attributed to low pellet density (80% T.D.) and reaction of water with uranium carbide contaminant in the UO_2 pellet. The initial defect was intentional.

B. Purpose of Test

A review of the literature indicates that, for future automated glove box type fabrication of noxious or radioactive fuels, the vibrational compaction process may prove to be cheaper than the pelletization process.[28,29,30,31] General agreement with this opinion is evidenced by the fact that almost every laboratory with interest in nuclear fuels is investigating the vibratory compaction process; for example, Argonne, Oak Ridge, Savannah River, Hanford, General Electric, Babcock and Wilcox, Combustion Engineering, Belgo-Nucleaire, Fiat and AECL Chalk River. Further, technical feasibility of the process has been demonstrated. The degree of effort which the national laboratories have expended in developing this process was indicated in Section I. It is clear that they have been carrying out their function of developing, testing, and solving the problems associated with a new and promising process. It now remains to carry out bulk engineering demonstration tests which are required for general acceptance of a new process by reactor designers and by utilities. Irradiation of significant quantities of vibratory compacted fuel rods in the Saxton Reactor will represent a major step toward this goal.

C. Summary

1. Based on technology already developed at the national laboratories, no defects in vibrationally compacted fuel rods are anticipated.

2. The results of the national laboratory experiments show that no significant fuel washout and no waterlogging results from defects in rods containing loose powder fuel.

3. The economic incentives for developing vibratory compacted PuO_2-UO_2 fuels and the present state of the art provide strong motivation for carrying out engineering demonstration tests in the Saxton reactor.

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SAX-250 Planning and Analysis of Critical Experiments

F. L. Langford, W. L. Orr, H. I. Sternberg

A. Introduction

1. Objective

The objective of this task is to plan, design, and analyze the critical experiments that will be conducted to verify the nuclear characteristics of the unirradiated fuel before it is installed in the Saxton reactor. These experiments and the associated analysis will provide an evaluation of nuclear design methods used in the calculation of reactivity and power distribution of the plutonium fuel rods and may indicate a change in fuel assembly location or mode of operation is desirable for the subsequent irradiation test.

2. Experiment Sequence

Separate single-zone criticals with both plutonium and uranium bearing fuel rods and two-zone criticals consisting of distinct plutonium and uranium regions are planned. Areas of particular interest include the relative power produced in the plutonium region, the local power at the region boundaries between the PuO_2-UO_2 and the UO_2 fuel, and the reactivity contribution of the plutonium fuel.

3. Scope of Work

The analytical work of the quarter was carried out in the following two areas:

a. A comparison was made of analysis with experiment for six mixed oxide (PuO_2-UO_2) critical and approach-to-critical experiments performed at the Hanford Laboratories.

b. The number of mixed oxide fuel rods of the design enrichment required for criticality was calculated as a function of lattice pitch.

In the next quarter, the analysis will be directed to the planning of the critical experiment measurements program, the analysis of multi-zone criticals, a study of kinetic characteristics, and continued effort toward resolving the discrepancies between analysis and experiment as shown by the initial comparisons of the Hanford experiments.

B. Results

1. Comparison of Analysis with Experiment

Hanford mixed oxide (1.5 w/o PuO_2-UO_2) critical and approach-to-critical experiments[1,2], for six different hexagonal lattice spacings, were evaluated using the LEOPARD code[3]. The Hanford experimentally determined bucklings were used in the calculations. Table 1 summarizes the comparison between analysis and experiment for two different Pu-239 cross section sets. More recent cross sections, designated Leonard cross sections, result in a calculated k_{eff} that is higher than the experimental value for each of the six lattices. The Leonard cross sections were used as the reference cross section set because of the more favorable comparison of analysis with burnup for the Yankee core[4]. However, earlier cross sections with a higher capture to fission ratio (α_{49}) result in better agreement with the experiments. The α of this cross section set is in agreement with a more recent review of the 2200 m/sec nuclear parameters by Sher[5].

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Table 1

Comparison of Hanford Experiments and LEOPARD Calculations*

Lattice Spacing (inches)	H/Pu (atom ratio)	Calculated k_{eff} Leonard cross-sections	Calculated k_{eff} Earlier cross-sections
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

Note: Experimental $k_{eff} = 1.0$

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* The calculations, except for the 0.55 inch lattice, were carried out by L. E. Strawbridge.

The LEOPARD code used in the comparison of the Hanford experiments determines fast and thermal spectra based on a modified MUFT - SOFOCATE 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 non-thermal 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. This factor is then used in a subsequent MUFT calculation of the fast spectrum. Thus the L factor corrects for heterogeniety and Doppler broadening of U-238. 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. The calculated Pu-240 L factor is within $\approx$ 1% of that determined in the iterative procedure for U-238.

Calculations were made of the effect on calculated k_{eff} for each of the six lattices for a discrepancy in the Pu-240 L factor between that determined from the Pu-240 resonance parameters, geometry, and temperature and that used by the program based on the U-238 correlation.

At the dilute Pu-240 concentrations of the six Hanford experiments a large discrepancy in L factor produces a small effect in k_{eff}. For example, in the 0.90 lattice the calculated k_{eff} was reduced from 1.027 to 1.0248 using the L factor calculated independently. Consequently, it was concluded that the error introduced by the program L factor was not significant in the discrepancy between analysis and experiment. At the larger concentrations of Pu-240, corresponding to the 6% enrichment proposed for the Saxton loading, an L factor discrepancy results in a much larger change in k_{eff}. However, at the higher concentrations little error is made in the calculation of L factor by means of the program equation. Therefore, the effect on k_{eff} is small.

2. Fuel Rods for Criticality

The LEOPARD code was used with AIM-5, a one-dimensional few-group diffusion calculation, to determine the number of fuel rods required for criticality as a function of lattice pitch. Figure 1 summarizes the results. A k_{eff} allowance of 2.5% is included to account for the discrepancy between analysis and experiment determined from the comparison of the Hanford experiments. Figure 1 shows that a minimum of 460 fuel rods is required for criticality in a 0.55-inch lattice. Since it is important that the critical experiment be conducted with rods of identical nuclear characteristics, a total of 500 fuel rods of one type was specified for the critical experiment.

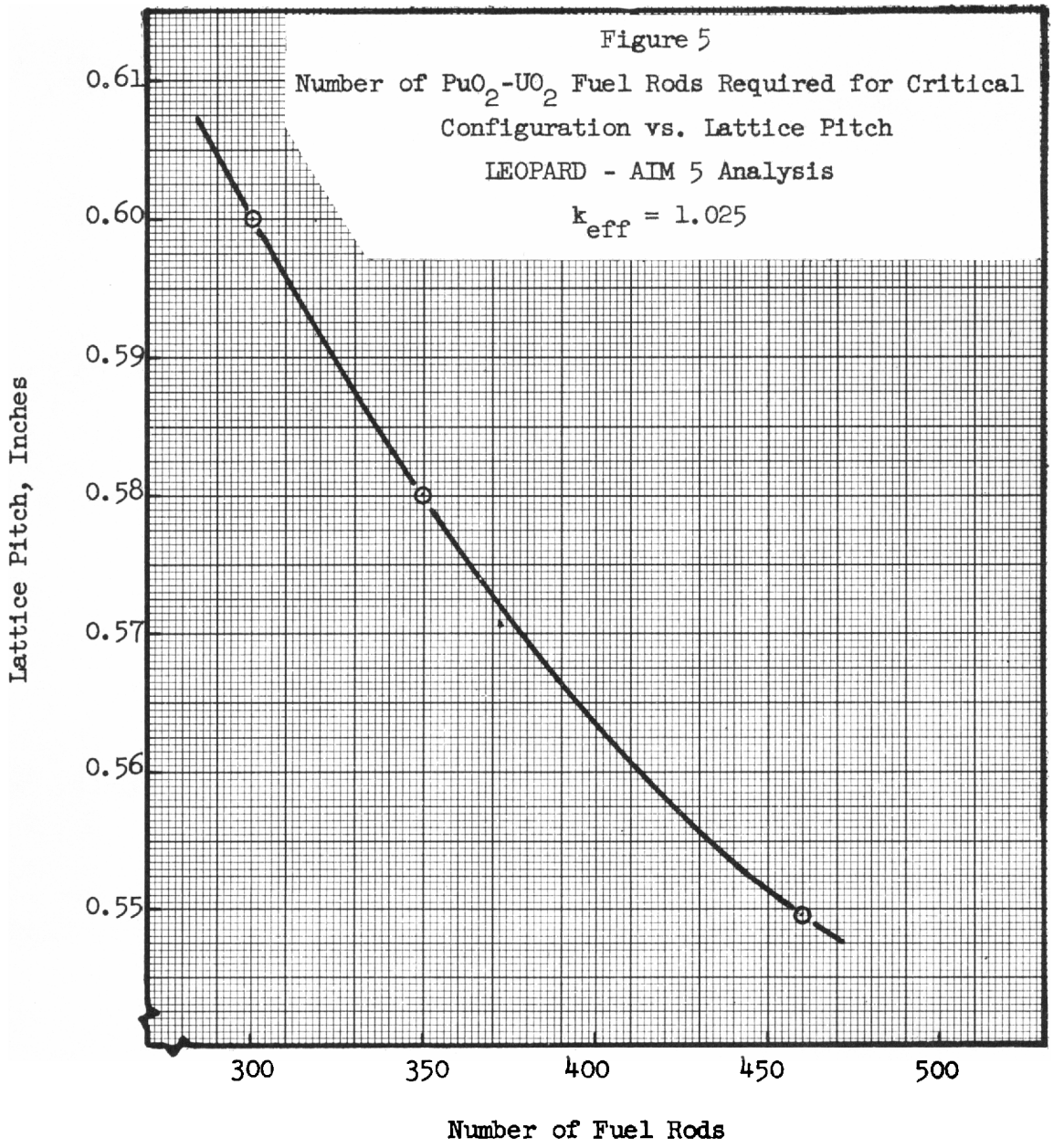
Figure 5
Number of PuO_2-UO_2 Fuel Rods Required for Critical Configuration vs. Lattice Pitch
LEOPARD - AIM 5 Analysis
k_{eff} = 1.025
Lattice Pitch, Inches
0.61
0.60
0.59
0.58
0.57
0.56
0.55
300
350
400
450
500
Number of Fuel Rods

SAX-310 Fuel Fabrication - Materials R. J. Allio

A. Biancheria

The costs, schedules, and deviations from bid specifications submitted by the NUMEC and the Hanford National Laboratories were reviewed with the respective laboratories. Agreement was reached with the Hanford Laboratories. Discussions with NUMEC are continuing. Resolution is expected early in October.

SAX-320 Fuel Inspection and Assembly

R. W. Brown

The objectives of this subtask are to assist vendors of materials and of fuel rods in inspecting their products to meet specifications, to conduct receiving inspection upon receipt of the fuel rods by Westinghouse, and to fabricate and inspect fuel assemblies.

During this period, specifications for fuel pellets and fuel rods were reviewed prior to their submittal to vendors for quotations. The vendors' comments on these specifications at the time of quoting were also reviewed and recommendations were made.

Purchase orders were placed for Zircaloy-4 tubing and for Zircaloy-4 end plug material. Stainless steel bar stock for end plugs is also on order. Excess stainless steel tubing procured for the original Saxton core is being evaluated for use in a small number of PuO_2-UO_2 fuel rods.

Springs and spacers will be ordered early in October.

SAX-330 New Fuel Shipping

H. E. Walchli

During this period work was limited to the preparation of information for and review of the PERT schedules and the preparation of shop order objectives and revised cost estimates.

SAX-340 Safeguards Analysis

R. A. Wieseman, R. C. Nichols, D. Yu

The PERT logic diagram for the project was reviewed and changes suggested to reflect critical items and dates for the preparation of the Safeguards Report and its submittal to the AEC. A preliminary and final Safeguards Report outline were prepared. The final outline is included below.

I. Introduction

- A. Objective and Scope
- B. Program Description

II. Mechanical Design

- A. Fuel Design and Characteristics
  - 1. Pellet Fuel
  - 2. Vibration Compacted Fuel
  - 3. Special Fuel
- B. Cladding Design
  - 1. Standard Fuel Rods
  - 2. Special Thick Clad Rods
- C. Fuel Assembly Design
  - 1. Improved Spring Clip Grid
  - 2. 9 x 9 and 3 x 3 Assembly Design

III. Thermal and Hydraulic Design

- A. Standard Fuel Assembly
- B. Special Fuel Assembly
- C. Design Conservatisms

IV. Nuclear Design

A. Introduction

1. Methods of Analysis

2. Comparisons of Analysis with Experiments

B. Reactivity Summary

1. Reactivity Available

2. Reactivity Coefficients

C. Control Summary and Effects

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2. Power Distributions

D. Core Lifetime Summary

1. K_{eff}

2. Burnup Characteristics

E. Kinetics Parameters

1. Delayed Neutron Fraction

2. Effective Neutron Lifetimes

V. Instrumentation

A. In-Core

1. Existing Instrumentation

2. Modifications to the System

B. Environmental Monitoring

VI. Accident Analyses

A. Reactivity Accidents

1. Uncontrolled Rod Withdrawal

2. Uncontrolled Heat Extraction

3. Boron Removal

B. Mechanical Accidents

1. Loss of Coolant

2. Loss of Flow

C. Maximum Hypothetical Accident

D. Conclusions

VII. Safety Considerations

A. Operation with Defected Fuel

B. Alpha Detection

C. Clean-up and Decontamination Procedures

VIII. Conclusions

The preliminary safeguards document for the plutonium core for the experimental boiling water reactor (EBWR) at the Argonne National Laboratory was reviewed for content and scope to aid in planning the Saxton Plutonium Safeguards Report. The final document will also be reviewed when issued later this year.

A study was undertaken to establish whether the use of plutonium-bearing fuel rods would present any additional radiation hazards both during normal operation and under maximum hypothetical accident conditions. The results indicate that the presence of plutonium enriched fuel would not add a significant amount of radiation hazards.

Under normal operation, among the factors that may contribute to radiation hazards are: reactor coolant fission product activity, fission product activity of spent fuels, excessive neutron and gamma-

ray fluxes, and any corrosive effect of coolant on the fuel.

The reactor coolant fission product activity for one percent defective fuel rods, including those enriched in plutonium, was found to be about the same as those with equivalent uranium enrichment. Therefore, radiation sources and dose rate surrounding the volume control tank and charging pump areas were found to be slightly changed from calculations based on enrichment in U-235 only.

The fission product activity of spent fuel was also found to be similar to that with enrichment in U-235 only. Thus, the present refueling shield water and storage well water facilities were found to be adequate.

A PIMG calculation was performed to determine the neutron irradiation and gamma ray heating inside the reactor vessel. Results indicate that the fast neutron flux inside the reactor vessel is only about four percent higher than that without plutonium enrichment. This increase, is, however, below the accuracy of the calculation method.

In order to determine if it is likely that fuel particles may be released into the coolant stream due to corrosive action of water on fuel, a study was performed to determine the oxidation and reduction properties of plutonium dioxide. It was found that plutonium dioxide is extremely stable both under oxidation and reduction conditions; and, furthermore, it is non-volatile even at 1775^{o}C. Therefore, any

release of plutonium dioxide fuel particles into coolant due to corrosive action of water is negligible.

Under maximum hypothetical accident conditions, we have to take into consideration any additional fission product activities associated with the fission yield of plutonium and any plutonium-bearing fuel particles that may be released into the containment.

A direct comparison of fission yields between U-235 and Pu-239 showed that the activity due to Pu-239 fission products is generally lower than that due to U-235 alone except for photons with an energy of 0.4 Mev. This leads to the conclusion that no additional radiation hazards with respect to fission products would be expected in a maximum hypothetical accident after 100 percent core meltdown at full thermal power of 23.5 Mw.

Since plutonium constitutes a major inhalation health hazard, effort was made to determine whether there would be any plutonium bearing fuel particles released into the atmosphere after a maximum hypothetical accident.

After a loss of coolant, which is assumed to lead to core melting, the cladding would react with the water vapor in the reactor pressure vessel. This oxidation reaction significantly contributes to the heat input into the core, so that some fragments of fuel particles may be released due to the impact of the reaction. These particles would, however, rapidly settle due to their own weight, and would

therefore not impose an inhalation hazard at this stage.

As the core temperature continues to rise, it is possible that some of the fuel material would reach its melting point. However, no fuel particles would be released since plutonium dioxide is quite non-volatile even at high temperatures.

Some accident and transient analyses have been started but delays in setting the fuel fabrication techniques have necessitated a revision of the Safeguards Report schedule.

The following critical items have been pinpointed and will be followed closely in order to assure that AEC approval of the plutonium core can be obtained within the framework of the original schedule:

1. Rapid completion and review of Safeguards Report by Westinghouse.
2. Expediting review of report by SNEC and Saxton Safety Committee.
3. Close coordination of AEC and ACRS review and approval of report.

SAX-350 Alpha Protection

R. Eastman, D. Yu

A review was made of the necessity of providing an alpha air monitoring system for the Saxton Plant while employing eight percent enrichment in plutonium in the outer core region. Results indicated that no alpha emitting nuclides will be released into air during normal operation, hence no permanent installation of an alpha air monitoring system will be required.

However, following a maximum hypothetical accident, some fuel particles may be released into the containment. Therefore, a portable alpha air monitor should be made available at entry into the vapor container following the accident.

An investigation was started of the instrumentation available to meet the above alpha monitoring requirements.

SAX-400 Performance of Critical Experiments

D. F. Hanlen

Applications were completed for amendment of License CX-6 to permit operation of Pu criticals and of License SNM-783 to permit storage of the fuel while at the Reactor Evaluation Center. Several telephone conferences with Division of Reactor Licensing personnel were held in connection with the contents of the applications and a meeting was held on September 28, 1964 with Division of Reactor Licensing personnel to discuss the applications in detail. At this meeting the details of experimental systems and of experimental programs were discussed. An official list of questions remaining to be answered is being prepared by the Division of Licensing.

Remaining Subtasks

F. Langford, et. al.

Work programs were written and cost estimates prepared for all remaining sub-tasks as follows:

SAX-510 Nuclear Analyses of Operation - F. Langford

SAX-520 Thermal-Hydraulic Analyses of Operations - E. A. McCabe

SAX-610 Post Irradiation Storage & Shipments - H. E. Walchli

SAX-620 Post Irradiation Examination - Transfer Building - D. T. Galm

SAX-630 Post Irradiation Examination - Hot Cells - D. T. Galm

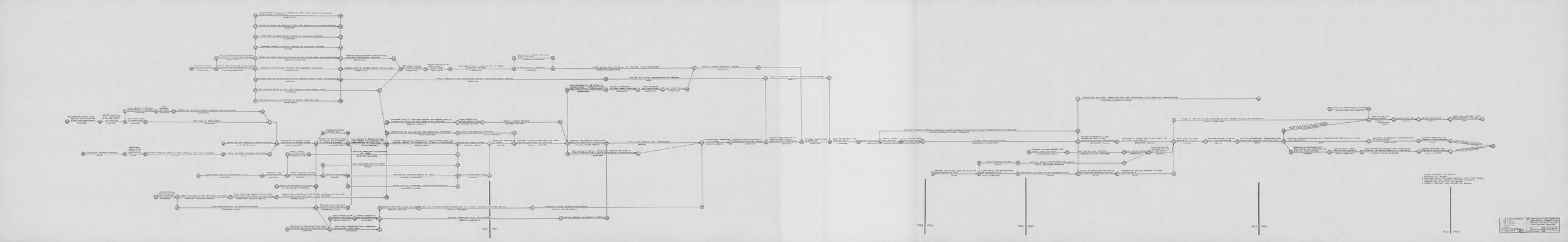
SAX-640 Post Irradiation Radiochemical Examination - B. D. Lamont

SAX-650 Waste Disposal - D. T. Galm

SAX-660 Materials Evaluation - R. J. Allio

SAX-670 Fuel Reprocessing - H. E. Walchli

Technical work in the preceding areas will commence much later in the program. A PERT-type summary schedule of the project is included at the end of this report.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