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PART IV

EUROPEAN ATOMIC ENERGY COMMUNITY - EURATOM

SAXTON PLUTONIUM PROGRAM

Quarterly Progress Report
for the period ending June 30, 1965

by

N.R. NELSON
(Westinghouse Atomic Power Division)

1966

EURATOM/US Agreement for Cooperation

EURAEC Report No. 1420 prepared by the
Westinghouse Electric Corporation, Pittsburgh, Pa. - USA

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SUMMARY

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All fuel rods to be used in Saxton Core II have been completed and are being installed in 9 × 9 enclosures. A 3 × 3 sub-assembly has been completed and has been operated satisfactorily in Saxton Core I for a short time.

A supplement to the Safeguards Report has been submitted. The supplement answers all questions raised to date by the ACRS and by the AED Division of Reactor Licensing.

Orders have been placed for all equipment needed for alpha protection.

The performance of critical experiments has been completed The experiments confirmed the nuclear design and no changes had to be made in the Safeguards Report.

CONTENTS

LIST OF FIGURES

SAX-100 Project Administration

N. R. Nelson

All fuel rods to be used in Saxton Core II have been completed by NUMEC and by Battelle and have been received by Westinghouse. Critical experiments with these rods have been completed and the rods are now being sent to Cheswick for installation into Saxton 9 x 9 enclosures. Completion is scheduled by the end of July.

The 3 x 3 subassembly containing four pelletized and four vipac rods plus a central instrumentation thimble has been completed, installed in a peripheral test hole in Saxton Core I and has been operated satisfactorily at power for a short time. During refueling in August, the subassembly will be shifted to the central test hole where it will be used for flux measurements during zero power and startup tests in Saxton Core II.

The Safeguards Report has been discussed at an ACRS sub-committee meeting and at a meeting with AEC Division of Licensing personnel. As a result of these meetings, questions have been received and answered by Westinghouse. A copy of these questions and the answers thereto are included at the end of the SAX-340 section in this report. The full ACRS committee meeting will be held on July 8th and approval action is expected.

Manuscript received on December 10, 1965.

A preliminary set of zero power and startup tests for Saxton Core II have been outlined.

License approval has been received for shipment of 9 x 9 PuO_2-UO_2 new fuel assemblies to Saxton. Shipment containers will be completed by mid-July.

SAX-210 Nuclear Fuel Design

F. L. Langford, W. L. Orr

The design work under this task was completed during the second quarter. A topical report is in preparation and will be issued during the first quarter of fiscal 1966.

SAX-220 Fuel Design - Mechanical, Thermal & Hydraulic

H. N. Andrews, N. J. Georges, E. A. Bassler

The objective of this subtask is to develop mechanical, thermal and hydraulic specifications and design for the PuO_2-UO_2 rods and assemblies.

The design and manufacture of the plutonium 3 x 3 fuel assembly was completed in this period and the assembly was installed in the Saxton reactor at the N-3 nozzle location.

The plutonium 3 x 3 fuel assembly is similar to previous Saxton 3 x 3 test assemblies in that it consists of a fuel subassembly and a latch assembly which is used for handling the subassembly and supporting it within the reactor. In the plutonium assembly, however, a flux wire thimble has been substituted in place of the center fuel rod. The fuel subassembly is contained on WAPD drawing 540F534 and the complete fuel assembly is contained on drawing 540F391.

The flux wire thimble is actually a part of the latch assembly and is supported axially from the conoseal adaptor at the top of the latch. Thus, when the latch assembly is disconnected from the fuel subassembly to gain access to the removable fuel rods, the thimble will be withdrawn from the subassembly. A special tool has been designed to aid in guiding the thimble back into the subassembly when the latch is reconnected. The tool is shown on drawing 540F762.

Engineering follow and consultation were provided during manufacture of fuel rods at NUMEC and at Battelle, during manufacture of the 3 x 3 subassembly at Forest Hills and at Cheswick, and during installation of the 3 x 3 subassembly in the N-3 hole in the Saxton reactor. Engineering follow at Cheswick will be provided during installation of fuel rods into the 9 x 9 enclosures.

SAX-230 Fuel Design - Materials

R. J. Allio, A. Biancheria

The work under this task leading to a set of material specifications was completed during the second quarter. A topical report is in preparation and will be issued during the first quarter of fiscal 1966.

SAX-250 Planning and Analysis of Critical Experiments

F. L. Langford, W. L. Orr, R. H. Chastain, H. I. Sternberg,
L. Bindler*, P. Deramaix**, R. J. Nath

A. Introduction and Summary

1. Introduction

The objective of this task is to plan, design, and analyze the critical experiments carried out at the Westinghouse Reactor Evaluation Center (WREC) to verify the Saxton plutonium nuclear design. The same fuel rods used in these experiments will be used in Saxton Core II, which will be operated in the Saxton reactor for about two years.

During this quarter, the WREC critical experiment program was completed. A detailed program of analysis is now in progress for comparison with the experimental results.

2. Summary

The following statements briefly summarize the work performed during the quarter:

a. The measurements program described in the previous quarterly[1] was completed. A major portion of the required data processing was also completed.

b. A criticality study for both the Hanford and WREC critical experiments was carried out using the LEOPARD[2] and LASER[3]

On leave from CEN, Mol Belgium and Belgo-Nucleaire, Brussels, Belgium** working on the Saxton Plutonium Program in the scope of the EURATOM/AEC/Westinghouse Contract.

codes. LASER, which includes a modified version of the THERMOS[4] code, produces a small difference in reactivity and in the reaction rate in Pu-239 and Pu-240 from that of LEOPARD. The use of different thermal cross section sets was also investigated. The cross sections reported by Wescott[5] at the 1964 Geneva Conference have been selected as the basic set to be used in the post-critical comparison of analysis with the WREC experiments.

c. The analysis of certain specific experimental configurations was completed and a comparison of the results with experimental values has been made. The comparisons show the following:

   (1) Reactivity calculations carried out in advance of the critical program are in excellent agreement with the experimental values when an allowance in calculated k_{eff} based on an analysis of Hanford criticals is included. This comparison confirms the validity of including the allowance in the Saxton design calculations and indicates that only a small revision in the reactivity and lifetime predictions are necessary as a result of the experimental information obtained.

   (2) Post-critical reactivity calculations using LASER and revised thermal cross sections agree well with experiment without the necessity of including an allowance in calculated k_{eff}. The correlation using LEOPARD is also improved with the revised cross sections.

(3) The measured reactivity worth of boron in a two-zone configuration simulating the Saxton design with an inner region of PuO_2-UO_2 fuel and an outer region of UO_2 fuel was in good agreement with the predicted boron concentration requirement.

(4) Power peaking effects were investigated in single region cores composed of UO_2 and PuO_2-UO_2 fuels. The analysis over-predicts power peaking for both fuels. Consequently, the calculated hot-spot factors for the Saxton design are believed to be conservative.

(5) The measurement of relative power by fuel rod gamma scan in cores composed of two different types of fuel is subject to error when the gamma decay characteristics and the energy per fission of the two fuels are different. Therefore, an experiment was carried out to relate the heat-rate to gamma activity after shutdown for both uranium and plutonium fuels. Based on these results, a time-dependent factor was developed to relate gamma activity to rod power. The factor was used in a comparison of analysis with experiment for two-zone cores. A second method involving the irradiation of foils of the two fuel materials was carried out. The time-dependent gamma decay in the two foil types was related to the number of fissions in each. This method

served as a check on the heat-rate experiment. While there are small differences in the two experiments and in the PDQ-3[6] analysis, previous calculations of the power sharing in multi-region cores are satisfactory.

B. Scope of the Experimental Program and Supporting Analysis

The planned measurements program was outlined in the previous quarterly report. During this quarter, the program was completed. In carrying out the program, certain changes were made in the sequence of experiments and it was necessary to add a number of experiments to those originally planned. To illustrate the extent of the measurements that were carried out and the scope of the analysis that is now in progress, a revised summary of experiments is included in Table 250.1.

The processing of the data from the experiments summarized in Table 250.1 required about one-half of the analytic effort of the quarter.

C. Criticality Study

1. Objective

Previous reactivity calculations using the LEOPARD code resulted in an average discrepancy of $\approx$ 2.6% $\Delta k/k$ for six

TABLE 250.1

MEASUREMENT PROGRAM SUMMARY

Single-Region Experiments

PuO_2-UO_2 Fuel

0.56-Inch Lattice

a. Cylindrical core-critical rods
b. Buckling by fuel rod scans (19 x 19 core)
c. Power map, foil traverses-U238*, Au, Dys (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. Vipac vs pelletized fuel-reactivity, power map
i. 3 x 3 UO_2 insert-reactivity, power map, flux-Dys
j. Fuel rod circumferential flux-Dys 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)

0.795-Inch Lattice

a. Cylindrical core-critical rods
b. Buckling by fuel rod scans (12 x 12 core)
c. Power map

0.52-Inch Lattice

a. Cylindrical core-critical rods
b. Buckling by fuel rod scans (22 x 23 core)
c. Pulse neutron experiments (23 x 23 core)

0.735-Inch Lattice

a. Cylindrical core-critical rods
b. Buckling by fuel rod scans (13 x 13 core)

1.04-Inch Lattice***

a. Cylindrical core-critical rods
b. Buckling by fuel rod scans (11 x 11 core)

UO_2 Fuel

0.56-Inch Lattice

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-Dys wire
k. Pulsed neutron experiments

0.795-Inch Lattice

a. Cylindrical core-critical rods
b. Buckling by fuel rod scans (13 x 14 core)
c. Power map

*U238 foils counted for fission activity and Np239 decay.

**Power measurements made by fuel rod scans.

***Not part of Saxton Plutonium Program. Included to complete the summary of the buckling data available.

Multi-Region Experiments

PuO_2-UO_2 Inside, UO_2 Outside

Clean

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, Dys
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

Borated Core

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, Dys
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

UO_2 Inside, PuO_2-UO_2 Outside

Clean

a. Fuel substitution in steps to inverted reference core (11 x 11 UO_2 inside, 19 x 19 overall)
b. Power map
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)

Borated Core

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, Dys

mixed-oxide (PuO_2-UO_2) critical and/or approach-to-critical experiments conducted at Hanford[7]. An allowance to account for this difference was included throughout the design calculations for the Saxton plutonium core and in the criticality predictions for the WREC critical experiments. A criticality study was completed during the quarter to investigate the reasons for this difference and to determine the best methods and cross sections to be used in the post-critical analysis of the WREC experiments. Specifically the study was directed to an investigation of the effect of:

a. variations in the heterogeneous thermal treatment of the cell,
b. variations in the scattering kernel for H_2O,
c. and variations in the thermal parameters for U-235, Pu-239, and Pu-241.

2. Methods

To study these variations, LASER and LEOPARD calculations were compared. The basic difference between these two programs is in the calculations performed in the thermal energy group. In LASER, the thermal calculation consists of a modification of the THERMOS code, a cell transport theory code in space and energy, that is expanded in energy to a cut-off of 1.855 ev. Thus the Pu-240 resonance at 1.05 ev is included in the thermal range. 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 a 0.625 ev cut-off. Both codes use a consistent B-1 MUFT IV calculation in the fast energy group. Therefore the difference in the fast group is the energy level at which the fast group ends.

3. Hanford Experiments

The heterogeneous treatment of the unit cell in space and energy in LASER leads to a spatially varying spectrum which in all regions of the cell is harder than the mean spectrum of the cell determined in LEOPARD. The harder spectrum results in a difference in the reaction rates in the plutonium isotopes and a difference in the calculated reactivity. The reactivity results from LASER and LEOPARD for the Hanford mixed-oxide experiments are summarized in Table 250.2 and Figure 250.1. The LASER results also show 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 250.2 the most favorable comparison of analysis with the Hanford experiments was obtained using a LASER calculation with the Nelkin kernel.

TABLE 250.2

CALCULATED REACTIVITY FOR HANDORD MIXED-OXIDE (PuO_2-UO_2)
EXPERIMENTS USING LEOPARD AND LASER

Lattice Pitch, in.	H/Pu	Calculated k_{eff} LASER, Nel*	LASER, F.G.**	LEOPARD
0.55	230	1.00666	1.01058	1.01652
0.60	326	1.01123	1.01484	1.02397
0.71	567	1.01761	1.02065	1.03144
0.80	794	1.01705	1.01983	1.02968
0.90	1077	1.01791	1.02066	1.02719

All calculations use Leonard cross sections.

*Nelkin kernel

**Free-gas kernel

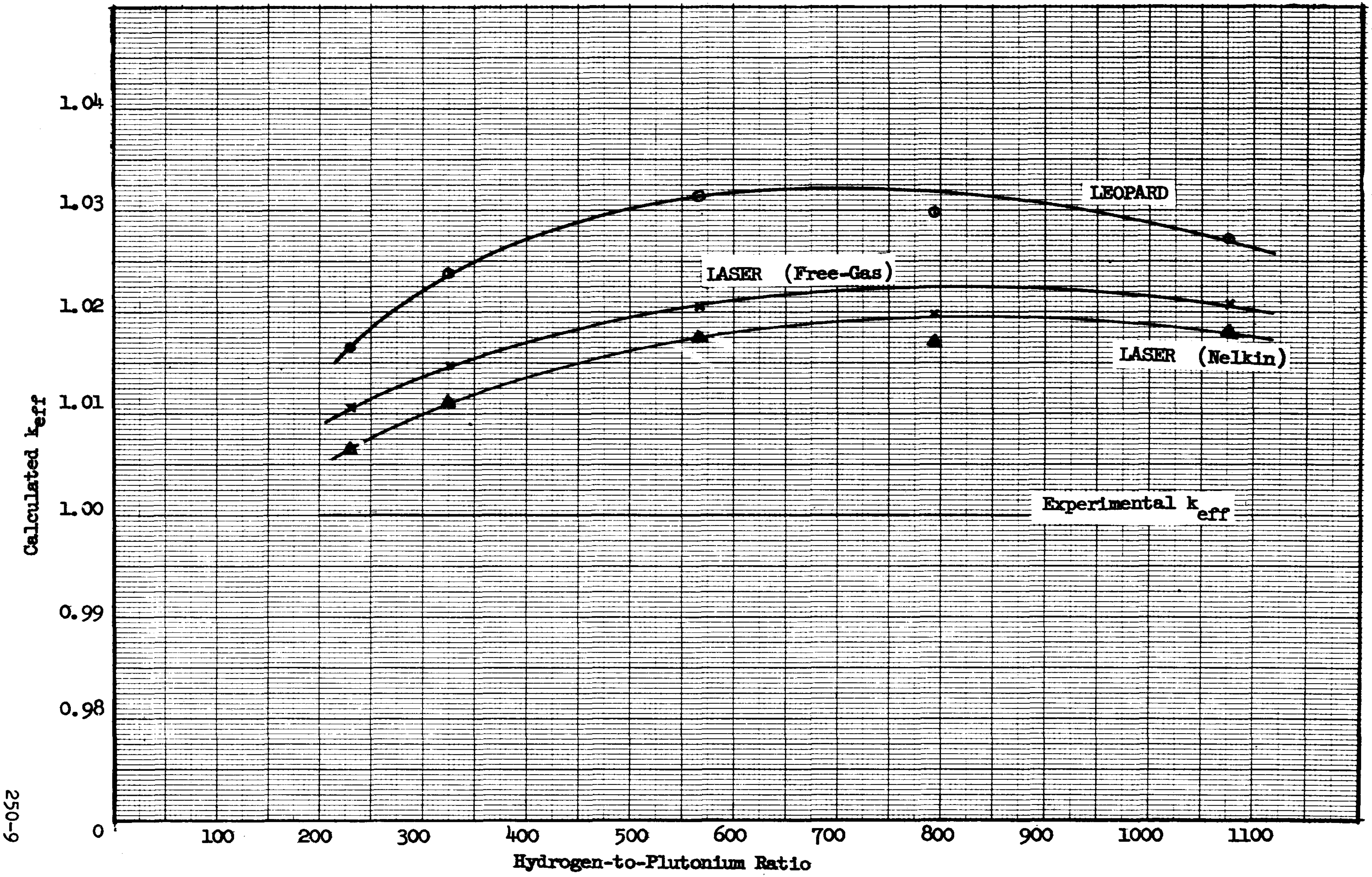

Figure 250.1 A Comparison of LASER Calculations with a Variation in the Scattering Kernel with LEOPARD Calculations for Five Hanford Mixed-Oxide (PuO_2-UO_2) Experiments.

As part of the criticality study, the influence of a variation in the thermal parameters of U-235, Pu-239, and Pu-241 was also investigated. Table 250.3 summarizes the 2200 m/sec parameters for three different cross section sets.

The three cross section sets of Table 250.3 were used in a series of LEOPARD and LASER calculations for four of the Hanford experiments. The results are summarized in Table 250.4 and Figure 250.2. The results show that with LASER the most favorable comparison of analysis with experiment is obtained with the 1964 Geneva Conference cross sections while for LEOPARD the best correlation is obtained with the Sher cross sections.

4. WREC Experiments

Criticality calculations were carried out using the LASER and LEOPARD code for two of the WREC critical lattices for both UO_2 and PuO_2-UO_2 fuels. The following results were obtained using Leonard cross sections.

	Calculated k_{eff}				
	UO_2			PuO_2-UO_2	
Lattice Pitch, In.	LEOPARD	LASER(F.G.)	LASER(Nel)	LEOPARD	LASER(Nel)
0.56	1.00478	1.00380	1.000217	1.01944	1.01822
0.792	1.00016	1.00407	0.99827	1.03065	1.02290

TABLE 250.3

CROSS SECTION PARAMETERS AT 2200 M/SEC FOR

THREE CROSS SECTION SETS

		Cross Sections (2200 m/sec)		
		Geneva Conf. 1964	Leonard[8]	Sher[9]
U-235	σ_a	678.4 ± 1.9	679.1	682.0 ± 2.6
	σ_f	577.5 ± 1.6	580.5	582.2 ± 2.2
	α	0.1748 ± 0.0018	0.1699	0.171 ± 0.003
	ν	2.44242 ± 0.0066	2.4388	2.430 ± 0.009
	η	2.0790 ± 0.0055	2.0846	2.074 ± 0.006
Pu-239	σ_a	1010.6 ± 4.3	1008.2	1030.1 ± 7.4
	σ_f	745.9 ± 3.3	752.8	743.2 ± 4.9
	α	0.3548 ± 0.0047	0.3393	0.377 ± 0.011
	ν	2.8759 ± 0.0020	2.8904	2.882 ± 0.016
	η	2.1227 ± 0.0089	2.1582	2.093 ± 0.014
Pu-241	σ_a	1376.1 ± 24.7	1371.2	
	σ_f	1012.7 ± 6.7	963.2	
	α	0.3589 ± 0.0252	0.4235	
	ν	2.9779 ± 0.0205	3.0209	
	η	2.1913 ± 0.0439	2.1221	

TABLE 250.4

CALCULATED REACTIVITY FOR HANFORD MIXED-OXIDE (PuO_2-UO_2)

EXPERIMENTS WITH DIFFERENT CROSS SECTIONS AND SCATTERING KERNELS

Cross Section Set	Analysis Method	Calculated k_{eff}			
		0.55-In. Lattice	0.60-In. Lattice	0.71-In. Lattice	0.90-In. Lattice
Geneva 1964	LEOPARD	1.00387	1.01016	1.01699	1.01326
Geneva 1964	LASER (F.G.)	0.99793	1.00103	1.00620	1.00673
Geneva 1964	LASER (Nel)	0.99401	0.99742	1.00316	1.00398
Leonard	LEOPARD	1.01652	1.02397	1.03144	1.02719
Leonard	LASER (F.G.)	1.01058	1.01484	1.02065	1.02066
Leonard	LASER (Nel)	1.00666	1.01123	1.01761	1.01791
Sher	LEOPARD	0.99563	1.00185	1.00914	1.00681
Sher	LASER (F.G.)	0.98968	0.99272	0.99835	1.00028
Sher	LASER (Nel)	0.98576	0.98911	0.99531	0.99753

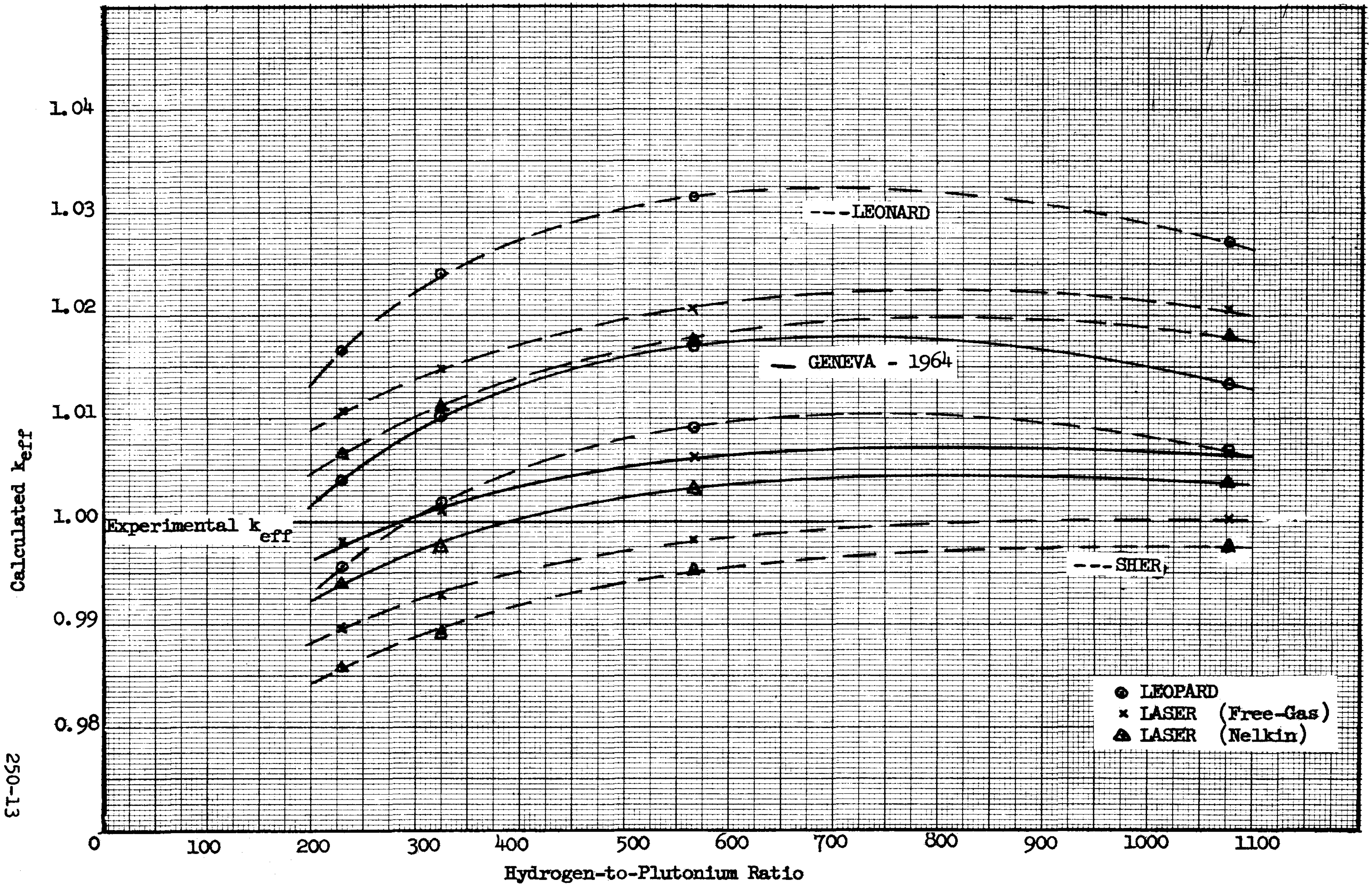

Figure 250.2 A Comparison of LASER and LEOPARD Calculations for Three Different Sets of Thermal Parameters for U-235, Pu-239, and Pu-241.

For the UO_2 critical lattices both LASER and LEOPARD are in good agreement with the experiments while for the PuO_2-UO_2 fuel the results obtained with LASER are slightly better for the two lattices than those obtained with LEOPARD. The discrepancy between the calculated k_{eff} and experiment for the LEOPARD results for both PuO_2-UO_2 lattices averages $\approx$ 0.025. This is the same allowance as that included in the criticality predictions for the PuO_2-UO_2 experiments.

The use of Sher cross sections for the UO_2 experiments would result in an unsatisfactory comparison of analysis with experiment. Therefore, LEOPARD* calculations for both fuels were carried out using the 1964 Geneva Conference cross sections. In addition, LASER calculations were carried out for the PuO_2-UO_2 experiments. The following results were obtained:

Lattice Pitch, In.	Calculated k_{eff} UO_2	PuO_2-UO_2 LEOPARD	PuO_2-UO_2 LASER(F.G.)
0.56	1.00589	1.00839	0.998
0.792	1.00205	1.01749	1.003

*The LEOPARD code used contains a number of small revisions from that used previously. The changes include the removal of k bias, a revised Dancoff, a revised SOFOCATE integration, and a correction in a U-235 cross section. The net effect on calculated k_{eff} due to these changes is small.

Good agreement between analysis and experiment was obtained for the UO_2 critical experiments and an improvement in the comparison was obtained for the PuO_2-UO_2 experiments. In the case of LASER the agreement for both lattices is excellent.

Based on this study the 1964 Geneva Conference cross sections were selected as the most satisfactory set to be used in the post-critical evaluation of the WREC experiments.

D. Comparison of Analysis with the WREC Experiments

1. Buckling Measurements and Criticality Calculations

Critical buckling measurements were made for five different lattices with the PuO_2-UO_2 fuel and two different lattices for the conventional Saxton UO_2 fuel. The basic lattice used for a major part of the experimental program was that containing 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 was also measured in a borated configuration containing 337 ppm boron in the 0.56 inch lattice. Table 250.5 contains a summary of the experimental results.

The measured bucklings of Table 250.5 were used in the revised version of LEOPARD with the 1964 Geneva Conference cross sections. These results for the PuO_2-UO_2 configurations are summarized in Table 250.6.

TABLE 250.5

BUCKLING AND REFLECTOR SAVINGS RESULTS FOR THE WREC CRITICAL EXPERIMENTS

	Lattice Loading	No. of Critical Rods	Critical Buckling (CM^{-2}) x 10^3	Radial Reflector Savings (CM.)	Axial Reflector Savings (CM-)	Water Temp. (°C)
Uranium Oxide	UO_2 5.74 w/o U235 SS Clad 0.792" Pitch	182 (13 x 14)	13.68 ± 0.19	12.43 ± 0.41	7.78 ± 0.08 (Bott.) 0.21 ± 0.04 (Top)	17.3
Uranium Oxide	0.56" Pitch	361 (19 x 19)	12.71 ± 0.14	14.31 ± 0.25	8.77 ± 0.14 (Bott.) 2.65 ± 0.05 (Top)	18.0
Plutonium Oxide	PuO_2-UO_2 6.6 w/o PuO_2 Zr.-4 Clad 0.792" Pitch	144 (12 x 12)	15.93 ± 0.22	12.90 ± 0.20	6.47 ± 0.12 (Bott.) 3.05 ± 0.12 (Top)	16.1
Plutonium Oxide	0.56" Pitch(2)*	361 (19 x 19)	12.15 ± 0.08	15.10 ± 0.14	8.10 ± 0.29 (Bott.) 4.52 ± 0.14 (Top)	16.4†
Plutonium Oxide	0.56" Pitch 337 ppm Boron	441 (21 x 21)	11.23 ± 0.10	13.98 ± 0.18	8.38 ± 0.15 (Bott.) 4.99 ± 0.15 (Top)	18.0
Plutonium Oxide	0.735" Pitch	169 (13 x 13)	15.96 ± 0.19	12.78 ± 0.25	6.83 ± 0.17 (Bott.)	24.1
Plutonium Oxide	0.52" Pitch	506 (22 x 23)	10.88 ± 0.13	15.76 ± 0.44 (22 Rods) 14.51 ± 0.37 (23 Rods)	8.56 ± 0.29 (Bott.) 4.86 ± 0.29 (Top)	25.8
Plutonium Oxide	1.04" Pitch	121 (11 x 11)	12.84 ± 0.14	12.27 ± 0.10	6.23 ± 0.09 (Bott.) 3.80 ± 0.09 (Top)	19.9

*Number in parentheses indicates number of experiments performed.

†Average temperature of the two experiments.

TABLE 250.6

CALCULATED REACTIVITY FOR THE WREC PuO_2-UO_2 CRITICAL EXPERIMENTS USING THE LEOPARD CODE AND THE 1964 GENEVA CONFERENCE CROSS SECTIONS

Lattice Pitch, inches	Measured Buckling	Calculated k_{eff}
0.52	10.88 ± 0.13	0.9890
0.56	12.15 ± 0.08*	1.0103
0.56 (337 ppm Boron)	11.23 ± 0.10	1.0148
0.735	15.96 ± 0.19	1.0128
0.792	15.93 ± 0.22	1.0175
1.040	12.84 ± 0.14	1.0167

*Average of two measurements.

2. Reactivity, Power Peaking, Power Sharing

In the last quarter[1], the measurements program was summarized. In that summary the number of fuel rods required for criticality and the boron content requirements for the expected configurations was included. The analysis on which these predictions were based was carried out using the LEOPARD-PDQ codes with Leonard cross sections. While the previous discussion shows that an improvement in the correlation can be obtained using a different cross section set, a comparison of the predictions with the measurements using the same methods as those used in the initial Saxton design calculations is necessary to determine if the expected performance is adversely affected by a difference between the analysis and experiment that may be indicated.

Three important areas from the standpoint of their influence on the operation of a plutonium core in the Saxton reactor are the following:

a. Reactivity - The initial reactivity available in the design is important from the standpoint of both lifetime and control.

b. Power Peaking Effects - In the Saxton design, the power level at which the core can be operated is limited by the maximum hot-spot that occurs at water slots within the plutonium region. Thus, it is important to know if the analysis correctly predicts power peaking effects in regions of increased moderation.

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 would be necessary to reduce the total core power to avoid exceeding the hot-spot limitation.

a. Reactivity

A comparison of the number of fuel rods required for criticality with the predicted requirements for both the UO_2 and the PuO_2-UO_2 critical configurations shows that the analysis and experiment are in good agreement. The analysis predicted 356 fuel rods would be required for criticality in a square core with a 0.56 inch pitch for the UO_2 fuel. A total of 346 rods was actually needed. For a square core of PuO_2-UO_2 fuel at the 0.56 inch pitch, the expected fuel rod requirement was 355 rods. The actual requirement was 343 rods. In the prediction of fuel rod requirements for configurations containing PuO_2-UO_2 fuel, an allowance was included to account for a possible discrepancy between analysis and experiment. As discussed in a previous paragraph, the discrepancy between the experiment and the calculated k_{eff} using LEOPARD for the WREC 0.56 inch and 0.792 inch lattices averaged $\approx$ 0.025

which is the same as the allowance included in the criticality predictions for the PuO_2-UO_2 experiments. The same methods of analysis and cross sections were also used in the Saxton design calculations. Because good agreement was demonstrated in the reactivity predictions, only a small revision in the original reactivity and lifetime predictions for the design core is necessary at this time. Another major test of the analysis methods will be available after the zero power physics tests are completed in the Saxton reactor under a separate task.

Based on the analysis of a two-region borated core, a just-critical boron concentration of 1525 ppm was expected. A boron worth measurement was made at 1430 ppm boron at partial water height for a core consisting of a 27 x 27 rod assembly with an inner region of 361 PuO_2-UO_2 fuel rods (19 x 19) and an outer region of 368 UO_2 fuel rods. Extrapolating the measurement to the boron requirement for a fully inundated core indicates $\approx$ 1550 ppm would be required. Thus, no adverse effects are expected in the design core due to a discrepancy in boron worth.

b. Power Peaking Effects

In the single region cores, power peaking effects were investigated near water slots. In these cores, 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 an aluminum slab installed in the water slot to displace part of the water. Conventional methods of analysis, LEOPARD-PDQ-3, were then applied to the specific experimental configurations. The following comparison of analysis with experiment was obtained for the rod nearest the slot where the maximum error occurs. (Both analysis and experiment are normalized to a rod that is not influenced by the slot.)

	Analysis/Experiment	
	UO_2 Core	PuO_2-UO_2 Core
H_2O Slot	1.056	1.078
Al-H_2O Slot	1.010	1.040

In these calculations, the group constants used for the water slots were obtained from a LEOPARD calculation using the material composition of the slot alone. Thus the constants, designated soft-spectrum constants, were determined by the use of a flux spectrum that is not representative of the spectrum that exists in the slot. For the UO_2 experiments, constants for the slot were also 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 were then determined from the group averaged microscopic cross sections and the number density of the slot materials. The following list compares the results obtained using the two methods.

	Analysis/Experiment	
	Soft-Spectrum	Extra Region
H_2O Slot	1.056	1.026
Al-H_2O Slot	1.010	1.003

Additional calculations in the PuO_2-UO_2 cores are now in progress using cross sections selected for the post-critical comparisons and alternate methods suggested by these initial results.

The results of these studies show the analysis over-predicts the power peaking near water slots for both fuels. The largest discrepancy occurs in the plutonium fueled cores. Consequently, the hot channel factors calculated for the Saxton design are believed to be conservative.

c. Power Sharing

In cores composed of different types of fuel, it is difficult to determine the relative power production by a gamma scan of fuel rods if the gamma source and decay characteristics of the two fuels are different. Since the Saxton core will contain separate regions of uranium and plutonium fuel, it is necessary to know the amount of power produced in each region to avoid exceeding an imposed hot-spot limit expected to occur in the plutonium region near water slots. Consequently, power measurements in two-region cores were carried out during the WREC critical program. However, to interpret the data it is necessary to relate the measured gamma activity to the power produced in the fuel rod.

Two different methods were used to determine the desired relationship. In the first, an experiment was conducted in which the heat-rate in the fuel rods was measured and related to the gamma activity after shutdown as determined by the subsequent gamma counting of the rods. Three separate measurements were performed using various uranium and plutonium fuel rods. The first heat-rate experiment was made with two UO_2 fueled rods of different enrichment, 1.6 w/o enriched UO_2 and 3.7 w/o enriched UO_2. Since the gamma activation in each uranium fuel rod is proportional to power, the ratio of gamma activity to heat-rate would be a constant if the heat-rate was also directly proportional to power. Figure 250.3 shows that the ratios for the two rods were the same. Therefore, it was concluded that the method was a reasonable one to use in a comparison of uranium and plutonium fuels.

The same type of experiment was carried out for the Saxton UO_2 fuel rods (5.7 w/o U-235) and the PuO_2-UO_2 fuel rods (6.6 w/o PuO_2) made with both vibratory-compacted and pelletized fuel. The resulting ratio of gamma activity after shutdown to the thermal power in a uranium fuel rod relative to the same ratio in a plutonium fuel rod as a function of time after shutdown is shown in Figure 250.4. This curve represents a time-dependent multiplication factor that is applied to the measured gamma activity in the plutonium fuel. The size of the factor used depends on the time after shutdown the rod is scanned.

The second method* of relating gamma activity to power involved the irradiation and subsequent gamma scan of foils composed of the two different fuel materials, foils of Pu-U-Al from the PuO_2-UO_2 and foils of U-Al from the UO_2.

*Data developed by the foil irradiation method was supplied by G. N. Hamilton.

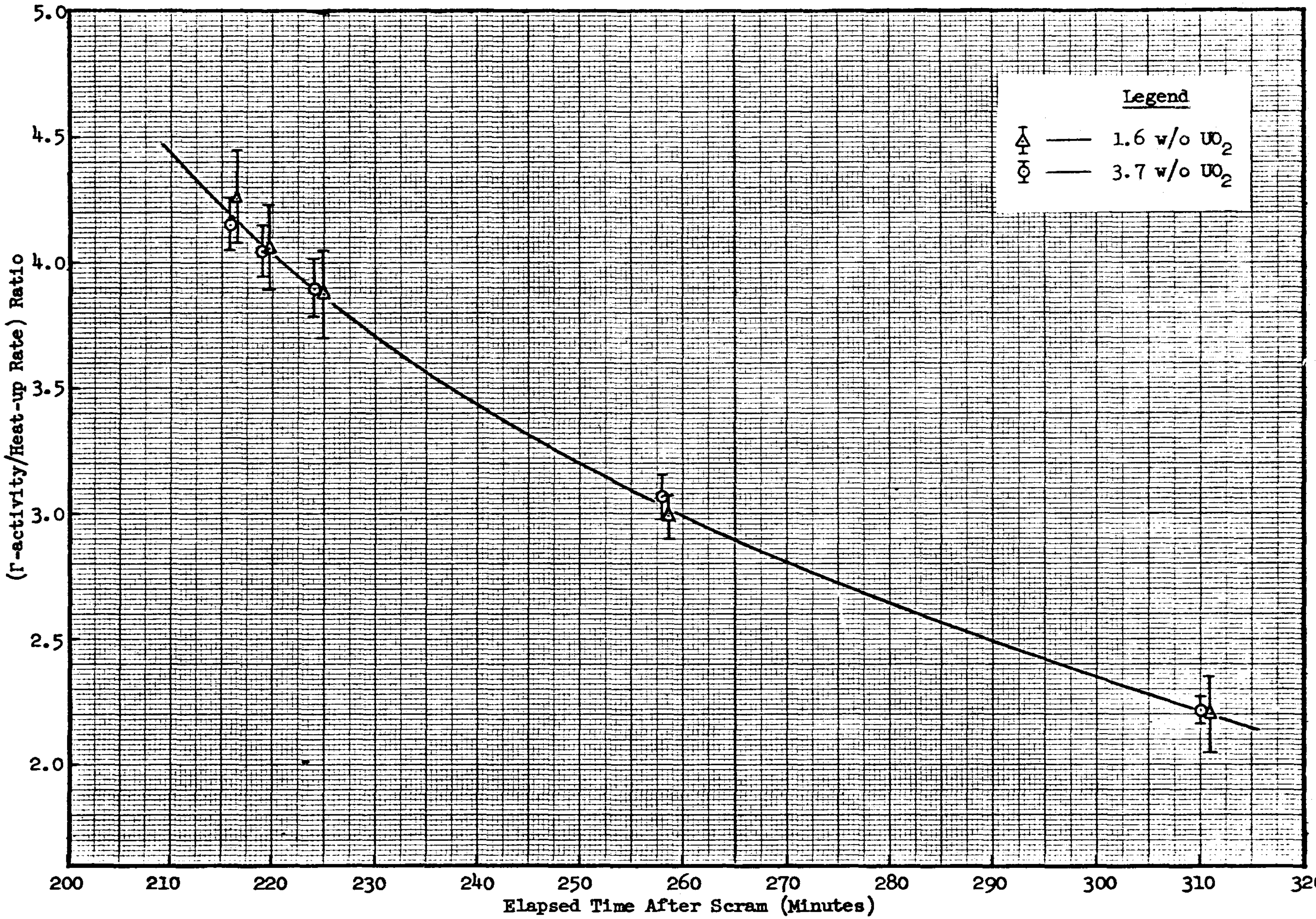

Figure 250.3 Average Ratio of (Γ-activity/Heat-up Rate) for Six Axial Positions on the 1.6 w/o and the 3.7 w/o UO_2 Fuel Rods

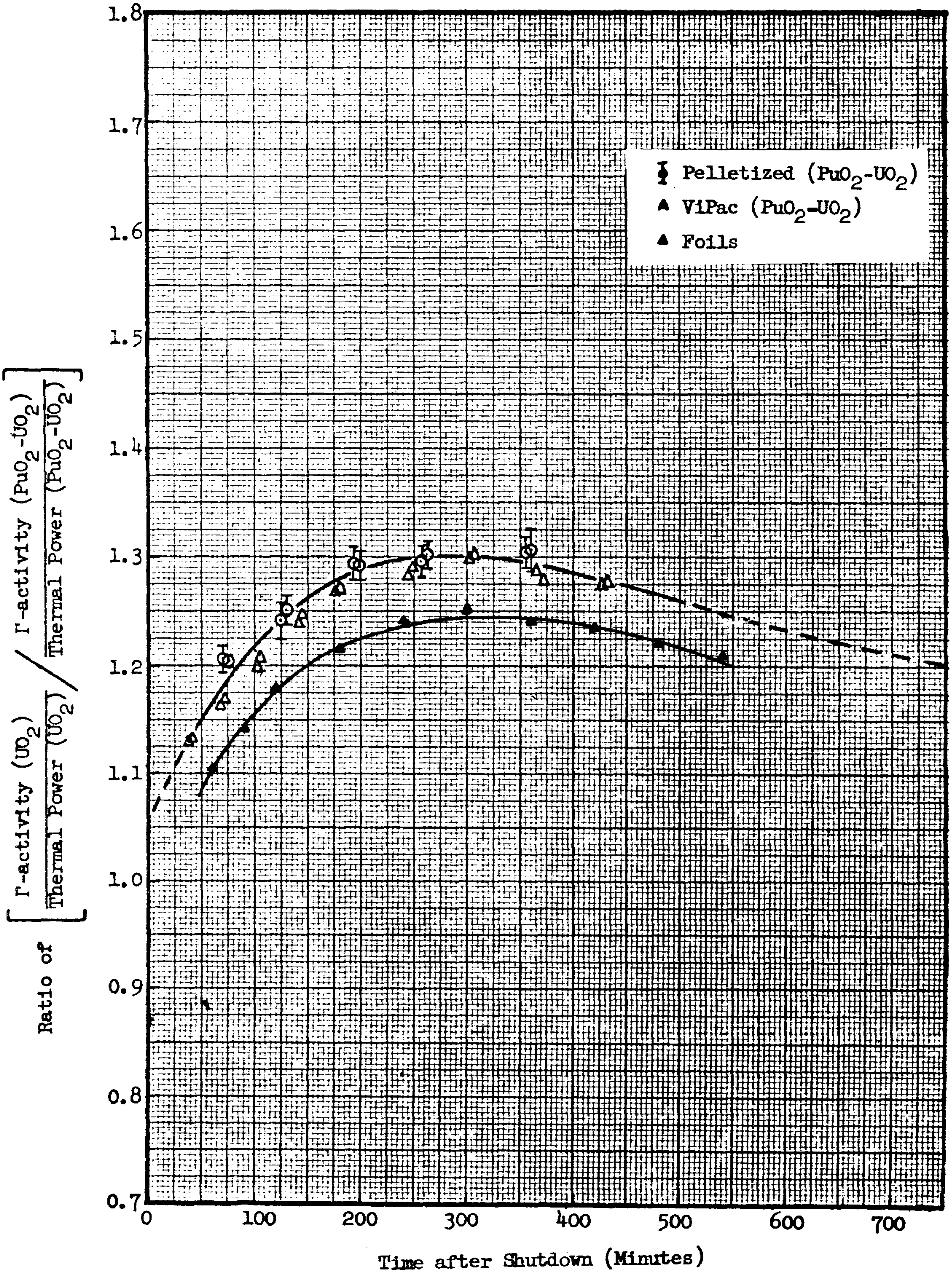

Figure 250.4 P-UO_2 Factors versus Time after Shutdown

The gamma activity of the foils as a function of time after shutdown was determined. Later, the La 140 activity in each foil was measured. Using the yields for each type of fission, the number of fissions occurring in each foil was established. The resulting ratio of gamma activity after shutdown per fission event was corrected for the difference in energy release per fission and a like ratio to that of the heat-rate experiment was developed. This time-dependent ratio based on the foil experiment is also shown in Figure 250.4. It is similar in shape but approximately 5% below that determined by the first method.

Power distributions for a number of the two-region experiments conducted at the WREC were determined from measurements using the relationships shown in Figure 250.4. Figure 250.5 compares the analytic and measured power distributions for a 19 x 19 core containing 121 PuO_2-UO_2 fuel rods (11 x 11) in an inner region. The results show comparatively good agreement is obtained with both methods. From the initial comparisons, it is believed that little adverse effect on performance is introduced from the standpoint of a possible discrepancy in power sharing. The evaluation of other two-region experiments is in progress.

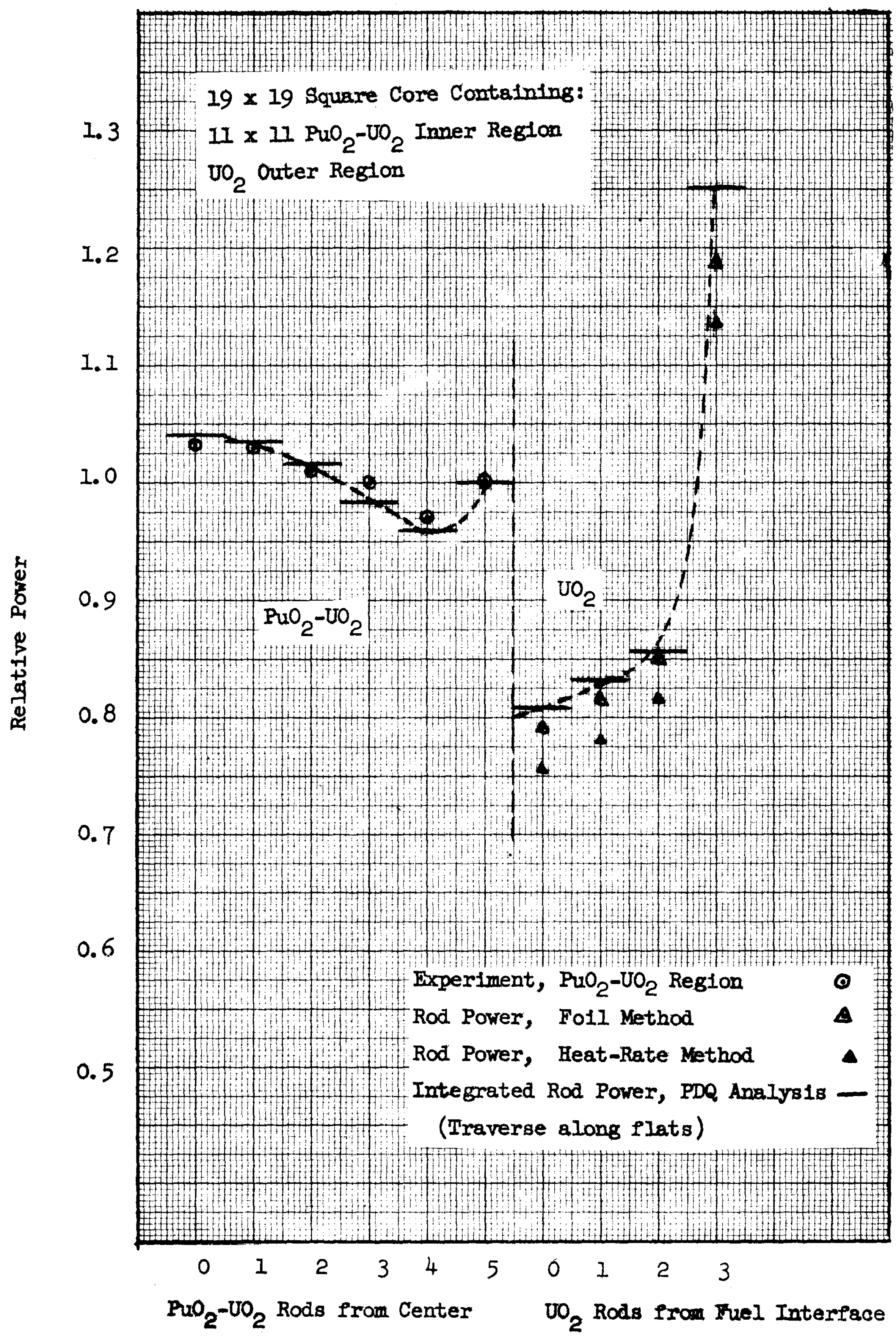

Figure 250.5 Comparison of Calculated and Measured Power Distributions for a Two-Region Assembly

References

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5. C. H. Westcott, "Survey of Nuclear Data for Reactor Calculations," A/Conf. 28/P.717 (1964).

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8. B. R. Leonard, Jr., "Survey of the Status of Low Energy Cross Sections of Fissile Nuclides," HW-69342 (1961).

9. R. Sher and J. Felberbaum, "Least Squares Analysis for 2200 m/sec Parameters of U-233, U-235, and Pu-239," BNL-722, June 1962.

SAXF-310 Fuel Fabrication - Materials

R. J. Allio, A. Biancheria, R. N. Stanutz, M. D. Houston

The objective of this subtask is to procure the required number of PuO_2-UO_2 bearing fuel rods for the program and to assure that manufacturing and quality control procedures meet Westinghouse requirements.

Vibrationally Compacted Fuel

During the period, Battelle Northwest Laboratories repacked their dynapak tie punch and re-densified the Batch A powder. Although a portion of the recycled powder was slightly below the specified particle density, its use was authorized with the proviso that the specified density in the rods must be achieved. The powder was employed to load the remaining required rods. Supplemental chemical analyses by NUMEC indicated that the powders in Batch A and Batch B were within specification.

All the vibrationally compacted fuel rods have been loaded, welded, inspected, shipped and received. Quality control and inspection records are being reviewed to insure proper completion of the contract.

Pelletized Fuel

NUMEC has loaded and welded all of the pelletized rods. Seven rods, which were being held at NUMEC as possible rejects, were examined by

Westinghouse personnel. Six of the rods were accepted for use in the critical experiments only, and one was rejected. The seventh rod is being repaired by NUMEC. These rods and the remaining rods at NUMEC will be shipped as a unit as soon as the rejected rod is repaired. Due to equipment difficulties, NUMEC has not completed all of the contractual chemical analyses required for record purposes. The remaining analyses are expected to be completed during the next report period.

NUMEC has started to reprocess scrap. During the next quarter, all scrap should be reprocessed, returned to the AEC and settlement made for losses.

SAXF-320 Fuel Inspection and Assembly

W. E. Ray, R. Duncan, R. H. Rahiser, M. A. Parker

The objectives of this task are to assist vendors of materials and of fuel rods in inspecting their products to meet specifications, to conduct receiving inspections upon receipt of the fuel rods by Westinghouse and to fabricate and inspect fuel assemblies (exclusive of 9 x 9 enclosures supplied by Westinghouse on a non-reimbursable basis).

During this period the inspection of PuO_2-UO_2 fuel and fuel rods was completed at Battelle Northwest Laboratories and at NUMEC on all rods to be used in the core. Additional autoclave corrosion tests and final inspections remain for about seven rods being accepted conditionally at NUMEC. Review of quality control and inspection records is in progress.

The fabrication and inspection of the 3 x 3 subassembly have been completed. In addition, the related subassembly holding down latch and the guidance tool for thimble insertion were completed.

The use of PuO_2-UO_2 fuel rods and of Core II 5.7% enriched UO_2 fuel rods for critical experiments has been completed. Shipment of rods to the Westinghouse Fuel Manufacturing Plant at Cheswick has been started. Installation of fuel rods into Saxton 9 x 9 enclosures will be completed by the end of July.

SAX-330 New Fuel Shipping

H. E. Walchli, H. W. Keller

Design drawings for the Saxton 9 x 9 PuO_2 fuel assembly shipping containers have been completed. A modified SELNI container will be used to ship the normal 9 x 9 assemblies in a horizontal position. A special drum type container will be used to ship the special 9 x 9 assembly in an inclined position. Fabrication of both containers has been initiated. Completion is scheduled by mid-July.

The license for Shipment of 9 x 9 fuel assemblies from Cheswick has been received.

SAX-340 Safeguards Analysis

R. C. Nichols

The change requests to the Saxton Technical Specifications and Operating License along with a safeguards analysis were submitted to the AEC Division of Reactor Licensing to cover the plutonium fueled 3 x 3 subassembly. The necessary license changes were granted by the AEC and the 3 x 3 is presently operating in a peripheral location.

The Safeguards Report for the partial plutonium core II and the necessary change requests were also submitted to DRL. Information meetings were held with the DRL staff and an ACRS subcommittee. The ACRS subcommittee had not had sufficient time to become familiar with the details of the report and as a result their questions were mostly general in nature and were answered at the meeting. Four areas were covered which the subcommittee stated would probably be covered more fully at the full ACRS Committee meeting. These areas were:

1. How much plutonium might reach the vapor container following the hypothetical accident and failure of core cooling?
2. How well do the critical experiments check with the predicted results?
3. What type of reactivity follow will be conducted?

4. What is the unexplained reactivity limit beyond which the reactor would not be operated?

The DRL staff had covered the report in great detail and as a result had a great many questions. Most of these questions were resolved at the meeting. However, the staff did have eight areas in which they felt additional information was required. These areas were outlined informally at the meeting and received officially at a later date. Answers were prepared and submitted to the AEC as Supplement No. 1 to the Safeguards Report. Supplement No. 1 is included at the end of this section.

It was learned at these meetings that the plutonium core would not be on the May agenda for the ACRS. Subsequent to these meetings, it was learned that the ACRS was not going to consider any cases at the June meeting and that the review of the plutonium core would not be conducted until the July meeting. Efforts on the part of the DRL staff to have a special meeting of the ACRS were not successful.

Work has been initiated to set up criteria for and to determine the maximum reactivity anomaly that could be tolerated in the operation of the Saxton reactor. This work is necessary as a result of the ACRS subcommittee suggestion that the applicant be prepared to provide such a number at the full ACRS meeting.

SUPPLEMENT NO. 1 TO SAFEGUARDS REPORT FOR THE SAXTON REACTOR PARTIAL PLUTONIUM CORE II

Question #1 - In order to provide a basis for evaluating the conservatism of the parameters used in the accident evaluation sections of the report, provide verification that the physics parameters measured in the critical experiment at WREC are at least as conservative as those assumed for the accident evaluations. In addition, verify that the proposed loading will be with a central plutonium region.

Answer: The series of critical experiments outlined in the Safeguards Report for the Partial Plutonium Core II is now in progress at the Westinghouse Reactor Evaluation Center (WREC). Although the entire series is not yet completed, the results obtained to date show that experiment and analysis are in excellent agreement and verify that a conservative approach was followed in the design of the Partial Plutonium Core II. While additional experiments and data processing and reduction are continuing, the program is sufficiently complete to be able to state that:

(a) Any data and results obtained in the future are not expected to significantly alter the above conclusions and

(b) The initial core loading will be with the nine plutonium enriched fuel assemblies in the center of the core.

The preliminary results of the criticals which are available are summarized below. The experimental program and series of criticals being conducted at the WREC are outlined in Table 1-1. Predictions as to the number of fuel rods required for criticality, calculated k_{eff} and corresponding boron concentrations are included in this table. The status of the experimental program of Table 1-1 is shown in the following list:

Configuration	Type	Status
A	UO_2-One Region Clean Core	Completed
1	PuO_2-UO_2-One Region Clean Core	Completed except for 1(e)
2	PuO_2-UO_2, UO_2-Two Region Clean Core PuO_2-UO_2 in Inner Region	Completed except for 2 (a)
3	PuO_2-UO_2-One Region Borated Core	Completed
4	PuO_2-UO_2-UO_2-Two Region Borated Core, PuO_2-UO_2 Inner Region	In Progress
5	Two Region, UO_2 Fuel in Inner Region, Clean and Borated	Clean Core-Completed Borated Core-In Progress
6	PuO_2-UO_2-One Region Clean Core, Larger Pitch	To be done

Reactivity Experiment Results

The results of two critical experiments are available for comparison with predicted results. A major portion of the experiments was done with the same H/Pu ratio that will exist in the Saxton reactor at operating temperature (T_{mod}=530°F).

			Fuel Rods Req'd for Criticality	
Configuration	Fuel	Pitch	PDQ Analysis	Experiment
1(c)	PuO_2-UO_2	0.56 in.	355	343
A(3)	UO_2	0.56 in.	356	346

Using the same cross-section data and calculational methods employed in the core design, experimentally determined values of buckling were used to calculate the effective multiplication factors for various lattices and fuels.

Fuel	Configuration	Lattice Pitch	Calculated k_{eff} LEOPARD (Total Buckling)	Calculated k_{eff} X-Y PDQ (Axial Buckling)
UO_2	A(3)	0.560 in.	1.0042	1.0045
UO_2	A(2)	0.792 in.	0.9997	-

Fuel	Configuration	Lattice Pitch	Corrected k_{eff} LEOPARD (Total Buckling)	Corrected k_{eff} X-Y PDQ (Axial Buckling)
PuO_2-UO_2	1(b)	0.560 in.	0.9950	0.9966
PuO_2-UO_2	1(c)	0.792 in.	1.0063	-

For all of these experiments, the experimental k_{eff} was 1.0. Evaluation of k_{eff} for the PuO_2-UO_2 lattices included an allowance of 0.025 which is based on previous comparisons of analysis by these methods with experimental results of a number of Hanford mixed oxide critical experiments so that [Corrected k_{eff} = Calculated k_{eff} - 0.025].

The value of 0.025 was selected prior to completion of the experiment so that its selection was not influenced by prior knowledge of the experimental results of the buckling measurements. The excellent agreement between the analytical predictions and the experimental results shows that the allowance selected was a reasonable one. No allowance was included in the evaluation of the UO_2 results.

From the standpoint of the Saxton core design, the results of the experiments lead to the following conclusions:

(a) There is no need to modify the expected core lifetime or installed reactivity predictions used in the reference design of the Safeguards Report.

(b) The good agreement between analysis and experiment for a wide range of H/Pu ratios indicates that one of the most important factors of the moderator temperature coefficient, the density effect, is correctly calculated by the analytical methods used in the core design.

Power Peaking Results

Power peaking experiments in fuel rods adjacent to water slots have been carried out in both single region and two region cores. Only the results of the single region cores have been analyzed to date. In the single region experiments, a water slot was formed by removing five center fuel rods from a square lattice. The power level in the adjacent fuel rods was measured with and without the water slot. Experiments were also carried out with an aluminum slab in the water slot to displace some of the water. Using the various lattice characteristics, PDQ-3 analyses to predict the peaking effect have been carried out and are compared with experimental measurements.

Peaking Factor Ratio: Analysis/Experiment

Core	H_2O Slot	H_2O + Al Slot
PuO_2-UO_2	1.0779	1.0400
UO_2	1.0555	1.0104

These results demonstrate that the analytical methods used in the Core II evaluation are conservative in that they over-predict the power peaking effects in water slots. These results are representative of the actual conditions which will be present in Core II as installed in the reactor because the peak in the core occurs within the boundary of the Pu fuel region and is therefore more characteristic of a single region core than peaking at the boundary of a two region core. The results of this analysis demonstrate that the hot channel factors assumed in the core design are conservative and that the initial power level shown in the Core II Safeguards Report may be raised from 21.6 MWt, probably up to 23.5 MWt. Additional testing and low power experiments will determine the actual hot channel factors and initial power level for Core II.

Boron Worth Results

Boron worth measurements were made in the two region core of configuration 4(b). The predicted boron concentration required for a full water height critical was 1525 ppm. The experimental results extrapolated to full water height conditions showed a concentration of 1550 ppm which is in excellent agreement with the prediction.

Kinetic Parameter Results

The kinetic characteristics of single region and two region cores are presently being investigated using pulse neutron techniques. An additional experiment has been completed for a single region Pu core which measured the neutron lifetime by measuring the reactivity change for a small addition of boron ($\sim$25 ppm) to the moderator. Although all of the experiments being conducted to determine the kinetic characteristics are not yet complete, these preliminary comparisons of analyses and experiments are available:

Fuel	Lattice	Prompt Neutron Lifetime, ℓ (μ sec) LEOPARD	PDQ (1/v Poison)	Boron Addition	Pulse Neutron
One Region, PuO_2-UO_2	0.56 in.	8.5	19.4	15.8	20.5 (Calculated from $\beta = 0.0034$ and measured β/ℓ = 166 sec^{-1}
One Region,	0.56 in.	15.0	20.4	-	30.3 (Calculated from $\beta = 0.00795$ and measured β/ℓ = 262 sec^{-1}

As the table shows, the values of ℓ if calculated for the experiment by LEOPARD are much shorter than those inferred from the experiments. This indicates that the actual values of ℓ for Core II will be longer than those predicted by the LEOPARD calculation and reported in the Core II Safeguards Report.

Table 1.1

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, hot.
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 368 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	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/ℓ measurement

Question #2 - It is proposed that some of the PuO_2 fuel in Core II will operate a specific power levels of up to 16 Kw/ft. To enable us to evaluate any significant safety problems associated with operation at this proposed specific power, provide a discussion of the results of such operation involving UO_2 fuel at the Saxton reactor.

Answer: The peak specific power level of 16 Kw/ft is a conservative design limit based upon present Westinghouse fuel element design practice and techniques. This limit is believed to be a reasonable upper boundry for the initial operation of the mixed oxide, partial plutonium core for Saxton. A great deal of experimental data exists on the successful operation of test fuels of these types (sintered pellets and vibration compacted powder) at specific power levels greatly in excess of 16 Kw/ft and even, in some cases, with significant center melting of the fuel.

The limit of 16 Kw/ft is a reasonable step up from the maximum conditions so far experienced in the Saxton core (14.5 - 15 Kw/ft) as less than two dozen rods of Core II would operate above 14.5 Kw/ft if the peak rod were to operate at 16 Kw/ft.

Because the Saxton reactor is an experimental plant, sustained periods of operation at the maximum rated power of 23.5 MWt have not been obtained in the past. The peak specific power in any fuel rod in Saxton is dependent on a great many factors; fuel enrichment, boron concentration, control rod position and reactor power level. Therefore, the peak specific power depends on the condition of the above parameters at the time the measurement is made.

With the reactor above 22 MWt, the maximum specific power level of the core is nominally 13-14 Kw/ft. This number is based on the same methods that would determine the 16 Kw/ft limit, that is, a 10% uncertainty in the measurements and an engineering hot channel factor of 1.045. The highest measured specific power has been 13.87 Kw/ft at a reactor power level of 22.9 MWt. When extrapolated to 23.5 MWt, a maximum of 14.56 Kw/ft is obtained from 12.16 Kw/ft at 19.63 MWt. With the uncertainties involved, it is not possible to say that with the reactor at 23.5 MWt that specific powers in excess of 14.5 Kw/ft have been experienced in the Saxton core. All of the peak values referred to above have occurred in the central 9 x 9 which contains experimental fuel that is licensed to operate up to 16 Kw/ft.

Successful operation of fuel at or above this level has been demonstrated by several Westinghouse experiments. Six capsules containing three fuel rod samples from the CVTR core were irradiated in the Westinghouse Test Reactor to a maximum power rating of 24 Kw/ft.(1) The capsule configuration was a 5-inch column of UO_2 pellets, .430 inches in diameter, 94 $\pm$ 1.5% of theoretical density clad with Zircaloy-2. The capsules were all successfully irradiated with no evidence of central melting.

Two additional capsules were irradiated in the Westinghouse Test Reactor.(2) One capsule contained three fuel rods with a 38-inch fuel length and was irradiated at peak fuel rod power levels of 17 to 19 Kw/ft to a maximum fuel burnup of 3,450 $\frac{MWD}{MTU}$. The other capsule contained four fuel rods with 6-inch fuel length. Average fuel rod power levels of > 18 Kw/ft were maintained during irradiation to 6,250 $\frac{MWD}{MTU}$. The rods contained UO_2 pellets .430 inches in diameter and 94 $\pm$ 1.5% dense. The capsules were clad in Zircaloy-2. The capsules were successfully irradiated and indicated that thermal reactors could be operated at these high rod powers safely and successfully.

UO_2 fuel capsules are being irradiated in the NASA - Plum Brook Reactor as part of the High Power, High-Burnup Irradiation Program.[3] Fuel pins containing 0.3 inch diameter pellets 96% dense with a 6-inch fuel column are clad with 304 stainless steel. The capsules are being irradiated at power ratings of 20 to 60 Kw/ft, to a maximum burnup of 80,000 $\frac{MWD}{MTU}$. Four capsules have been irradiated to 10,000 $\frac{MWD}{MTU}$ at a peak power rating of 39 Kw/ft. Three of these irradiations were completely successful; the fourth failed due to excessive fuel melting. Approximately seventy-five percent of the cross-sectional area of the pellets was molten. The failure occurred after long exposure at high rod power.

Three capsules were irradiated in the Plum Brook Reactor in a program designed to measure the thermal conductivity of UO_2 at the columnar grain growth threshold temperature.[3] The pins were 4-1/2 inches long and 1-1/4 inches in diameter. They were successfully irradiated at rod powers of 20-24 Kw/ft.

Two vibratory compacted pins and one pelleted fuel pin were successfully irradiated in the GETR at peak rod power of 21 Kw/ft.[4] The pins were 5.2 inches long and had an active fuel diameter of .56 inches. The pelleted rod was 88.3% dense while the vipac were 81.8% and 86.7%.

In addition, GE has run some very extensive, long irradiation high power level experiments in the GETR with fuel enriched to $\sim$ 20% in Pu.[5] Two pelletized rods with no central voids were operated at peak specific powers of $\sim$ 15.5 Kw/ft and $\sim$ 17.8 Kw/ft for burnup of 23,100 $\frac{MWD}{MTU}$ and 17,600 $\frac{MWD}{MTU}$ respectively. The experiments were very successful with no adverse effects due to these operating conditions.

Based on the experimental evidence available, the possible operation of some rods in the Pu region of Core II at 16 Kw/ft power levels will present no significant safety problems in the operation of Core II and is a very conservative extrapolation from the power levels already experienced in Saxton.

References

1. Duncan, R. N., "Rabbit Capsule Irradiation of UO_2," CVNA-142, 1962.

2. Duncan, R. N., "CVTR Fuel Capsule Irradiation," CVNA-153, 1962.

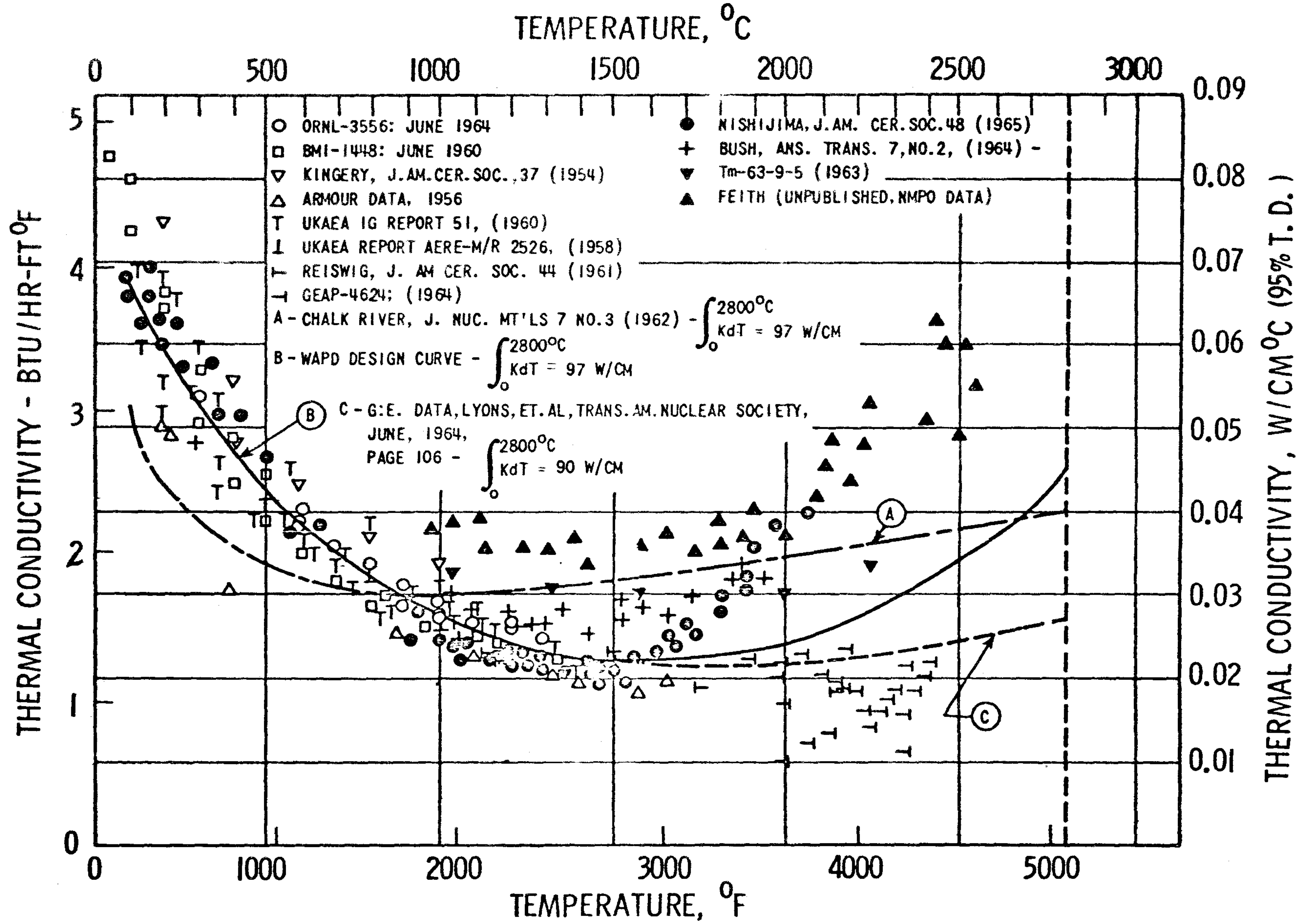
3. WCAP-2500, 2640, 2689, 2732.

4. Balfour, M. G. and Ferrari, H. M., "Irradiation of Vibratory Compacted UO_2 Fuel Elements," WCAP-2729.

5. Gerhart, J. M., "The Post-Irradiation Examination of a PuO_2-UO_2 Fast Reactor Fuel," GEAP-3833, 1961.

Question #3 - We understand that new information concerning the conductivity of uranium dioxide at high temperatures is available. Provide a curve of uranium dioxide conductivity as a function of temperature on which these new data points are included.

Answer: The attached figure is to replace Figure III-7 of the Core II Safeguards Report.



THERMAL CONDUCTIVITY OF URANIUM DIOXIDE

FIGURE III - 7

Revision 1

Question #4 - The Saxton reactor is the first licensed nuclear power reactor in which a plutonium core loading is to be used. To enable us to evaluate a possible manner in which plutonium might be released to the environs, provide a discussion of those operating procedures which will assure that plutonium which may be in the containment building as contamination will not be transported to the remainder of the site or to the environs.

In addition, discuss why the limits of sensitivity of the various monitoring equipment and health physics procedures proposed are adequate to assure that 10 CFR 20 limits for plutonium will not be exceeded.

Answer: Because of the conservative assumptions and methods used in the plutonium fuel design and the rigorous testing and inspection performed on the fuel during its manufacture, the probability of fuel clad failure throughout the planned life of Core II is very small. In addition, the fuel rods and the fuel assemblies are monitored for alpha contamination prior to shipment to Saxton so that there is little likelihood that tramp plutonium will cause a contamination problem during fuel storage and loading.

In the event some plutonium contamination should be present inside the containment, there are only three methods available for transporting plutonium contamination from the containment building:

(a) Personnel

Saxton's present radiation protection procedures have proven adequate to prevent the spread of contamination from the containment vessel. Access to the containment vessel is allowed only under the provisions stipulated by a radiation work permit which specifies, among other things, protective clothing to be worn. Step-off pads and storage for protective clothing are provided in the air lock. Monitoring of personnel for alpha contamination prior to leaving the vessel will be accomplished as required in the radiation work permit.

(b) Ventilation Exhaust

Since the containment vessel has no exhaust flow during reactor operation, the installed alpha monitoring system which will be added to the present containment air activity monitors will give a reliable history of containment vessel air activity. At a time when entry is desired, the reactor will be shut down and the containment vessel air activity will be known. Ventilation exhaust flow rate will be adjusted, if necessary, to insure that any release to the atmosphere is within the limits established by 10 CFR 20. It is expected that the containment vessel air activity attributable to plutonium will be below its MPC at all times and that it will not be necessary to regulate the containment vessel air release rate.

(c) Liquid Effluents

Liquid effluents from the containment vessel will be handled without any change to the present waste disposal or chemistry sampling system. The only procedural change will be an increased monitoring of areas for alpha contamination. Present procedures for monitoring effluents are adequate to assure that 10 CFR 20 limits for plutonium will not be exceeded.

After discussions between Saxton personnel and personnel at the Plutonium Recycle Test Reactor, we have concluded that the problems associated with radiation protection due to plutonium are no different from those which already exist, due to the presently installed uranium fuel. As quoted from U. S. Atomic Energy Commission Research and Development Report HW-83601, PROGRESS IN PLUTONIUM UTILIZATION by Hanford Laboratories:

> "Plutonium fuels have been stored and handled in the same manner as uranium fuel, and irradiated fuels have been routinely handled for special examinations and core changes without difficulty. No unusual procedural controls have been made necessary, nor has any specialized operator training been required specifically as a result of using plutonium fuels in the PRTR.
>
> "The PRTR experience has shown that the effects of plutonium fuel failures are no different than those for uranium fuels. Emissions have been virtually limited to fission gases with no evidence of particulate washout. Alpha contamination, usually of primary concern in fabricating plutonium fuels, is of little concern in reactor operation, as gamma contamination governs procedures for almost all maintenance work."

The activity concentration requirement of 10 CFR 20 for Pu-239, Pu-240 and Pu-241 for radiation workers exposed for 40 hours per week, is a maximum airborne concentration of 2.0×10^{-12} μc/cc. This activity level, defined as the radioactivity concentration guide for a 40 hour week (RCG/40), represents that concentration of plutonium in air to which a "standard man" may be exposed for 40 hours per week, 50 weeks per year for a total period of 50 years so that at the end of 50 years the total activity fixed in the "standard man's" body will not exceed the recommended maximum permissible body burden (MPBB) of 0.04 μc of plutonium.

This MPBB as set by both the International Commission on Radiological Protection and the National Commission on Radiological Protection is defined as that amount of material which may be maintained indefinitely in the body of a "standard man" without producing any significant somatic or genetic effects throughout the life of the "standard man".

The sensitivities of the air particulate monitors, both the moving filter vapor container monitor and the fixed filter portable monitors, have been revised slightly from those given in the Core II Safeguards Report. The minimum sensitivity for these instruments for a 1-hour sample period and following a delay period (about 6 hours) to remove the Radon-Thoron background is given as 2.5×10^{-12} μc/cc. As stated before, containment access is not possible during power so that detection of this level of activity is more than adequate to assure that containment vessel purge prior to entry will not produce off-site plutonium levels above 10 CFR 20 levels. Containment vessel purge procedures can be altered, if required, if the containment vessel concentration is significantly above the limit of detection. Purge of the containment vessel prior to entry will assure adequate working conditions upon entry.

The portable air particulate monitors can be moved throughout various areas of the plant as required to sample for airborne activity. Alpha monitoring during such operations as main coolant sampling in the sample room or analysis work in the radiochemical laboratory is provided by these instruments. These instruments are capable of detecting near RPG/40 levels with the 6-hour delay for Radon-Thoron decay. A more rapid readout of higher concentrations may also be obtained. Following a one-hour sampling time and the presence of a high Radon-Thoron background (600 cpm) the minimum sensitivity is about 2×10^{-10} μc/cc which is a factor of 100 above RPG/40. If this high plutonium concentration were detected, work in the area could be suspended and corrective action initiated. Workers exposed to these higher than RPG/40 concentrations could be restricted from working in possibly contaminated areas for a period of time to allow averaging of this exposure. For example, a one-hour exposure to 100 x RPG/40 concentration is equivalent to about 2-1/2 working weeks at RPG/40 so that return to work with RPG/40 concentrations would be permissible after 2-1/2 weeks of no exposure to plutonium.

Higher concentrations of plutonium can be detected in even shorter periods of time due to the fact that the count rate of the sample, due to Pu, increases linearly with exposure time and is proportional to the concentration. For a high Radon-Thoron background of 240 cpm a plutonium concentration of 1×10^{-9} μc/cc can be detected after a five minute sample time. Exposure to 1×10^{-9} μc/cc or 500 x RPB/40 for five minutes is almost equal to a 40-hour exposure to RPG/40, so that one week of non-exposure to plutonium would then allow return to work in RPG/40 levels.

The procedure of curtailing work following exposure to levels above RPG/40 is a standard practice and where combined with the instrument sensitivities described will assure that personnel exposures are well within the limits of 10 CFR 20.

Question #5 - In the accident analysis section of the report it is stated that each accident was analyzed using that combination of system parameters which would give the most serious consequences. Indicate the manner in which it can be assured that the most adverse combination of parameters has been selected, and provide the range of parameters considered for each accident analysis.

Answer: Two basic premises which underly accident and reactor transient analyses are to develop realistic yet conservative models and then to apply these models using realistic yet conservative parameters. Analog computers are normally used to simulate the reactor. The selection of the basic parameters depends on the transient being studied. The parameters are chosen on the basis of adding the most reactivity to the transient or providing the least help in limiting or preventing the transient.

As a specific example, the detailed reasoning for the choice of parameters of the control rod withdrawal at power accident are outlined below.

During this transient, heating of the fuel and the moderator will add negative reactivity to the systems and tend to depress the transient. For this reason, the moderator coefficient assumed was smaller than the expected value and would correspond to a boron concentration in excess of 2000 ppm. The Doppler coefficient chosen was less than expected values.

Overpower scram initiation is set to trip at 115% of nominal full power and is a redundant circuit to assure reliability. However, errors in fixing set points and in power measurements are assumed to delay scram initiation until a power level of 122% is reached.

Upon initiation of scram, an instrumentation delay of 0.5 sec. is assumed to delay rod motion. Actual instrumentation delay times are less than 0.3 seconds. A further delay in scram of 0.6 seconds is assumed for control rod motion in a region of small effectiveness and 0.9 seconds is assumed for completion of the rod insertion into the core. Actual measured control rod drop times for Saxton are on the order of 0.9 seconds or less so the actual scram completion time will be about 1.2 seconds or less compared to the 2.0 seconds assumed in the analysis.

Control rod scram worth upon insertion was assumed as 0.02 $\Delta k/k$. The nominal operating conditions of this accident, that is early in life with large hot channel factors and high boron concentrations (1500-2000 ppm), will result in about 0.15-0.18 $\Delta k/k$ reactivity in control rods out of the core. Even if the most reactive rod (0.05 $\Delta k/k$) were to stick, the reactivity insertion by control rods would be about 0.10 $\Delta k/k$. The only time that a reactivity insertion on the order of 0.02 $\Delta k/k$ would be possible would be very early in core life at very low boron concentrations (rodded control) which is a condition not compatible with the moderator coefficient chosen for the analysis.

A final conservative assumption is in the reactivity insertion rate of the control rods during withdrawal. The maximum insertion rate of the most reactive rod group (the two inner rods or the four outer rods) is 7.25×10^{-5} $\Delta k/k$/sec. and assumes the control rods to be in the most reactive region and moving at the maximum withdrawal speed. The value of 2.5×10^{-4} $\Delta k/k$/sec. which was assumed for this analysis is a much larger rate than could possibly be experienced by the reactor during this transient.

The same general reasoning has been applied to the other transients and accidents analyzed. The following tables present a comparison of the parameters assumed for the analyses and those which might be expected to exist in the reactor.

I. Rod Withdrawal, Cold Startup

		Value Used	Expected Value
1.	Moderator Temperature Coefficient (at 70°F, 2000 ppm boron)	$+ 0.3 \times 10^{-4}$ Δk/k/°F	0.0 Δk/k/°F
2.	Doppler Coefficient	$- 1.1 \times 10^{-5}$ Δk/k/°F	$- 2.0 \times 10^{-5}$ Δk/k/°F
3.	Reactor Subcritical by	0.02 Δk/k	> .05 Δk
4.	Overpower Scram Initiation	122%	115%
5.	Control Rod Drop Time	1.5 sec.	< 0.9 sec.
6.	Scram Reactivity Insertion by Rods	0.02 Δk/k	0.1 - 0.15 Δk/k
7.	Reactivity Insertion Rate	2.5×10^{-4} Δk/k/sec.	$< 7.25 \times 10^{-5}$ Δk/k/sec.

II. Rod Withdrawal, Hot Startup

		Value Used	Expected Value
1.	Moderator Temperature Coefficient (at 530°F, 2000 ppm boron)	$- 2.7 \times 10^{-4}$ Δk/k/°F	$- 3.0 \times 10^{-4}$ Δk/k/°F
2.	Doppler Coefficient	$- 1.0 \times 10^{-5}$ Δk/k/°F	$- 1.3 \times 10^{-5}$ Δk/k/°F

3. Thru 7. - Same as for Case I

III. Rod Withdrawal, At Power

		Value Used	Expected Value
1.	Moderator Temperature Coefficient	$- 2.7 \times 10^{-4}$ Δk/k/°F	$- 3.0 \times 10^{-4}$ Δk/k/°F
2.	Doppler Coefficient	$- 1.0 \times 10^{-5}$ Δk/k/°F	$- 1.1 \times 10^{-5}$ Δk/k/°F
3.	Primary Coolant Pressure (For DNB Calculations) { ΔH-DNB	2050 psi	2000 psi
	Q-DNB	1950 psi	

III. Rod Withdrawal, At Power (Cont'd)

		Value Used	Expected Value-
4.	Instrument Delay Time Control Rod Drop Time	0.5 sec. 1.5 sec.	< 0.3 sec. < 0.9 sec.
5.	Reactor Power Level, % of Nominal	103%	95-100%
6.	Overpower Scram Initiation	122%	115%
7.	Scram Reactivity Insertion by Rods	0.02 Δk/k	0.10-0.15 Δk/k
8.	Maximum Specific Power	16.5 Kw/ft	14-15 Kw/ft

IV. Steam Break

1.	Moderator Temperature Coefficient (Worst Case, End of Life - 0 ppm Boron Concentration)	$- 4.1 \times 10^{-4}$ Δk/k/°F	$- 4.0 \times 10^{-4}$ Δk/k/°F
2.	Safety Injection Functions	No	Yes

V. Loss of Flow Accident

1.	Moderator Temperature Coefficient	$- 2.7 \times 10^{-4}$ Δk/k/°F	$- 3.0 \times 10^{-4}$ Δk/k/°F
2.	Control Rod Drop Time	1.5 sec.	< 0.9 sec.
3.	Reactor Power Level - % of Nominal	103%	95-100%
4.	Scram Reactivity Insertion of Rods	0.02 Δk/k	0.10-0.15 Δk/k
5.	Maximum Fuel Power Density	16.5 Kw/ft	14-15 Kw/ft
6.	Primary Coolant Pressure (For DNB Calculations) { ΔH-DNB Q-DNB	2050 psi 1950 psi	2000 psi

Question #6 - In the report it is stated that the results of the chemical shim experiment program have demonstrated that a boron release accident as originally postulated is not credible and, accordingly, the requirements of an unexplained reactivity limit are no longer required. Provide a description of the results of the chemical shim work at Saxton so that we may evaluate the safety considerations of deleting this requirement.

Answer: To answer this question, copies of WCAP-2599, "The Saxton Chemical Shim Experiment," are submitted herewith.

Question #7 - Provide an estimate of the amount of plutonium that might be released to the containment in the event of the "maximum hypothetical accident" to enable a more definitive evaluation of the consequences of this accident. In addition, provide an evaluation of the amount of plutonium that might subsequently reach the environs.

Answer: A conservative evaluation of the amount of plutonium oxide in the containment vessel following the maximum hypothetical accident has been completed. The maximum amount of PuO_2 that could be in the containment vessel would be less than 50 mg and maximum amount available for leakage in the form of an aerosol would be less than 35 mg. These amounts would result in a maximum two hour inhalation exposure at the site boundary of less than 10^{-8} of the permissible body burden for plutonium.

Evaluation of the maximum hypothetical accident for the Saxton reactor partial plutonium Core II considered a condition in which the emergency systems to provide core cooling did not function following a loss-of-coolant accident. For such a situation, decay heat generated in the core will result in extensive melting of the clad and internal supports and will eventually cause the core to collapse into the bottom of the reactor vessel. This situation will expose a large amount of fuel surface to the atmosphere in the reactor vessel and the high temperatures involved will cause volatilization of the fuel.

The amount of fuel which can be volatilized under these circumstances will be severely limited because of the geometry of the system, the presence of an air atmosphere and the fact that the fuel may be partially wetted by the molten clad or even partly submerged in a pool of molten cladding and structures.

As shown in Figure 7-1, experimental evidence[1,2] indicates that the vapor pressures of plutonium dioxide and uranium dioxide follow the same curve as a function of the reciprocal of the absolute temperature as measured in a vacuum. Also shown on this figure are the experimental data[3] for the vapor pressure of PuO_2 in an air atmosphere. As would be expected, the presence of an air atmosphere reduces the vapor pressure below that measured in a vacuum. For this calculation, it will be assumed that PuO_2 and UO_2 have the same vapor pressure - temperature relationship in an air atmosphere.

An empirical relationship has been developed which correlates the weight loss rate, vapor pressure, absolute temperature and molecular weight for a system vaporizing a substance in an insulated crucible with a small opening. The relationship is as follows:[4]

$$P_{(atm)} = 6.267 \times 10^{-9} \quad \mu/Ka \sqrt{\frac{T}{M}} \tag{1}$$

P = partial pressure of the effusing species, atm

μ = weight loss rate - mg/hr

K = Klausing factor [$K = 1/(1 + 0.5\ L/R)$]

a = effective orifice area - cm^2 (730 cm^2)

T = absolute temperature - °K

M = molecular weight

L = orifice length (assumed as 1 ft.)

R = orifice radius (1/2 ft.)

The Klausing factor is applied because the actual orifice has some finite physical dimensions while the correlation was developed for an ideal orifice. The molecular weight of the fuel will be taken as an average of 271. Using these constants, Eq. (1) becomes:

$$\mu = \frac{P}{\sqrt{T}} \times 9.59 \times 10^{11} \text{ mg/hr} \tag{2}$$

If an average temperature of $\sim$ 2400°K is assumed for the core material which is slowly heating throughout the meltdown, the corresponding pressure is $\sim 10^{-6}$ atm. The weight loss rate is then:

$$\mu = \frac{10^{-6}}{2400} \times 9.59 \times 10^{11} \text{ mg/hr}$$

$$\mu = 1.96 \times 10^{4} \text{ mg/hr}$$

$$\mu = 19.6 \text{ g/hr}$$

The PuO_2 in the core is 6.6 w/o of the central nine assemblies. As there are 21 assemblies in the core, the average PuO_2 w/o is $6.6 \times \frac{9}{21} = 2.5$ w/o. If it is assumed that the volatilized material has the same weight fraction of PuO_2, then the

$$\mu_{PuO_2} = 0.49 \text{ gm/hr.}$$

The value of $\mu_{PuO_2} = 0.49$ gm/hr would be the limiting value if the entire reactor vessel were at the temperature assumed for the hot fuel as was the case in the experiments of Reference (4). Most of the reactor vessel will be at temperatures considerably lower (500-600°F) than the 4000°F used for the average of the fuel mixture. Because of this situation, a great deal of the vaporized fuel material will not leave the reactor vessel but will plate-out on the relatively cold internal surfaces of the vessel.

The surface area of the inside of the reactor vessel which might be available for plate-out is estimated at about 2.5×10^5 cm^2. The cross-sectional area of the main coolant pipe is about 730 cm^2 so that the ratio is about 3×10^{-3}. Therefore, a conservative estimate of the rate at which the vaporized plutonium oxide leaves the break would be 1% (three times the area ratio) of the rate calculated by Equation (2). The rate at which PuO_2 leaves the vessel is therefore 4.9 mg/hr. In the unlikely event that a condition of no core cooling were to occur, it is not expected that it would exist for more than a few hours so that the total amount of PuO_2 release to the containment vessel would be less than 50 mg.

Because of the large amount of relatively cold surface available in the containment vessel for plate-out of the volatilized material, it is not expected that there will be any significant airborne concentration of PuO_2 which might cause an inhalation hazard. As an upper limit on the evaluation, it will be assumed that all of the PuO_2 leaving the reactor vessel is of the proper particle size to remain in the containment atmosphere as an aerosol.

Studies(5) on the reduction rate of the mass concentration of aerosols indicates that a half life of 4-5 hours is typical. Assuming a half life of 5 hours, an equilibrium state for the amount of PuO_2 in aerosol form is soon reached. The equilibrium amount is calculated as follows:

$$\frac{dN}{dt} = R - \lambda N \quad (3)$$

N = amount of PuO_2 in the aerosol, mg

R = release rate - mg /hr

λ = decay constant - hr^{-1}

Solution of equation (3) yields the familiar result

$$N_{(t)} = \frac{R}{\lambda} [1 - e^{-\lambda t}] + N_{(o)} e^{-\lambda t} \quad (4)$$

At equilibrium with $N_{(o)} = 0$

$$N_{eq} = \frac{R}{\lambda} \quad (5)$$

As shown before R = 4.9 mg/hr

$$\lambda = \frac{.693}{5}$$

$$N_{eq} = \frac{4.9 \times 5}{0.693} = 35.4 \text{ mg}$$

An equilibrium amount of 35.4 mg PuO_2 gives a total weight of Pu of 31.2 mg. Assuming that the Pu aerosol has the same isotopic concentrations as were present in the fuel, we have 2.7 mg of Pu-240 and 28.5 mg Pu-239. These weights give activities of 0.6 x 10^{-3} curies of Pu-240 and 1.77 x 10^{-3} curies of Pu-239 or a total of 2.37 x 10^{-3} curies of Pu.

The original off site inhalation hazards for the Saxton maximum hypothetical accident have resulted in a Technical Specification containment leak rate limit of 0.4% of the contained volume per day. This leak rate is based on a design pressure of 30 psig existing throughout the accident. The design pressure was based on the total energy release of the reactor coolant at saturated water conditions and 2000 psi. The actual energy content of the reactor coolant is considerably less than that assumed previously. Also, Figure 506.1 in the Final Hazards Report for Saxton indicates that the containment pressure will drop very rapidly from the initial peak.

Using the generalized Gaussian dispersions equation(6) for a ground level point source and assuming Pasquill type "F" conditions with a wind speed of 1 meter per second, a dispersion factor $\frac{\chi \bar{u}}{Q} = 6 \times 10^{-3}\ m^{-2}$ is obtained at the exclusion radius of 300 meters. Additional credit(6) can be taken because of dispersion and dilution in the wake of the containment building so that $\frac{\chi \bar{u}}{Q}$ becomes:

$$\frac{\chi \bar{u}}{Q} = \frac{1}{\pi \sigma_y \sigma_z + cA} \qquad (\pi \sigma_y \sigma_z = 167\ m^2)$$

cA = building dilution factor

A = building cross section = 250 m^2

c = factor ranging from 0.5 to 2 depending on the building, assumed as 0.5

Therefore:

$$\frac{\chi \bar{u}}{Q} = 3.42 \times 10^{-3}\ m^{-2}$$

Fall-out of particles as the plume travels will also provide additional reduction of the plume concentration. This reduction factor can be estimated for this case using the method proposed by Chamberlain.(7)

The deposition reduction factor (DRF) is given by

$$(DRF) = \exp\left(- \frac{2}{\pi} \frac{v_g}{\bar{u}} \int_0^x \frac{1}{\sigma_z} \ell^{-\frac{1}{2}\left(\frac{h}{\sigma_z}\right)^2} dx\right)$$

For this case the release height, h = 0

$$(DRF) = \exp\left(- \frac{2}{\pi} \frac{v_g}{\bar{u}} \int_0^x \frac{1}{\sigma_z} dp\right)$$

For Pasquill "F" conditions $\int_0^{300} \frac{1}{\sigma_z} dx$ is about 300. Data from Stewart[8] indicates that the deposition velocity, v_g, for plutonium oxide with particle sizes to be expected in the aerosol size range is in the range of 3-5 cm/sec. If a value of v_g = 4 cm/sec is chosen then:

$$DRF = \exp\left(- \frac{2}{\pi} \frac{4}{100} \times 300\right)$$

$$DRF = \exp(- 7.64)$$

$$DRF = 4.92 \times 10^{-4}$$

The plume concentration of Pu at the site boundary is then given by:

$$\chi_{300} = Q \times 3.42 \times 10^{-3} \times 4.92 \times 10^{-4}$$

$$Q = 2.37 \times 10^{-3} \times \frac{4 \times 10^{-3}}{24 \times 3600} = 1.1 \times 10^{-10} \text{ c/sec}$$

$$Q = 1.1 \times 10^{-4} \ \mu\text{c/sec}$$

$$\chi_{300} = 1.85 \times 10^{-10} \ \mu\text{c/m}^3$$

If an active adult breathing rate of 1.25 m^3/hr and an uptake retention factor of .25[9] are assumed, the two hour uptake of Pu is:

$$D_{Pu} = 1.85 \times 10^{-10} \times 2 \times 1.25 \times .25 = 1.16 \times 10^{-10}\ \mu c$$

The maximum permissible body burden of Pu is 0.04 μc[10] so the accident uptake is 2.9×10^{-9} below the permissible body burden. Because of the large deposition fraction within the exclusion radius, there will be no significant plutonium released beyond the site boundary.

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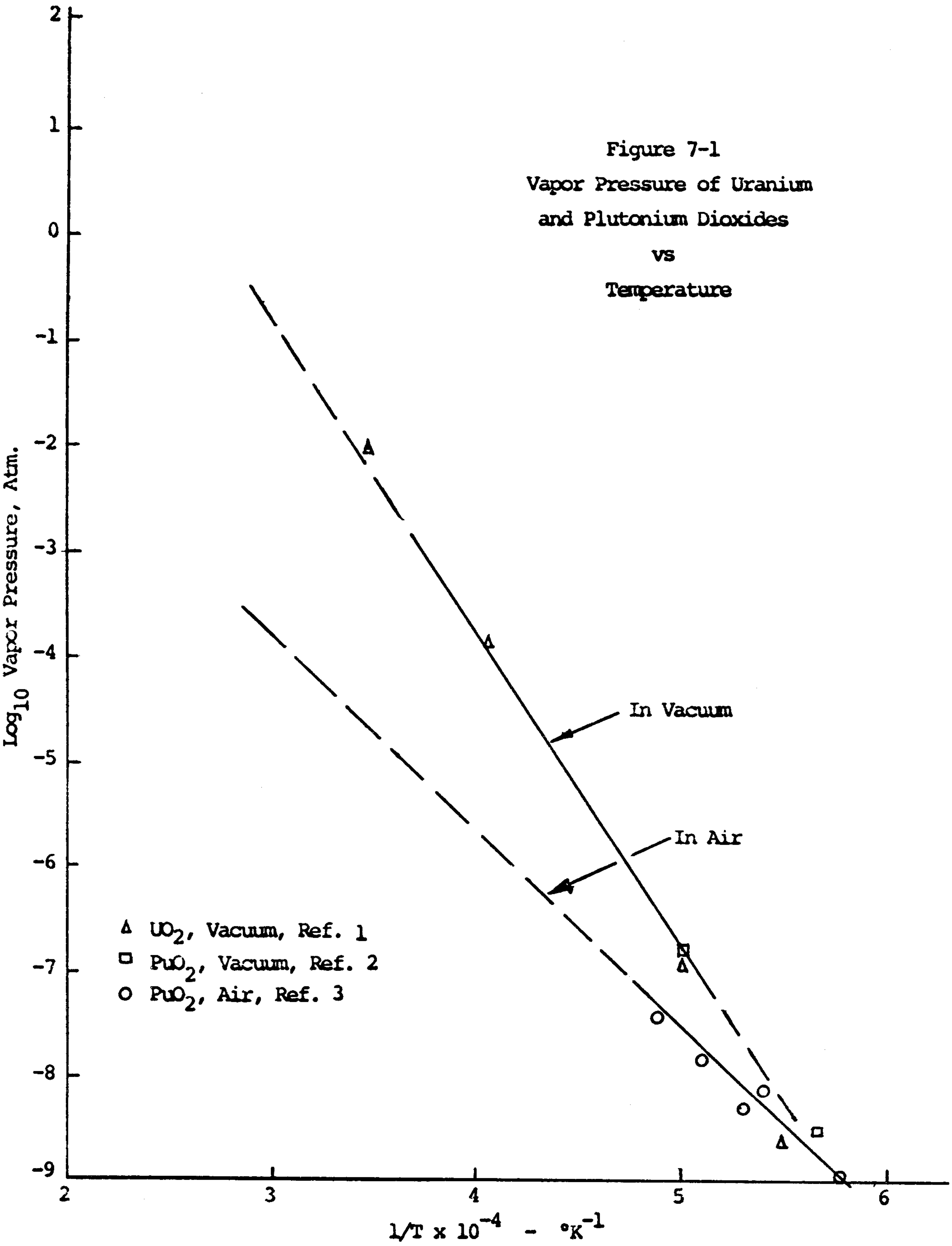

Figure 7-1
Vapor Pressure of Uranium
and Plutonium Dioxides
vs
Temperature
In Vacuum
In Air
Δ UO2, Vacuum, Ref. 1
□ PuO2, Vacuum, Ref. 2
○ PuO2, Air, Ref. 3
Log10 Vapor Pressure, Atm.
1/T x 10-4 - °K-1
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
2
3
4
5
6

Question #8 - Since plutonium requires somewhat more stringent consideration of the reactivity requirements for fuel storage than uranium, provide an evaluation of the adequacy of the Saxton fuel storage facilities for plutonium fuel.

Answer: Evaluation of the adequacy of the Saxton fuel storage facilities for the plutonium enriched fuel were carried out using PDQ-3 calculations to determine the subcritical multiplication factors of the UO_2 and PuO_2-UO_2 fuel assemblies when installed in the fuel storage racks.

The physical dimensions of the fuel storage racks consist of a 3.2-inch surface-to-surface fuel element separation in each row and a 12-inch separation between rows. Ambient water temperature conditions with 0 ppm of boron were assumed for the calculation although the fuel storage water is actually borated. The results of the calculations are shown below:

Fuel	Calculated k_{eff}
UO_2	0.838
PuO_2-UO_2	0.898

The calculated k_{eff} for the PuO_2-UO_2 fuel includes a correction to account for the discrepancy between the experimental results of the WREC criticals and the predicted analytical results. From the data in the above table, it is concluded that there will be no criticality problems or hazards in storing either type of fuel assembly at Saxton.

SAX-350 Alpha Protection

J. W. Power

The alpha protection system design has been completed and all equipment has been ordered. Delivery is being expedited for receipt of the last items by August 15th.

The equipment being supplied is:

Item No.	Description and Use of Item
1	One (1) each Stationary, Continuous, Moving Filter, Alpha Scintillation Vapor Container Air Particulate Monitor - (Channel RIC-11-P)

Consisting of:

a. One (1) each MA-1B* Filter Tape Transport Mechanism

b. One (1) each MD-3B Alpha Scintillation Detector

c. One (1) each RM-20BS(V) Transistor Log Ratemeter (with Spectrometer Dual Meter-Relay)

d. One (1) each RM-30 Blank Plug-in Panel

e. One (1) each RM-40B High Voltage & Ratemeter Power Supply

f. One (1) each MX-14C Pumping System

g. One (1) each MX-15A Purge System

h. One (1) each MX-1A Air Flow Alarm

i. One (1) each MX-2A Filter Feed Alarm

j. One (1) each MX-9A Alpha Check Source

k. One (1) each MX-19A Remote Control Panel

*All Model Nos. Tracerlab Identification

Item No.	Description and Use of Item
	l. One hundred (100) ft. A218701 Signal-Control Cable
	m. One (1) each FP-1 Filter Paper Rolls
	n. One (1) each Motor Starter
	o. One (1) each High Quality Piping & Connectors
2, 3	Two (2) each Portable, Continuous, Fixed Filter, Alpha Scintillation
	Radio-Chemistry Lab / Waste Disposal Bldg. — Air Particulate Monitor — Channel-RIC-21-P / Channel-RIC-24-P
	Each consisting of:
	a. One (1) each AIM-3** Detector Assembly (with stand pump)
	b. One (1) each RC-2 Alpha Scintillation Detector
	c. One (1) each Regulated Air Flow Meter
	d. One (1) each Flash Alarm Lite
	e. One (1) each Audio Alarm Bell
	f. One (1) each Elapsed Timer Meter
	g. One hundred (100) each HV-70 Filter Paper (2" Dia.) Disks
	h. One (1) each Wall Mounting Bracket
	i. One (1) each Remote Sampling Adapter
	j. One (1) each SD-1 Alpha Check Source

** All Model Nos. Eberline Identification

Item No.	Description and Use of Item
4, 5	Two (2) each Stationary, Continuous, Fixed Filter, Alpha Scintillation

Sampling Room / Charging Pump Room — Air Particulate Monitor — Channel RIC-22-P / Channel RIC-23-P

Each consisting of:

a. One (1) each MA-5B* Fixed Filter Sampling Assembly

b. One (1) each MD-3B Alpha Scintillation Detector

c. One (1) each MM-6B Transistor Log Ratemeter (with high voltage power supply dual contact meter relay)

d. One (1) each CX-1 Bench Cabinet

e. One hundred (100) each FP-5 Filter Paper (1-3/4") Disks

Item No.	Description and Use of Item
6, 7	Two (2) each Portable Intermitent Battery-Operated, Alpha-Scintillation Monitor

Gamma-G.M.

General Plant - Surface Contamination Monitors — Channel RIZ-73-P / Channel RIA-74-P

Each consisting of:

a. One (1) each PAC-1SAGA** Count-Rate Meter

b. One (1) each AC-3 Alpha Probe

c. One (1) each RASP-1 Alpha Probe

d. One (1) each PG-1 Gamma Probe

e. One (1) each SPA-1 Alpha Probe

f. One (1) each SK-1 Count-Rate Speaker

g. One (1) each AC-3F Spare Face Plate

h. One (1) each CS-1 Alpha Check Source

i. One (1) each SC-2 Spare Scintillation Crystal

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*All Model Nos. Tracerlab Identification

**All Model Nos. Eberline Identification

Item No.	Description and Use of Item
8, 9	Two (2) each Stationary, Continuous, Alpha Scintillation

Vapor Container Entrance Station / Laundry Room — Surface Contamination Monitors — Channel RIA-63-P / Channel RIA-64-P

Each consisting of:

a. One (1) each RM-3A** Count-Rate Meter

b. One (1) each AC-3A Alpha Probe

c. One (1) each CS-1 Alpha Check Source

d. One (1) each AC-3F Spare Face Plate

e. One (1) each AC-2 Spare Scintillation Crystal

10 One (1) each Alpha Instrumentation Calibration Sources for Items #4, 5, 6, and 7.

Consisting of:

a. Four (1) each S-94A Alpha Sources

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*All Model Nos. Eberline Identification

SAX-400 Performance of Critical Experiments

D. F. Hanlen, R. D. Leamer

A. Cores composed of 5.7% UO_2 stainless steel clad fuel in the 0.56" lattice plates.

Buckling and reflector savings measurements have been made in 361 and 441 rod square cores (19 x 19 and 21 x 21). The results agree quite well with each other and with that calculated from LEOPARD. Relative power distributions were measured through a 0.56" water slot, through the same water slot containing a 0.25" aluminum plate, and through a "slab" of five 0.403" Ag-In-Cd rods. Flux scans were also made using gold and U-238 foils and dysprosium wires.

The reactivity worth of a uniform array of six Ag-In-Cd control elements in a 17 x 27 fuel rod array was measured. Their worth relative to the water hole case was $5.6.

The just critical loading for this fuel in a W/U of 6.3 was found to be 235.4 rods in a best circle confirgulation. This W/U ratio was obtained by omitting every-other-rod from every-other-row.

The temperature coefficient was measured at two elevated temperatures in the 19 x 19 rod core. It was -1.9¢/°C at 72.5°C and -1.7¢/°C at 61.8°C.

Pulsed neutron measurements were made at six shutdown reactivities from \$0.05 to \$1.30. The value of β/ℓ (extrapolated to just critical) was 262 sec^{-1}.

B. Cores composed of 6.6% PuO_2-UO_2 zirconium clad fuel in the 0.56" lattice plates.

The just critical size in a circular configuration with the normal lattice was found to be 336.2 fuel rods, and the peripheral fuel rod worth was \$0.172 per rod. In the loose lattice (0.792" pitch) the just critical circular core contained 130.1 fuel rods, and the peripheral fuel rod worth was \$0.360 per rod. Buckling and reflector savings measurements were made in both normal and loose lattice loadings.

Fuel rod and foil scans (U-238 and gold foils) were made through a water slot, through the slot containing a 1/4" aluminum plate, and through a "slab" of five 0.403" Ag-In-Cd rods.

Pulsed neutron measurements were made, and data taken for a noise analysis determination for comparison. Temperature coefficient data were also obtained.

The core was borated, and buckling and reflector savings measurements were made. Criticality data were obtained at various boron concentrations, and pulsed neutron measurements made.

C. Cores of both fuels in the 0.56" lattice plates.

Cores were loaded with PuO_2-UO_2 fuel in the center surrounded by a region of UO_2 fuel (the normal configuration), and also in the inverted configuration with the PuO_2-UO_2 fuel outside. In all cores fuel rod scans, dysprosium and U-238 foil scans, and pulsed neutron data were obtained, and water slot and Ag-In-Cd slab characteristics also measured. In addition, the temperature coefficient, boron worth, and noise analysis data were obtained for the normal configuration.

D. Cores composed of 6.6% PuO_2-UO_2 zirconium clad fuel in the 0.52" lattice plates.

Critical sizes have been measured for this fuel in both the normal and loose lattice loadings in these plates. The normal (.52") lattice just critical circular loading was 471.5 rods with a peripheral fuel rod worth of $0.115 per rod; the loose (0.735") lattice required 151.2 rods with a peripheral worth of $0.413 per rod. Buckling data have been obtained in both lattices, and dysprosium and U-238 scans made in the normal loading.

E. Thermal Response and PuO_2-UO_2 Foil Measurements

In order to interpret fuel rod scans in two-region cores, it is necessary to know the gamma outputs of the different fuels at a known heat output. This measurement has been done two ways: (a) by direct measurement of the temperature rise of the two different fuel rods and subsequent gamma scannings, or (b) by irradiating foils of the different fuel materials, gamma scanning, and getting the ratio of total fissions from the production of specific fission products. Agreement between the two methods is satisfactory.

X-510 Nuclear Analyses of Operation Performance

F. L. Langford

The objective of this task is to compare the expected performance of the plutonium fuel in the Saxton reactor with experimental results and to evaluate the differences between analysis and experiment that are found. A second objective is to provide supporting analysis during the irradiation period. The supporting analysis will include an evaluation of the reactivity and power distribution changes with time corresponding to the operating history of the core.

A preliminary plan for core follow during operation has been suggested. In this plan the principal work events are:

1. Zero Power Physics Tests at Ambient Temperature
2. Reactor Heat-up
3. Zero Power Physics Tests at Operating Temperature
4. Power Escalation
5. Analyses of Start-up Data
6. Long Term Irradiation Follow
7. Repeat of (1) through (5) for at least two shutdowns and startups during the two year operating period.
8. Periodically review operating data and test results with regard to limiting core conditions and objectives of the project. Recommend any indicated alterations in power level and/or control rod program.
9. Issue Final Topical Report.

Remaining Sub-Tasks

E. A. McCabe, et. al.

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 - D. T. Galm

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 later in the program. The PERT-type summary schedule included at the end of the first Quarterly Report, WCAP-3385-1, applies in general except that the date for loading fuel in Saxton has been delayed by two months.

To disseminate knowledge is to disseminate prosperity — I mean general prosperity and not individual riches — and with prosperity disappears the greater part of the evil which is our heritage from darker times.

Alfred Nobel

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