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**SOUTHWEST EXPERIMENTAL
FAST OXIDE REACTOR
DEVELOPMENT PROGRAM
TWENTY-FOURTH QUARTERLY REPORT
FEBRUARY-APRIL 1970**

U. S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-540

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**SOUTHWEST EXPERIMENTAL FAST OXIDE
REACTOR DEVELOPMENT PROGRAM**

**Twenty-Fourth Quarterly Report
February—April 1970**

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ABSTRACT

This report summarizes the work performed on the Research and Development Program for the Southwest Experimental Fast Oxide Reactor during the period from February 1 through April 30, 1970. The work was performed under contract AT(04-3)-540 with the United States Atomic Energy Commission.

1. INTRODUCTION

Southwest Experimental Fast Oxide Reactor (SEFOR) is a 20 MWt, sodium-cooled fast reactor with characteristics similar to the large, soft spectrum fast breeder reactors fueled with mixed $\text{PuO}_2\text{--U-238 O}_2$, for which studies indicate the potential for producing low cost power. SEFOR will be used to obtain physics and engineering data at fuel compositions, temperatures, and crystalline states characteristics of power reactor operating conditions. SEFOR is particularly designed for the systematic determination of the Doppler coefficient of reactivity at temperatures up to the vicinity of fuel melting.

The SEFOR project consists of two major parts: the design and construction of the reactor facility and the related research and development program. Funds for the design and construction of the facility are being provided by the Southwest Atomic Energy Associates (a group of 17 investor-owned utility companies located in the south and southwest of the United States), together with the Karlsruhe Laboratory of The Federal Republic of Germany, Euratom, and the General Electric Company.

The United States Atomic Energy Commission is supporting the research and development program. The R and D program consists of two phases:

Phase I --Preoperational research and development,

Phase II --The reactor test program after the start of reactor operation

Phase I consists of 10 major tasks divided into 28 subtasks. This report summarizes the tasks scheduled for activity at this time. The remaining tasks will be included in future reports as the work progresses.

The following periodic reports have been issued in partial fulfillment of United States Atomic Energy Commission Contract AT(04-3)-540.

1. "Southwest Experimental Fast Oxide Reactor Development Program, First Quarterly Report, April-June 1964," GEAP-4594 (July 1964).
2. "Southwest Experimental Fast Oxide Reactor Development Program, Second Quarterly Report, July-September 1964," GEAP-4742 (October 1964).
3. "Southwest Experimental Fast Oxide Reactor Development Program, Third Quarterly Report, October 1964-January 1965," GEAP-4799 (February 1965).
4. "Southwest Experimental Fast Oxide Reactor Development Program, Fourth Quarterly Report, February-April 1965," GEAP-4864 (May 1965).
5. "Southwest Experimental Fast Oxide Reactor Development Program, Fifth Quarterly Report, May-July 1965," GEAP-4934 (August 1965).
6. "Southwest Experimental Fast Oxide Reactor Development Program, Sixth Quarterly Report, August-October 1965," GEAP-4994 (November 1965).
7. "Southwest Experimental Fast Oxide Reactor Development Program, Seventh Quarterly Report, November 1965-January 1966," GEAP-5108 (February 1966).
8. "Southwest Experimental Fast Oxide Reactor Development Program, Eighth Quarterly Report, February-April 1966," GEAP-5160 (May 1966).
9. "Southwest Experimental Fast Oxide Reactor Development Program, Ninth Quarterly Report, May-July 1966," GEAP-5208 (August 1966).
10. "Southwest Experimental Fast Oxide Reactor Development Program, Tenth Quarterly Report, August-October 1966," GEAP-5301 (November 1967).
11. "Southwest Experimental Fast Oxide Reactor Development Program, Eleventh Quarterly Report, November 1966-January 1967," GEAP-5442 (February 1967).
12. "Southwest Experimental Fast Oxide Reactor Development Program, Twelfth Quarterly Report, February-April 1967," GEAP-5498 (May 1967).
13. "Southwest Experimental Fast Oxide Reactor Development Program, Thirteenth Quarterly Report, May-July 1967," GEAP-5533 (August 1967).
14. "Southwest Experimental Fast Oxide Reactor Development Program, Fourteenth Quarterly Report, August-October 1967," GEAP-5561 (November 1967).

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13. "Southwest Experimental Fast Oxide Reactor Development Program, Thirteenth Quarterly Report, May-July 1967," GEAP-5533 (August 1967).
14. "Southwest Experimental Fast Oxide Reactor Development Program, Fourteenth Quarterly Report, August-October 1967," GEAP-5561 (November 1967).

15. "Southwest Experimental Fast Oxide Reactor Development Program, Fifteenth Quarterly Report, November 1967-January 1968," GEAP-5595 (February 1968).
16. "Southwest Experimental Fast Oxide Reactor Development Program, Sixteenth Quarterly Report, February-April 1968," GEAP-5629 (May 1968).
17. "Southwest Experimental Fast Oxide Reactor Development Program, Seventeenth Quarterly Report, May-July 1968," GEAP-5673 (August 1968).
18. "Southwest Experimental Fast Oxide Reactor Development Program, Eighteenth Quarterly Report, August-October 1968," GEAP-5705 (November 1968).
19. "Southwest Experimental Fast Oxide Reactor Development Program, Nineteenth Quarterly Report, November 1968-January 1969," GEAP-5754.
20. "Southwest Experimental Fast Oxide Reactor Development Program, Twentieth Quarterly Report, February-April 1969," GEAP-10010 (May 1969).
21. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-First Quarterly Report, May-July 1969," GEAP-10010-21 (August 1969).
22. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Second Quarterly Report, August-October 1969," GEAP-10010-22 (November 1969).
4. Protsik, R., "Epithermal Capture and Fission Distribution in a Fuel Rod-The Computer Code REAX," GEAP-4973 (April 1966).
5. Novak, P. E., and Asamoto, R. R., "An Out-of-Pile Evaluation of W-Re Thermocouple Systems for Use to 4700°F in Mixed Oxide," GEAP-5166 (May 1966).
6. Pflasterer, G. R., "SEFOR Experimental Program Planning, Vols. I and II," GEAP-5092 (August 1966).
7. Morrell, R. P., "SEFOR β/ℓ Measurements by Noise Analysis Techniques," IDO-17210 (August 1966).
8. Williams, R. F., and Polomik, E. E., "Pressure Drop and Dye Tests with Water on the SEFOR Prototype Fuel Bundle," GEAP-5129 (February 1966).
9. Craig, C. N., Ryer, C. M., and Thompson, M. L., "Plutonium-Uranium Mixed Oxide Fuel Pellet Fabrication Development for the Southwest Experimental Fast Oxide Reactor," GEAP-5285 (October 1966).
10. Polomik, E. E., and Williams, R. F., "Coolant Surface Turbulence Study, SEFOR Reactor Vessel," GEAP-5099 (December 1966).
11. Wilkinson, L., "Use of Neutron Filters for Fast Reactor Fuel Irradiations in a Thermal Core," GEAP-5035 (April 1967).
12. Wilkinson, L., et al., "Devices for Cyclic and Transient Power Control in a Steady-State Neutron Flux Gradient," GEAP-5036 (May 1967).
13. Reynolds, A. B., and Steward, S. L., "Analysis of the SEFOR Mockup Critical Experiments in ZPR-3," GEAP-5294 (March 1967).
14. Wilkinson, D., "Weighted Doppler Analysis Code-WEDOP," GEAP-5513 (October 1967).
15. Craig, E. R., Fulton, R. H., Damon, J. J., "Design and Testing of Solenoid and Instrument Lead Connector for Use in SEFOR Reactor Vessel," GEAP-5307 (March 1967).
16. Asamoto, R. R., Perry, K. J., and Zebroski, E. L., "The Effective and Diametral Thermal Expansion of UO_2 Under a Radial Temperature Gradient," GEAP-5284 (October 1966).

In addition, the following topical reports have been issued:

1. Novak, P. E., and Asamoto, R. R., "A Survey for a High Temperature Sensor for SEFOR," GEAP-4903 (July 1965).
2. Quinn, E. P., "Vibration of SEFOR Fuel Rods in Parallel Flow," GEAP-4966 (September 1965).
3. Johnson, M. L., "A Description of Fuel and Other Core Materials Procured for SEFOR Critical Experiments in ZPR-III," GEAP-5133 (December 1965).

17. McVean, R. L., Weitzberg, A., Kremser, J. A., Leridon, A. M., and Long, J. K., "Critical Studies of the Southwest Experimental Fast Oxide Reactor (SEFOR) in ZPR-3," ANL-7348.
18. Polomik, E., "Fluid Flow and Vibration Tests, 1/2-Scale Model of SEFOR Reactor Vessel," GEAP-5440 (April 1967).
19. Field, J. H., Johnson, M. L., and Novak, P. E., "An Evaluation of the Effect of Design and Operating Variables on SEFOR Fuel and Fuel Cladding," GEAP-5309 (December 1967).
20. Noble, L. D., "Final Specifications for the SEFOR Experimental Program," GEAP-5576 (April 1968).
21. Russell, C. E., "Flow Distribution Measurements in a 3-Scale Model of the SEFOR Fuel Bundle," GEAP-5184 (May 1968).
22. Cochran, J., "SEFOR Core Clamping Model Tests, One-Sixth Segment," GEAP-5465 (May 1968).
23. Asamoto, R. R., and Novak, P. E., "An In-Pile Evaluation of W-Re Thermocouple Systems for Use to 4700°F in SEFOR Fuel," GEAP-5466 (October 1968).
24. Cochran, J. T., "SEFOR Fuel Assembly Sodium Flow Tests," GEAP-5466 (September 1968).
25. Cochran, J. T., "SEFOR Fuel Rod Sodium Thermal Shock Tests," GEAP-5467 (August 1968).
26. Craig, E. R., "SEFOR Instrumented Fuel Assembly Design and Development," GEAP-5615 (April 1968).
27. Steamer, A. G., Brynsvold, G. V., and Franckx, L. C., "Fuel Handling and Other Tests in the SEFOR Refueling Cell Mockup," GEAP-5701 (March 1969).
28. Johnson, M. L., Becker, R. A., Haar, J. M., "SEFOR Fuel Handling," GEAP-13565 (December 1969).
29. Noble, L. D., et al., "Results of SEFOR Zero Power Experiments," GEAP-13588 (March 1970).

2. SUMMARY

2.1 COMPLETED TASKS

Experimental and/or analytical work has been completed on the tasks listed below, although in some cases work on topical reports is continuing:

TASK	TITLE
1.1	Specifications for the Reactor and Experimental Program
1.2	Final Specifications of the Experimental Program
2.0	Experimental Program in ZPR-III
3.0	Fuel Development
4.0	Special Instrumentation
5.0	Steady-State and Transient Characteristics of the Core Assembly
6.0	Reactivity Controls Required for Experimentation
7.0	Fuel Handling
8.0	Core Design

2.2 TASK 9—FUEL FABRICATION

One instrumented fuel assembly (IFA) was reworked at Vallecitos Nuclear Center (VNC) and returned to SEFOR. Work continued on the sixth IFA at VNC.

Negotiations were conducted with Nuclear Fuel Services (NFS) concerning the fabrication of new and replacement fuel rods.

2.3 TASK 10—STARTUP TESTING

A symmetric loading of B_4C and guinea pig rods were arranged within the core to provide enough excess reactivity for the approach to power. The fine reflectors were calibrated for this core configuration (Assembly I-F), as well as a core (Assembly I-G) with excess reactivity reduced to approximately 1.5 dollars to simulate the reflector configuration at power. Reflectors were also calibrated in the present core loading (Assembly I-I) and in a similar core loading (Assembly I-H) containing instrumented fuel assemblies. In addition, apparent coarse reflector worths were measured in Assembly I-H.

The flow coefficient of reactivity was measured at flows up to the maximum rated value of 5000 gal/min. In agreement with previous measurements up to flows of 4000 gal/min, the data indicate that the flow coefficient of reactivity is essentially zero.

The reactivity worth of the poison oscillator rod was measured as a function of axial rod position. The shape of the reactivity-position curve is as expected, and the worth (33 cents) with the lower tip of the rod at the core midplane is in agreement with previous B_4C rod reactivity worth measurements. There is an apparent discrepancy at some positions of approximately 10% in the static and dynamic calibration of the slope (cents/inch) of the oscillator rod worth core. Additional static measurements have been proposed to clarify this apparent discrepancy.

The reactivity feedback between 360°F and 760°F for the present core loading (Assembly I-I) was determined. The feedback effects are in agreement with the previous measurements in Assembly I-E.

Further analysis of the previously performed fission rate measurements has provided generally better agreement between experiment and calculation by accounting for the effect of the aluminum-filled foil holder rod on the measured distributions. However, the measured axial distribution is still somewhat flatter than calculated. The measured fission ratios near the core center that were obtained by counting the 1.6 Mev decay gamma of La-140 are in good agreement with previous² predictions.

Calculations were performed to determine the influence of a flatter power distribution on the Doppler coefficient and on the fuel temperatures. The effect on the Doppler calculation is less than 3%, and the maximum fuel temperature was reduced (as is to be expected).

The zero power oscillator test was performed at eleven frequencies between ~ 0.25 and 0.0017 cycles per second. The entire frequency range was traversed three times. The measured transfer function between power and reactivity is in good agreement with the transfer function calculated using Keepin's³ delayed neutron data and the calculated fractions of fission in Pu-239 and U-238.

The SEFOR power coefficient of reactivity has been estimated from data that were taken during operation up to a power level of ~ 1 MW. The estimated value between zero and one megawatt is $-19.5\text{¢/MW} \pm 6.2\text{¢/MW}$. The confidence in this value cannot be reliably determined from the limited amount of steady-state power data. However, the indicated range represents the estimated 80% confidence interval based on one degree of freedom.

This compares with the calculated power coefficient of reactivity between zero and one megawatt of $-16.5\text{¢}/\text{MW}$. Of this, $-16.0\text{¢}/\text{MW}$ is the calculated Doppler effect. The other component ($-0.5\text{¢}/\text{MW}$) is the calculated clad and fuel axial expansion power coefficient.

2.4 TASK 11—TRANSIENT TESTS

A draft of the license application for a fuel transfer cask has been completed and sent to the AEC for approval.

2.5 SEFOR SAFETY AUDIT

Recommendations made in audit reports for the past year were reviewed to assure proper action on all items. Action was initiated on a recommendation to install new thermocouples in the reflector region.

2.6 SEFOR SUPPORT

During the report period, thermocouples located in the reflector guide structure failed at an accelerated rate. A repair program was initiated to provide substitute instrumentation in the reflector control and guide structure region.

A reactor vessel flow-level anomaly was corrected with the installation of the drywell into the central core position. Tests completed at the site verified that no observable level changes occurred over the full primary flow range.

The two remaining type II reactor vessel level probes were delivered to the site. A malfunction occurred in the type I through head level probe, which required its removal, repair, and replacement.

2.7 SEFOR OPERATION

The experimental program continued, but progress was slowed by malfunctions of sodium level probes, sodium pipe heater circuits, and the oscillator mechanism. During the increase in sodium temperature to 760° , a verification of the isothermal temperature coefficient was obtained, and reflector calibrations were performed. Flow coefficients of reactivity measurements were extended to 100% flow (5000 gpm). Intercalibrations of the venturi and electromagnetic flow meters were obtained as a function of sodium temperature. Static and dynamic measurements were made of the reactivity worth of the oscillator poison rod as a function of core position. The zero power frequency response measurements were completed, during which the reactor was critical for the longest continuous period to date. Reactor power was increased to 1 MW, plant heat balances were made, neutron instrument calibrations were accomplished, natural circulation cooling by the air blast coolers was established, and preliminary radiation shielding measurements were made. Continuation of the experimental program was delayed to replace malfunctioned reflector guide structure thermocouples. Procedure writing and measurements documentation continued.

3. TASK 9—FUEL FABRICATION

3.1 PURPOSE

The purpose of Task 9 is to procure fuel rods and associated core components for SEFOR. Fabrication of the hexagonal fuel channel and all of its internal components are a part of this Task, as well as the instrumented fuel assembly fabrication.

3.2 PAST ACCOMPLISHMENTS

The complete fuel-rod order had been received at SEFOR in previous quarters. Of 50 rods rejected for helium leaks and dimensional deficiencies, some 20 were corrected in a previous quarter. Of the five instrumented fuel assemblies (IFA) completed and shipped to SEFOR, two have been installed in the reactor.

3.3 CURRENT RESULTS

3.3.1 Instrumented Fuel Assemblies (IFA)

One IFA (197R243 G6) was shipped from the reactor site to Vallecitos for correction of a fabrication error in the sensor lead wires. Errors in the electrical lead splice were corrected by opening up the final weld and making the required corrections to the splice arrangement. After final

inspection, the assembly was returned to the SEFOR site for insertion in the reactor at a later date. The shipment date was March 8, 1970.

The flowmeter that was fabricated for use on the sixth IFA (197R243 G6) was checked and found to be low in magnetic flux as compared with earlier similar fabrications. This flowmeter was disassembled and repaired for later use in the sixth IFA assembly.

One of the fuel rods for the sixth IFA (G7) was damaged during a welding operation at the Pu Laboratory. The sheathed thermocouple leads were twisted off at a point very close (approximately 1/4 inch) to the lead guard, which is brazed to the fuel rod end spacer. The fuel rod was sealed off at this assembly stage (two sections completed) and plans were made to attempt a repair of the subassembly by splicing the thermocouple leads.

The second fuel rod (G8) for the sixth IFA was partially completed, then delayed waiting for the mating fuel rod to be repaired. The fuel rods are fabricated in pairs to improve fabrication scheduling.

The preparation of the fabrication report was continued. The draft copy was near completion at the end of this report period.

4. TASK 10—STARTUP TESTING

4.1 PURPOSE

The purpose of Task 10 is to investigate the startup characteristics of a $\text{PuO}_2\text{-UO}_2$ fueled fast reactor and to compare the experimental results to the theoretical calculations. The wet critical tests comprise Task 10.3. Task 10.4 includes the initial approach to full power.

4.2 PAST PERFORMANCE

The following portions of the experimental program at zero power have been performed. A minimum critical fuel loading of 522 fuel rods was attained in good agreement with the predicted¹ value of 519 fuel rods. The control strength of all reflector segments was estimated as 9.7 dollars using initial fuel loading data. This compares with the predicted value of 11.2 ± 1.8 dollars. The reactivity worths of fuel, guinea pig, B_4C , UO_2 , stainless steel, and BeO tightener rods were measured in two different core loadings. The magnitude of the experimental reactivity worths was approximately 10% larger than the calculated values. The ratio (λ/β) of the prompt neutron lifetime to the delayed neutron fraction was measured. The measured value (0.205×10^{-3} sec) was about 11% higher than the calculated value of 0.184×10^{-3} sec. Axial and radial fission distributions of U-238, Pu-239, and U-235 foils were measured. The flow and pressure coefficients of reactivity were measured to be essentially zero in agreement with expectations. The temperature reactivity effect between 350°F and 760°F was measured and found to be in excellent agreement with reactivity effects calculated, using the predicted² Doppler $T \frac{dk}{dT}$ of -0.0082 and the combined density-expansion coefficient of -0.36% /°F.

4.3 CURRENT RESULTS

4.3.1 Core Loading and Reflector Calibration

A symmetrical loading of B_4C and guinea pig rods was arranged within the core to provide enough excess reactivity for the approach to power. This loading (Assembly I-F shown in Figure 1) was critical at 350°F with three reflectors lowered and one reflector partially raised.

Fine Reflector No. 8 was calibrated using period measurements with this (Assembly I-F) core loading at 350°F . For this calibration, Reflectors No. 5 and No. 10 were down, No. 6 was at ~ 10 cm, and all coarse reflectors were raised. The results are shown in Table 1 and Figure 2.

Table 1

REFLECTOR NO. 8 CALIBRATION—ASSEMBLY I-F

Reflector No. 8 (cm)		Period (sec)	Reactivity (ρ)
From	To		
1.03	9.14	143.15	6.588
9.14	17.73	96.10	9.090
17.73	24.35	97.11	9.016
24.35	30.30	98.85	8.892
30.30	35.36	101.73	8.693
35.36	40.18	103.03	8.606
40.18	44.93	101.59	8.702
44.93	49.53	111.69	8.070
49.53	54.06	120.06	7.613
54.06	59.88	89.18	9.633
59.88	66.27	87.16	9.804
66.27	72.64	102.02	8.673
72.64	80.02	107.79	8.303
80.02	88.50	137.23	6.823
88.50	98.18	269.55	3.805

As part of the program to provide fine reflector reactivity calibrations for the core loadings and reflector configurations which will occur during reactor heatup and power operation, the excess reactivity of the core was reduced by removing fuel, guinea pig, and tightener rods from Assembly I-F. The new core loading (Assembly I-G shown in Figure 3) also contained two instrumented fuel assemblies (IFAs). Each IFA, which contains two temperature-instrumented fuel rods and a central tightener rod, as well as sodium temperature thermocouples and a flow meter, is described in detail in Reference 4. In Assembly I-G there were eight vacant (sodium-filled) fuel rod locations and three vacant tightener rod locations. Reflector No. 8 was calibrated in this core at 350°F with Reflector No. 5 at ~ 95 cm and all other coarse reflectors raised. The results are shown in Table 2 and Figure 4.

After calibration of the fine reflector, the vacant fuel and tightener rod locations were filled. The resultant core loading was designated Assembly I-H. A loading diagram for this core is not shown, since it is identical (except for the central dry well) to the subsequent loading (Assembly I-I). Fine Reflector No. 3 was calibrated at 350°F in this core with Reflectors No. 5 and No. 10 down, No. 6 at ~ 46 cm, and all other coarse reflectors raised. The results are shown in Table 3 and Figure 5.

Figure 1. Core Loading for Assembly I-F

Figure 2. Fine Reflector No. 8 Calibration Curve at 350°F (Assembly I-F)

Figure 3. Core Loading for Assembly I-G

Figure 4. Fine Reflector No. 8 Calibration Curve at 350°F (Assembly I-G)

Figure 5. Fine Reflector No. 3 Calibration Curve at 350°F (Assembly I-H)

Table 2
REFLECTOR NO. 8 CALIBRATION—ASSEMBLY I-G

Reflector No. 8 (cm)		Period (sec)	Reactivity (β)
From	To		
0.65	9.10	135.50	6.895
9.10	17.83	82.83	10.193
17.83	24.38	89.61	9.597
24.38	29.52	103.90	8.549
29.52	34.51	97.11	9.016
34.51	39.27	94.37	9.220
39.27	43.71	104.33	8.521
43.71	48.56	89.32	9.621
48.56	53.33	94.66	9.198
53.33	58.47	89.18	9.633
58.47	64.72	76.05	10.870
64.72	71.49	81.39	10.330
71.49	79.85	77.78	10.689
79.85	88.61	119.34	7.651
88.61	98.18	229.00	4.400

The present core loading (Assembly I-I) was obtained by inserting the central drywell* into Assembly I-H. (The measured reactivity worth of the drywell is $\sim +3\beta$.) This assembly is illustrated in Figure 6. In this assembly, Reflector No. 8 was calibrated at 550°F and 760°F, while No. 3 was calibrated at 760°F. For the calibration of No. 8 at 550°F, Reflector No. 5 was down, No. 10 was at ~ 43 cm, and all other coarse reflectors were raised. At 760°F, Reflector No. 8 was calibrated with No. 5 at ~ 36 cm, and Reflector No. 3 was calibrated with No. 10 at ~ 38 cm. In both cases, all other coarse reflectors were completely raised. The results of these calibrations are shown in Tables 4 through 6 and Figures 7 through 9.

In addition to the fine reflector calibrations described above, the effects of different reflector configurations on apparent reflector worth were investigated in Assembly I-H. The data are shown in Table 7. The position versus worth calibration curve (as well as the total worth) of Reflector No. 3 in Assembly I-H at 350°F was used to estimate the worth of Reflector No. 8 during these measurements, since the worths of No. 3 and No. 8 appear to be the same for

* The drywell provides an in-core argon atmosphere for the oscillator rod.

Table 3
REFLECTOR NO. 3 CALIBRATION—ASSEMBLY I-H

Reflector No. 3 (cm)		Period (sec)	Reactivity (β)
From	To		
0.4	8.96	152.35	6.254
8.96	17.07	100.84	8.753
17.07	23.57	98.53	8.914
23.57	29.11	101.42	8.713
29.11	34.08	107.62	8.313
34.08	38.86	100.12	8.802
38.86	43.75	95.94	9.102
43.75	48.17	110.37	8.148
48.17	52.69	116.28	7.813
52.69	58.07	91.90	9.412
58.07	64.16	83.39	10.141
64.16	70.40	94.06	9.243
70.40	77.35	100.70	8.763
77.35	85.12	121.62	7.534
85.12	97.67	146.87	6.449

Table 4
**REFLECTOR NO. 8 CALIBRATION—ASSEMBLY I-I
(550°F)**

Reflector No. 8 (cm)		Period (sec)	Reactivity (β)
From	To		
1.03	9.58	125.23	7.357
9.58	18.38	76.18	10.857
18.38	25.01	84.40	10.048
25.01	30.32	96.52	9.059
30.32	35.26	98.10	8.944
35.26	40.01	98.83	8.893
40.01	44.73	97.67	8.975
44.73	49.49	96.66	9.049
49.49	54.30	97.09	9.017
54.30	59.51	95.22	9.156
59.51	65.67	83.68	10.114
65.67	71.52	104.89	8.485
71.52	77.96	112.30	8.030
77.96	85.01	138.79	6.760
85.01	98.16	144.99	6.518

Figure 6. Core Loading for Assembly I-I

Figure 7. Fine Reflector No. 8 Calibration Curve at 550°F (Assembly I-I)

Figure 8. Fine Reflector No. 8 Calibration Curve at 760°F (Assembly I-I)

Figure 9. Fine Reflector No. 3 Calibration Curve at 760°F (Assembly I-I)

Table 5
REFLECTOR NO. 8 CALIBRATION—ASSEMBLY I-I
(760°F)

Reflector No. 8 (cm)		Period (sec)	Reactivity (ρ)
From	To		
1.02	9.63	117.17	7.767
9.63	17.04	96.83	9.038
17.04	23.74	84.42	10.048
23.74	29.15	97.38	8.996
29.15	34.35	90.46	9.527
34.35	39.11	95.94	9.102
39.11	43.73	95.94	9.102
43.73	48.73	87.86	9.744
48.73	53.56	94.35	9.221
53.56	58.57	96.68	9.047
58.57	64.01	94.52	9.208
64.01	69.70	100.43	8.781
69.70	75.92	106.93	8.356
75.92	82.99	125.40	7.349
82.99	90.00	197.84	5.002
90.00	98.18	350.79	2.996

Table 6
REFLECTOR NO. 3 CALIBRATION—ASSEMBLY I-I
(760°F)

Reflector No. 3 (cm)		Period (sec)	Reactivity (ρ)
From	To		
1.04	9.50	120.90	7.571
9.50	16.98	95.94	9.102
16.98	23.13	93.63	0.276
23.13	28.47	93.92	9.254
28.47	33.52	93.78	9.265
33.52	38.54	89.01	9.646
38.54	43.02	103.01	8.607
43.02	47.51	102.43	8.646
47.51	52.12	100.70	8.763
52.12	57.08	99.26	8.863
57.08	62.58	90.31	9.539
62.58	68.68	89.74	9.587
68.68	73.69	136.84	6.840
73.69	80.91	110.22	8.156
80.91	89.11	150.40	6.322
89.11	97.68	303.18	3.422

similar symmetric reflector configurations (e.g., the calibration curves at 760°F in Assembly I-I). The data of Table 7, with the previously determined (see above) differential and total worth of Reflector No. 3 (and No. 8) were used to estimate the "apparent worths" listed in Table 8. These values again demonstrate the previously⁶ reported "shadowing" effect of a lowered reflector on the worth of its adjacent neighbor. Some of the worths in Table 8 are apparent rather than true values, because the fine reflector worths were also affected by the reflector configuration; i.e., the fine reflectors were calibrated with their adjacent neighbors completely raised, and they were then used as "yardsticks" even when an adjacent reflector was completely lowered. The largest differences between apparent and true worths are thus expected to occur with Reflectors No. 4 and No. 9, where the true worth in this configuration may be ~10% larger than indicated in Table 8.

4.3.2 Flow Coefficient Measurements

The flow coefficient of reactivity was measured at flows up to the maximum rated value of 5000 gal/min in Assembly I-I. These measurements duplicate and extend the range of previous⁵ measurements for flows up to 4000 gal/min. The data are tabulated in the sequential order of data collection in Table 9 and are illustrated in Figure 10. The data indicate little correlation between flow and reactivity, and the effects of flow upon reactivity can be neglected.

The data from the previous⁵ measurements in Assembly I-E showed no correlation. The present data (see Table 9 and Figure 10) indicate that there may be a loss in reactivity of the order of one cent as the flow is increased from 500 gal/min to 3500 gal/min and that this reactivity is regained as the flow is increased to 4000 gal/min. However, in view of the small magnitude of this apparent change, the relatively large temperature changes (~20°F) that occurred during these measurements, and the lack of any such apparent correlation in the previous measurements, it appears that there is no appreciable correlation between flow and reactivity.

4.3.3 Oscillator Rod Calibration

The reactivity worth of the poison oscillator rod was measured as a function of axial rod position. The data are tabulated in Table 10 and illustrated in Figure 11. The worth of the rod at a position of -1.74 inches was determined at a different time than the other values, by using a calibrated reflector control segment to measure the incremental change in reactivity as the tip of the rod was

Table 7
REFLECTOR POSITIONS IN ASSEMBLY I-H

Measurement Number	1	2	Reflector Number and Position (cm)										Temperature (°F)	Relative** Worth at 392.0° F (cents)
1			0		0			54.09		0			389.0	79.9)
			0		0			58.71		0			400.0	80.3)
2			56.08		0			0		0			393.0	80.1 Average
3		0	46.49		0					0			392.5	63.6
4			26.42	0	0					0			392.0	28.3
5	0		38.97		0					0			392.25	49.8
6			0		0	0		32.98					392.0	39.4
7					0	0		38.85		0			392.0	50.1
8					0		0	45.62		0			392.0	62.8
9					0			29.53	0	0			392.25	33.3
10		0	46.31					0		0			392.25	63.5

* Reflectors were raised unless otherwise indicated.

** This is the worth of the partially raised fine reflector obtained using the calibration curve of reflector No. 3 at 380° F in Assembly I-H. Temperature corrections were made using the measured temperature coefficient of -0.66¢/°F.

moved from +0.26 inches to -1.74 inches. This measured increment was added to the previously measured worth at 0.26 inches to form a continuous curve.

The total rod worth at the core center (~33¢) is in agreement with previous B_4C rod worth measurements, but the slope of this static curve (cents/inch) appears to be approximately 10% lower at some points than the worth determined from the dynamic calibration (see below). Additional oscillator rod worth measurements by the "period" method are planned to clarify this apparent discrepancy.

Table 8
APPARENT REFLECTOR WORTHS IN
ASSEMBLY I-H

Reflector Number	Apparent** Worth (cents)
1	96 ¹
2	110 ¹
3	126.4 ⁺
4	75 ¹
5	127 ²
6	96 ³
7	109 ³
8	126.4 ⁺
9	80 ³
10	137 ⁴

* Worth obtained from period measurements.

+ Set equal to worth of No. 3.

** The worths were obtained from the data of Table 7.

1. This is the apparent worth of the indicated reflector when it is swapped with reflector No. 8 while reflectors No. 5 and No. 10 are down and reflector No. 3 is adjusted to maintain criticality.
2. This is the apparent worth of reflector No. 5 when it is swapped with reflector No. 8 while reflectors No. 2 and No. 10 are down and reflector No. 3 is adjusted to maintain criticality.
3. This is the apparent worth of the indicated reflector when it is swapped with reflector No. 3 while reflectors No. 5 and No. 10 are down and reflector No. 8 is adjusted to maintain criticality.
4. This is the apparent worth of reflector No. 10 when it is swapped with reflector No. 3 while reflectors No. 2 and No. 10 are down and reflector No. 8 is adjusted to maintain criticality.

4.3.4 Temperature Reactivity Effects in Assembly I-I

The temperature-dependent reactivity feedback between 360°F and 760°F at zero power was measured in Assembly I-I. Data taken during heating to 760°F are tabulated in Table 11. In determining the indicated change in reactivity, the reactivity calibration curve for Reflector No. 8 in Assembly I-F at 350°F was used for Readings No. 1 through 9; the calibration curve for Reflector No. 8 at 550°F in Assembly I-I was used for Readings No. 10 through 21; the calibration curve for Reflector No. 3 at 760°F in Assembly I-I was used for Readings No. 22 through 25; and the calibration curve for Reflector No. 8 at 760°F in Assembly I-I was used for the remaining readings. The different portions of the reactivity data were joined together to form a smooth curve of reactivity feedback as a function of temperature with the origin selected to indicate the reactivity changes from a reference temperature of 360°F.

The reactivity feedback effects in Assembly I-I are in good agreement throughout the entire temperature range with the previous⁶ measurements in Assembly I-E. Thus,⁶ they are also in agreement with the calculated reactivity effects obtained using a Doppler coefficient ($T \frac{dk}{dT}$) of -0.0082 and a constant density-expansion coefficient of -0.36¢/°F.

4.3.5 Fission Rate Distributions

Relative fission rates for Pu-239, U-238, and U-235 foils were measured at selected positions in the core. The foils were placed in an aluminum-filled foil holder rod, and the rod was placed at selected locations in the core. These locations are illustrated in Figure 12. The foil positions within the core, and the measured relative counting rates for the fission product gamma activity in the energy range between 430 and 730 keV, are shown in Tables 12 through 14. (The positions within the core are slightly different than those indicated in Reference 5.) Figures 13 through 20 compare the measured fission rate distributions with the distributions obtained using two-dimension diffusion theory calculations in 13 energy groups. Infinitely dilute cross sections were used to calculate the fission rate distribution, and the U-238 axial profiles were corrected as discussed below.

Analysis of the U-238 fission distribution measurements indicated that the U-238 fission rate within the aluminum-filled foil holder rods may be about 5% lower (except in the gap region) than in the adjacent fuel rods. Also, the analysis indicated that the axial profile within the foil holder rod is significantly less depressed through the fuel-gap region than is the U-238 fission distribution within

Table 9
FLOW-REACTIVITY MEASUREMENTS
 (Assembly I-I)

Coolant Flow (gpm)	Reflector No. 8 Position (cm)	Worth* (ϵ)	Temperature (°F)	Relative System Reactivity at 390°F** (ϵ)	Deviation from*** Mean Reactivity (ϵ)
500	54.09	46.99	389.0	46.31	+0.33
500	53.89	47.32	388.25	46.13	+0.15
500	53.68	47.67	387.75	46.14	+0.16
1000	53.55	47.89	387.0	45.85	-0.13
1000	53.15	48.55	386.25	46.00	+0.02
1000	52.90	48.97	385.5	45.91	-0.07
1500	52.76	49.20	385.0	45.80	-0.18
1500	52.55	49.55	384.5	45.81	-0.17
1500	52.38	49.84	384.0	45.76	-0.22
2000	52.31	49.96	383.75	45.71	-0.27
2000	52.15	50.22	383.25	45.63	-0.35
2000	52.01	50.46	383.0	45.70	-0.28
3000	51.93	50.59	382.5	45.49	-0.49
3000	52.09	50.32	382.75	45.39	-0.59
3000	52.22	50.11	383.0	45.35	-0.63
3500	52.19	50.16	382.75	45.23	-0.75
3500	52.25	50.06	382.75	45.13	-0.85
3500	52.15	50.22	382.5	45.12	-0.86
4000	58.71	39.31	400.0	46.11	+0.13
4000	58.74	39.26	400.0	46.06	+0.08
4000	58.57	39.54	400.0	46.34	+0.36
4000	58.55	39.57	399.75	46.20	+0.22
4450	58.68	39.36	400.0	46.16	+0.18
4450	58.84	39.10	400.25	46.07	+0.09
4450	58.86	39.06	400.5	46.20	+0.22
4450	58.93	38.95	400.5	46.09	+0.11
4900	59.37	38.23	401.5	46.05	+0.07
4900	59.49	38.04	402.0	46.20	+0.22
4900	59.78	37.57	402.5	46.07	+0.09
4900	59.77	37.58	402.75	46.25	+0.27
500	57.69	40.98	399.25	47.27	+1.29
500	57.52	41.26	398.0	46.70	+0.72
500	57.35	41.55	397.25	46.48	+0.50
500	57.35	41.55	397.25	46.48	+0.50

Average = 45.98

* Relative to worth in the completely raised position (~98 cm) using the reflector calibration curve for Assembly I-F.

** Using a temperature coefficient at 390°F of -0.68 ϵ /°F.

*** Estimated standard deviation in measurements is $\pm 0.5\epsilon$.

Table 10
POISON OSCILLATOR ROD CALIBRATION

Rod* Position (inches)	Reflector No. 3 Position (cm)	Worth (ϕ)	Temperature 741.5°F (°F)	Worth** Relative to 741.5°F (ϕ)	Change ⁺ is Worth (ϕ)
20.19	30.57	38.99	741.5	38.99	-0.09
18.00	30.80	39.41	742.0	39.13	-0.23
16.00	31.29	40.31	742.0	40.03	-1.13
14.00	32.17	41.94	742.0	41.66	-2.76
12.00	33.64	44.69	742.5	44.14	-5.24
10.00	35.44	48.12	742.75	47.43	-8.53
8.00	37.86	52.79	743.5	51.69	-12.79
6.00	40.25	57.39	743.5	56.29	-17.39
4.00	42.73	62.15	742.5	61.60	-22.70
2.00	44.77	66.09	741.75	65.95	-27.05
0.26	47.42	71.18	740.0	72.01	-33.11
2.00	45.07	66.66	741.5	66.66	-27.76
4.00	42.52	61.75	742.0	61.47	-22.57
6.00	39.95	56.81	741.75	56.67	-17.77
8.00	37.60	52.29	741.5	52.29	-13.39
10.00	34.89	47.07	741.5	47.07	-8.17
12.00	33.28	44.01	741.5	44.01	-5.11
14.00	32.17	41.94	742.25	41.53	-2.63
16.00	31.53	40.75	743.0	39.92	-1.02
18.00	31.17	40.09	743.0	39.26	-0.36
20.19	31.22	40.18	744.0	38.80	+0.10
0.26	47.01	70.41	762.0	69.27 ⁺⁺⁺	
-1.74	50.69	77.45	762.0	76.31 ⁺⁺⁺	-40.12 ⁺⁺
0.26	47.19	70.75	762.5	69.33 ⁺⁺⁺	

* Distance above the core mid-plane of the lower end of the 26-inch long poison oscillator rod.

** Corrected for temperature effects using a temperature coefficient of $-0.55\phi/^{\circ}\text{F}$.

+ Relative to the average worth, in the adjacent column, at 20.19 inches.

++ Normalized by adding -7.01ϕ (as indicated by the data in the adjacent column) to the value of -33.11ϕ at 0.26 inches.

+++ These worths are relative to 760°F .

Figure 10. Flow Reactivity Measurements (Assembly I-I)

Figure 11. Static Calibration of Oscillator Rod

Table 11
TEMPERATURE-REACTIVITY EFFECTS (360-760°F)
(Assembly I-I)

Reading Number	Reflector Position (cm)		Reactor Inlet+ Temperature (°F)	Negative Reactivity* Feedback (cents)
	No. 6	No. 8		
1	31	1.03	360	0.0
2	31	11.07	371	8.4
3	31	25.31	400	26.1
4	31	35.59	427	42.7
5	31	44.35	450	58.5
6	31	52.71	472	73.0
7	31	60.86	495	86.5
8	31	69.63	515	99.4
9	31	79.56	532	111.2
10	up	37.43	537	114.3
11	up	41.37	549	121.7
12	up	44.33	557	127.3
13	up	46.37	564	131.3
14	up	47.06	564	132.6
15	up	50.00	574	138.2
16	up	53.62	588	145.0
17	up	60.27	602	156.7
18	up	64.38	614	163.4
19	up	67.67	623	168.5
20	up	70.12	628	172.0
21	up	74.53	639	177.9
22				

Reflectors No. 5 and No. 10 were down; all other reflectors were up.

* Relative to 360°F.

+ Estimated uncertainty in temperature is $\pm 2.5^\circ\text{F}$.

Table 11 (Continued)

TEMPERATURE-REACTIVITY EFFECTS (360-760°F)
(Assembly I-I)

Reading Number	Reflector Position (cm)		Reactor Inlet+ Temperature (°F)	Negative Reactivity* Feedback (cents)
	No. 3	No. 8		
22	72.05	1.03	643	180.8
23	75.51	1.03	651	185.2
24	83.60	1.03	665	193.5
25	94.50	1.03	675	200.0
26	up	8.50	688	207.4
27	up	14.65	700	214.5
28	up	27.48	731	233.8
29	up	28.68	738	235.9
30	up	29.43	740	237.2
31	up	31.86	748	241.6
32	up	32.42	749	242.6
33	up	32.92	750	243.6
34	up	33.88	753	245.4
35	up	34.59	756	246.7
36	up	35.01	758	247.5
37	up	35.27	758	248.0
38	up	35.51	759	248.4
39	up	35.39	759	248.2
40	up	35.39	759	248.2

Reflector No. 5 was down; all other reflectors were up.

* Relative to 360°F.

+ Estimated uncertainty in temperature is $\pm 2.5^\circ\text{F}$.

Figure 12. Foil Holder Rod Locations (Assembly I-D)

Table 12
FISSION RATE MEASUREMENTS FOR Pu FOILS
(Relative Values)

Rod Number		FB	FG	FC	FF	FE	FD
Radial							
Distance* (cm)		5.8	19.6	23.2	43.3	43.3	44.5
Axial Position							
No.	Distance** (cm)						
8	90.8	0.6013	-	-	-	-	-
7	73.5	0.7388	-	-	-	-	-
6	58.2	0.9198	-	-	-	-	-
5	54.9	0.9396	-	-	0.5769	-	0.5871
4	53.0	0.9423	-	-	-	-	-
3	37.0	1.0000 ⁺	0.8564	0.8111	0.5543	-	0.5747
2	26.1	0.9419	-	-	-	-	-
1	1.2	0.6901	-	-	-	-	-

* Distance from core center

** Distance above bottom of core - (core height = 91.2 cm)

+ The specific activity at this point was 6398.3 counts/min/mg.

Table 13
FISSION RATE MEASUREMENTS
FOR ENRICHED URANIUM FOILS
(Relative Values)

Rod Number		FB	FG	FC	FF	FE	FD
Radial							
Distance* (cm)		5.8	19.6	23.2	43.3	43.3	44.5
Axial Position							
No.	Distance** (cm)						
8	90.8	0.6240	0.5418	0.5018	-	-	-
7	73.5	0.7778	0.6629	0.6184	0.4311	-	0.4747
6	58.2	0.9515	0.8222	0.7807	0.5673	-	0.6055
5	54.9	0.9750	0.8679	0.8093	0.5979	0.5003	0.6114
4	53.0	0.9932	0.8806	0.7949	0.5953	-	0.6001
3	37.0	1.0000 ⁺	0.8599	0.8154	0.5703	0.5450	0.5782
2	26.1	0.9633	0.8524	0.7742	0.5275	-	0.5523
1	1.2	0.7191	0.6114	0.5739	-	-	-

* Distance from core center.

** Distance above bottom of core -(core height = 91.2 cm)

+ The specific activity at this point was 6410.2 counts/min/mg.

Table 14
FISSION RATE MEASUREMENTS
FOR DEPLETED URANIUM FOILS
(Relative Values)

Rod Number		FB	FG	FC	FF	FE	FD
Radial							
Distance* (cm)		5.8	19.6	23.2	43.3	43.3	44.5
Axial Position							
No.	Distance** (cm)						
8	90.8	0.4156	0.3667	0.3453	-	-	-
7	73.5	0.7032	0.6503	0.6075	0.3185	-	0.2875
6	58.2	0.7873	0.7179	0.6607	0.3607	-	0.3485
5	54.9	0.7825	0.6976	0.6365	0.3638	0.3177	0.3360
4	53.0	0.7743	0.6976	0.6729	0.3684	-	0.3555
3	37.0	1.0000 ⁺	0.8873	0.8345	0.4205	0.4151	-
2	26.1	0.9476	0.8713	0.8085	0.4051	-	0.3806
1	1.2	0.5165	0.4601	0.4200	-	-	-

* Distance from core center.

** Distance above bottom of core - (core height = 91.2 cm)

+ The specific activity at this point was 230.8 counts/min/mg.

Figure 13. PU-239 Fission Distribution in Rod FB

Figure 14. U-238 Fission Distribution in Rod FB

Figure 15. U-235 Fission Distribution in Rod FB

Figure 16. U-235 Axial Fission Distribution at Core Radial Boundary

Figure 17. U-238 Axial Fission Distribution at Core Radial Boundary

Figure 18. PU-239 Radial Fission Distribution at Axial Position 3

Figure 19. U-235 Radial Fission Distribution at Axial Position 3

Figure 20. U-238 Radial Fission Distribution at Axial Position 3.

the fuel rods. This effect is apparently caused by the sharp decrease in the high-energy flux (and U-238 fissions) in those core regions (such as the fuel gap, central channel, and foil holder rods) that contain no fission source. The difference between the shape of the calculated axial U-238 fission distribution at a radius of 3.17 cm (in the calculational model, this is in a sodium steel central channel region and is ~ 1 cm from the inner radius of the fuel zone) and at 5.97 cm (or ~ 1.7 cm within the fuel zone) is indicated by the fission rate ratios shown in Figure 21. The calculations indicate that this difference in axial shape occurs within a distance of about 1 cm from the fuel region and undergoes little additional change at further distances away from the fuel. Since the shortest distance between fuel rod centers is ~ 2.8 cm, these calculations indicate that corrections should be applied to the calculations of the U-238 fission distribution to account for the absence of a fission source in the foil holder rods. Such corrections were applied using the curve in Figure 21 to obtain the calculated axial U-238 distributions illustrated in Figures 14 and 17. Calculations indicated that no corrections of this nature were required for the U-235 and Pu-239 distributions. Fissions in these isotopes are predominantly the result of lower (< 1 MeV) energy neutrons, which are not so directly influenced by the lack of an in-rod fission source.

The measured relative radial fission rates near the core midplane are shown in Figures 18 through 20, and the axial distributions in the uranium near the radial core boundary are shown in Figures 16 and 17. The fission rates in these regions near the edge of the core are difficult to calculate because of the unusual geometry. The cylindrical calculational model in which the volume of the core was preserved does not provide sufficient detail to accurately represent the geometry in and around the three foil holder rods on the edge of the core. With this model, for example, one of the foil holder rods (FD) is located at a radius (44.5 cm) outside the periphery (44.1 cm) of the core. In addition to geometrical deficiencies in the model, diffusion theory results in a more isotropic angular flux distribution than would be expected in such a region.

The measured values of the reaction rates in all the foils are higher, relative to the rates in the center, than the calculated values near the core reflector interfaces at the upper and lower boundaries of the core. Preliminary comparisons of two-dimensional transport (S_4) and diffusion theory calculations indicate that corrections for transport effects would decrease these disparities by no more than $\sim 3\%$.

The general agreement between the calculated and measured reaction rate profiles throughout most of the core and the apparent (see above) underprediction of the reaction rates near the core boundaries indicate that the power (and, thus, fuel temperature) distribution may be somewhat flatter than calculated. Even if additional calculations do not remove the disparities, the following calculations indicate that the Doppler effect is relatively insensitive to the shape of the flux and temperature distributions. It is expected that the observed disparities in the power distribution will have only a small effect ($\leq 3\%$) on the calculated reactivity coefficients.

4.3.6 Central Fission Ratios

In addition to the data described above, supplementary count rate data have been taken on the three foils near the core center (can number 3, rod FB) to determine approximate foil fission ratios. The 1.6 MeV La-140 fission product emission was counted for the depleted uranium, enriched uranium, and plutonium foils in this can. The enriched uranium and plutonium foils were counted for one hour twenty minutes, and the depleted uranium foil was counted for fifteen hours thirty-seven minutes. The data were recorded on a multichannel analyzer, and the area under the total energy peak of the 1.6 gamma ray was integrated for each foil.

The area under the total energy peak consists of the 1.6 MeV events from the decay of La-140, plus background from other fission products. The background was assumed to have the same shape and magnitude as the spectrum just above the 1.6 MeV peak. This background was subtracted from the area of the total energy peak, and the resulting count rate was then normalized to the foil weight to obtain the specific La-140 activity for each type of foil.

No normalization was made for differences in counting time. The counts were treated as if they had been taken in the middle of the counting interval, and the three count intervals were assumed to be at the same time after shutdown. The first assumption results in less than 1% error, while the second assumption results in less than 2% error (as can be seen by using the daughter product relation):

Activity of La ($\lambda_{La} N_{La}$)

$$= N_{La,0} \frac{\lambda_{Ba} \lambda_{La}}{\lambda_{La} - \lambda_{Ba}} (e^{-\lambda_{Ba} t} - e^{-\lambda_{La} t})$$

for the times in question.

Figure 21. Ratio of U-238 Fission in the Central Channel and Adjacent Fuel Region

The foil fission ratios are as follows after accounting for count length, mass and yield:

$$\frac{\sigma \text{ Depleted U}}{\sigma \text{ Enriched U}} = 0.0290$$

$$\frac{\sigma \text{ Pu Mixture}}{\sigma \text{ Enriched U}} = 0.9215$$

The estimated outside error limit ($\sim 2\sigma$) on these ratios is about 10%. This limit is due to a combination of uncertainties in La-140 yield, counting statistics, background subtraction, weight normalization, and counting geometry reproducibility.

The calculated fission ratios for all the fissionable isotopes were previously calculated and reported in Reference 4. The measured (from La-140 decay) foil fission ratios discussed above can be combined with the foil isotopic content given in Table 15 to obtain isotopic fission cross-section ratios. These measured values are listed, with the previously calculated values, in Table 16. The measured U-238 to U-235 fission ratio in Table 16 is obtained directly from the measured foil fission ratios and the isotopic composition in Table 15. The measured Pu-239 to U-235 ratio of Table 16 was obtained from the measured foil ratios by treating the small amount of Pu-241 (see Table 15) as if it were Pu-239, and by treating the other isotopes as Pu-240. In addition, the calculated⁴ Pu-240 to U-235 fission ratio of 0.206 was used to obtain a correction of 0.013 in the value of 0.905 (see Table 16).

Table 15
FOIL ISOTOPIC COMPOSITIONS

Foil Type	Isotopic Composition	
	Isotope	Percent
Enriched Uranium	U-235	93.15
	U-238	6.85
Depleted Uranium	U-235	0.19
	U-238	99.81
Plutonium	Pu-238	0.036
	Pu-239	92.905
	Pu-240	6.444
	Pu-241	0.582
	Pu-242	0.033

Table 16
FISSION CROSS SECTION RATIOS
NEAR CORE CENTER

Ratio	Measured	Predicted ⁽⁴⁾
$\sigma_f^{238}/\sigma_f^{235}$	0.0252	0.0256
$\sigma_f^{239}/\sigma_f^{235}$	0.905	0.894

The measured and calculated fission cross-section ratios are in very good agreement. Even if the measured U-238 to U-235 ratio is increased by approximately 5% to account for the apparent high-energy flux depression in the foil holder rods, the agreement is still within $\sim 4\%$. However, in view of estimated uncertainty ($\sim 10\%$) in the measurement, it is possible that the true fission ratios are not calculated with the same accuracy as indicated by the table. It is expected that a more accurate measurement can be obtained from the proposed fission track-etch measurement.

4.3.7 Power Distribution in SEFOR

Calculations of the power distribution in SEFOR have been performed with a two-dimensional diffusion code. In Figures 22 and 23 these calculated* results are compared with the power distribution obtained as a composite of the experimental Pu-239 and U-238 fission rate traverses, using the calculated Pu-239/U-238 fission ratio at the core center. The calculated and measured fission ratios are in good agreement at that position (see Table 16). Self-shielding corrections caused by the influence of the foil holder rods and the reflectors and transport effects were estimated.

Because of the difference in atomic compositions of the aluminum-filled foil holder rods and the homogenized core, the neutron spectrum in the foil holder rods is not the same as in the surrounding core. This is because of the following effects: (1) the lack of a fission source in the foil holder rods decreases the number of high-energy neutrons compared to the core; and (2) differences in the fine structure of the neutron spectrum are to be expected in the resonance region of the heavy nuclides in the core below 10 keV.

* Both plots in axial and radial direction are traverses through the point of the maximum power.

Figure 22. Axial Power Distribution

Figure 23. Radial Power Distribution

Both effects were estimated by a two-dimensional diffusion calculation in which the central channel was replaced by a cylinder of aluminum. The only significant (more than 2%) change in the shapes of the fission profiles was in the gap region (see Figure 21). Since there is essentially no power in the gap, no correction for these effects was applied in evaluating the measured power distribution from the measured fission profiles. An additional numerical calculation was performed to obtain an independent estimate of the second effect discussed above, by using special resonance self-shielding factors for Pu-239 in the foil holder rod. This calculation also indicated that the maximum change in the Pu-239 fission rate at the core boundary was only 2%.

The diffusion calculations neglect surface effects on the self-shielding factors for reflected neutrons, because it is assumed that the neutron spectrum near the resonance energies in the core are not influenced by back-scattered neutrons from the Ni-reflectors, even close to the Ni-reflectors. More precise calculations could be performed by using space-dependent self-shielding factors. To estimate the influence of this effect on the power distribution, the power at the core surface was determined by using infinite dilute Pu-239 fission cross sections for calculating the Pu-239 fission rate at the core surface. This increased the power near the radial reflector by 14%, and near the upper and lower axial reflectors by 7% compared to the case, where the infinite medium shielding factors were used everywhere.

To estimate transport effects, two-dimensional diffusion and S_4 calculations with four energy groups have been compared. These calculations gave only a 3% higher fission rate at the radial core surface than the diffusion calculation.

All above-mentioned effects influence only the power distribution close to the reflectors. Figures 22 and 23 show theoretical values at this location which include these corrections.

The influence of changes in the power profiles on the Doppler effect and the fuel temperatures was checked. Both the Doppler effect and the core temperatures were calculated with the WFDOP⁷ code, using both a theoretically predicted power distribution and a power distribution with 20% higher values at the core surface, but with the same total power. This decrease of peak to average power density decreased the Doppler effect at 20 MW by 3%. At the same time, the maximum temperature in the fuel decreased by 15% and the minimum temperature increased by 20%.

4.3.8 Zero Power Oscillator Experiment

The primary objective of the zero power oscillator experiment was to determine the power-reactivity transfer functions in the critical reactor at a power level low enough for the reactivity feedback effects to be considered as negligible.

During the experiment, the reactor power level was maintained at $\approx 1\text{kW}$, and the poison rod was oscillated at 11 different frequencies ranging between 1.56 and 0.05 rad/sec (see Table 17).

The zero power oscillator experiment consisted of a dynamic calibration of the oscillator rod, followed by the measurement of the background noise and the power-reactivity transfer functions.

The dynamic calibration of the poison rod was carried out at the highest frequency attainable, namely $\omega_0 = 1.56$ rad/sec, corresponding to a period of ≈ 4 sec. The calculated value of the transfer function at this frequency, $G(\omega_0) = 1.125$ was used to obtain the desired measurement of the oscillator rod worth. The measurements were made at three different locations in the core (at a distance of 4 inches, 6 inches, and 8 inches, respectively, from core middle) and each time with two different amplitudes of the oscillations (2-inch and 4-inch). The results obtained have been compared and plotted (Figure 24) with the static value of the rod worth at these locations. The discrepancy

Table 17
FREQUENCIES FOR THE ZERO POWER
OSCILLATOR EXPERIMENT

Designation	Frequencies	
	Rad/sec	Cycles/sec
F-2	≈ 1.56	≈ 0.25
F-3	0.98	0.156
F-4	0.61	0.097
F-5	0.35	0.056
F-6	0.22	0.035
S-1	0.13	0.021
S-2	0.078	0.0125
S-3	0.049	0.0078
S-4	0.031	0.0049
S-5	0.018	0.0029
S-6	0.011	0.00175

The number of cycles varied between 16 cycles in the high frequencies down to 5 cycles for the lowest.

Figure 24. Dynamic Calibration of the Oscillator Rod Worth

between the static and dynamic measurements amounts to 10%; no satisfactory explanation has yet been found for it. Recalibration of the poison rod is planned, using the "period" method (the oscillator rod was previously calibrated by "swapping" it against a reflector control segment, with reactor power constant before and after the swap).

The zero-power transfer functions have been measured with the mean position of the oscillator rod at a distance of 5.5 inches from core middle. Three series of measurements have been made, each covering the whole range of frequencies available, and only differing from one another in the amplitudes of the oscillations of the poison rod. In the first two series of measurements, the rod was oscillated with an amplitude of 4 inches, resulting in a power level variation of about 10%. In the last series, the power level variation lay between 8% and 2%.

Each series of measurements, except the last one, was preceded by a measurement of the background noise at all the frequencies. These measurements were to be made with zero amplitude oscillations, but a slight oscillation of the rod was observed. This problem was solved by analyzing the higher harmonics of the noise signals. The amplitude of the background noise was found to be independent of the frequency and to alternate in sign, depending on the number of cycles and data set analyzed. The maximum magnitude of the noise was of the order of 0.2% of the mean power level. The true mean value of the noise is thus expected to be zero. The magnitude of the deviations about this mean was used to estimate uncertainties in the data. Thus, the contribution of the noise to the uncertainty in the measured signals for the transfer functions amounts to 2% in the most favorable case and goes up to 10% in the last series of measurements.

In the analysis of the data, the value of the transfer function was determined using the results of the dynamic measurement of the rod worth. This corresponds to a normalization of the results to the calculated value at the highest frequency. The measured value of the amplitude and phase angle of the transfer functions, with their calculated values, are plotted in Figures 25 and 26.

The measurements are being fitted, using a simplified expression similar to that given in Reference 8 for the inverse of the zero power transfer functions, to obtain an estimate of the fraction of fission neutrons produced by U-238 in SEFOR.

Approximate Power Coefficient of Reactivity

The SEFOR reactor was heated to approximately 1 MW on March 5, 1970. Reflector positions, as well as average core temperatures and power levels that occurred

during this operation, are listed in Table 18. The power and temperature data in Table 18 were calculated from the observed* primary coolant system flows and temperatures shown in Table 19.

In order to separate the reactivity effect of the increased power level from the reactivity effects associated with changes in average core coolant temperature, several zero power observations* during March were averaged, as indicated in Table 20.

The estimated uncertainty in the temperatures is $\pm 2.5^\circ\text{F}$, while the estimated uncertainty in the indicated nonzero power levels is in the range ± 0.1 to ± 0.15 MW. Thus, at zero power (when there is no uncertainty in power) the expected standard deviation in a reactivity measurement can be estimated using the measured temperature coefficient of $-0.58\text{¢}/^\circ\text{F}$ and the previously determined** standard deviation in a reactivity measurement of $\pm 0.35\text{¢}$ according to the following expression:

$$\sigma_1 = \sqrt{(0.35)^2 + (0.58)^2 (2.5)^2} = 1.5\text{¢}.$$

At power, the estimated standard deviation, using the indicated power coefficient of $-19.5\text{¢}/\text{MW}$ and an uncertainty in power of ± 0.1 MW, would be:

$$\sigma_2 = \sqrt{(1.5)^2 + (19.5)^2 (0.1)^2} \approx 2.5\text{¢}.$$

This value (2.5¢) is an estimate of the standard deviation that would be observed in a large number of reactivity measurements at power levels up to 1 MW. It does not necessarily reflect the accuracy of the present limited amount of data.

A least-squares fit of a straight line to the reactivity (K_i) and nonzero power (P_i) data in Table 18 gives a power

* Although the temperatures in Table 19 are taken from the data acquisition system (DAS), while those at zero power in Table 20 are taken from the resistance temperature detector (RTDs), the inlet temperatures indicated by the two detection methods agree within the measurement accuracy of $\pm 2.5^\circ\text{F}$. The RTD outlet temperature was 2°F to 6°F lower than the DAS outlet temperature.

** In the previous measurements, a temperature uncertainty of $\pm 0.25^\circ\text{F}$ (not $\pm 2.5^\circ\text{F}$) was taken into account (see Reference 6).

Figure 25. Amplitude of SEFOR Zero Power Transfer Function

Figure 26. Phase of SEFOR Zero Power Transfer Function

Table 18
POWER REACTIVITY DATA

Reflector ⁺⁺ Position (cm)	No. 10 Worth (ρ)	Reflector ⁺⁺ Position (cm)	No. 3 Worth (ρ)	Average Core ⁺ Temperature (°F)	Power ⁺ (MW)	Core Reactivity Change (ρ)
45.0*	66.07*	-	59.25*	737*	0	0
44.4	64.89	49.48	75.15	737	0.78	-14.7
45.0	66.07	51.18	78.37	737	0.96	-19.1

* This represents averaged zero power data (see Table 20).

+ The average core temperature and the power (at 0.78 and 0.96 MW) were determined from the flow and temperature data in Table 19.

** This represents the change in core reactivity as the power is increased from zero and the average core temperature is maintained at 737 °F.

++ Reflector No. 8 was at 33.00 cm and all other reflectors were up. The calibration curve for reflector No. 8 was used to estimate the reactivity effect of changes in the position of reflector No. 10.

Table 19
HEAT BALANCE DATA

Main Primary System			Auxiliary Primary System			Power*	Average Core* Temperature (°F)
Flow (gal/min)	Inlet ⁺ Temperature (°F)	Outlet ⁺ Temperature (°F)	Flow (gal/min)	Inlet ⁺ Temperature (°F)	Outlet ⁺ Temperature (°F)		
518	729	751.5	184.3	703	750	0.78	737
527	709	754.5	192	749.3	754.3	0.96	737

* Power levels were obtained from a heat balance. The average core temperature is the flow weighted average of the temperatures in the two systems. The estimated uncertainty in temperature is ± 2.5 °F. The estimated uncertainty in power is ± 0.1 to ± 0.15 MW.

+ These temperatures were recorded on the data acquisition system. The 2° F to 6° F difference between inlet and outlet temperature that was observed on the RTD's at zero power (see footnote in Table 20) was not observed here.

Table 20
ZERO POWER DATA

Date	Reflector ⁺⁺ Position (cm)	No. 10 Worth (¢)	Reflector ⁺⁺ Position (cm)	No. 8 Worth (¢)	Reflector ⁺⁺ Position (cm)	No. 3 Worth (¢)	Reactor Inlet ^{**} Temperature (°F)	Relative ⁺ Reactivity at 737°F (¢)
3/4/70	45.0	66.07	33.00	42.85	28.59	35.42	698*	58.04
3/5/70	45.0	66.07	33.00	42.85	29.40	36.87	700	58.33
3/6/70	45.0	66.07	33.00	42.85	20.39	21.63	669	61.07
3/7/70	45.0	66.07	33.00	42.85	24.69	28.56	686	58.14
3/17/70	48.0	71.91	50.00	75.78	28.42	35.11	761	59.96
3/17/70	48.0	71.91	50.00	75.78	28.40	35.08	761	59.93
Average =								59.25

* This is not a recorded inlet temperature. The recorded outlet temperature has been increased by 4°F to account for observed 2°F to 6°F lower outlet temperatures on the RTD's.

** Temperatures obtained from resistance temperature detectors (RTD's).

+ These relative values represent the worth of the raised portion of reflector No. 3 when reflector No. 10 is at 45.0 cm and No. 8 is at 33.00 cm. Temperature corrections were made using the measured coefficient of -0.58¢/°F.

++ All other reflectors were up. The calibration curve for reflector No. 8 was used to estimate the reactivity effect of changes in the position of reflector No. 10.

coefficient (A) of -19.5¢/MW. In order to test for the "goodness of fit", a quantity, chi-squared, defined by

$$\chi^2 = \frac{\sum_{i=1}^2 (K_i - AP_i)^2}{\sigma_2^2} = 0.064,$$

$$\text{where } A = \frac{\sum_i K_i P_i}{\sum_i P_i^2}$$

was calculated. A chi-squared table shows that for one degree of freedom (which is the case here) there is a 20% probability of obtaining a value of chi-squared as small as this. This may indicate either that the fit is a "little too good" or that the estimate of the standard deviation (σ_2) is too large. However, a 20% probability for χ^2 is generally considered acceptable and neither the fit nor the estimate of σ_2 should be rejected.

Since both the fit and the estimate of the standard deviation appeared to be acceptable, the following expression was used to obtain an 80% confidence interval for the coefficient A:

$$A = -19.5 \pm \frac{\sigma_2}{\sum_i P_i^2} t_{80} = -19.5 \pm 2.0 t_{80} \text{ ¢/MW}$$

$$= -19.5 \pm 6.2 \text{ ¢/MW},$$

where $t_{80} = 3.1$ was obtained from Fisher's table. Since there is only one degree of freedom, the confidence interval becomes much larger as the degree of confidence is increased. For example: the 90% confidence interval is $-19.5 \pm 12.6 \text{ ¢/MW}$.

The calculated value of the power coefficient between zero and 1 MW is -16.5¢/MW. This was calculated from the following expression:

$$\frac{-260}{1160} (65) (1.1) - 0.5 = -16.5 \text{ ¢/MW}.$$

The value -260 is the calculated Doppler coefficient (-0.0082) in cents, 1160 is the sodium temperature in degrees Rankin (°R), 65 is the expected rise (in °R) in effective fuel temperature per megawatt power change, and the factor 1.1 reflects the expected increase in Doppler coefficient due to spatial weighting. The value -0.5 is the expected clad and fuel axial expansion power coefficient of reactivity.

5. TASK 11—TRANSIENT TESTS

5.1 PURPOSE

The purpose of Task 11 is to investigate the behavior of a $\text{PuO}_2\text{-UO}_2$ fueled fast reactor for reactivity changes less than 1 dollar and compare the experimental results with theoretical calculations. The types of tests to be conducted at various power levels include static, oscillation, and subprompt-critical transient reactivity changes.

5.2 PAST PERFORMANCE

The San Francisco Operations Office of the AEC authorized rental of a cask intended for the shipment of selected SEFOR fuel rods to a "hot lab" for destructive examination. The cask has been shipped to the SEFOR Site for storage.

The license application for AEC and DOT authorization to use cask B of E 1233 for shipment of irradiated

SEFOR fuel rods has been completed and is being reviewed.

Detailed drawings were made for shielding around the vertical transfer value in the SEFOR refueling cell, and were sent to the SEFOR Site for comment. The shielding is designed as a semipermanent structure that need not be removed after each fuel transfer operation. Use of the full valve opening will require the removal of a portion of the shield assembly.

5.3 CURRENT RESULTS

Transfer equipment required for fuel transfer was sent to the SEFOR Site for storage.

The license application for AEC and Department of Transportation authorization to use the shipping cask has been submitted to these organizations for approval.

6. SEFOR SAFETY AUDIT

A comprehensive review of recommendations made in audit reports during the past year was completed. This review showed that action had been completed on 75% of these items and that actions in response to the remainder are in various stages of completion.

Review of sodium pump performance and flow meter calibration data showed that a temperature correction must be applied to the flow meter. This correction is similar to results found during testing of other large permanent magnet flow meters.

The failure rate for thermocouples in the reflector guide structure was found to be excessive. Since these thermocouples are not accessible for repair or replacement,

a recommendation was made to install substitute thermocouples, which could be replaced, if necessary, to assure adequate monitoring of reflector temperatures. Thermal-hydraulic analyses were made to determine an acceptable location for these thermocouples, and a position in the lower end of each reflector segment was selected. Installation of these thermocouples is now in progress. Additional replaceable thermocouples will be installed to measure the temperature of the coolant gas above the reflector segments. They will provide backup instrumentation for the thermocouples originally installed in this location to obtain engineering data on heat generation and cooling capability in the reflector region.

7. SEFOR SUPPORT

The plant engineering effort is a supporting function to the SEFOR facility.

Reactor power operation was initiated early in March, 1 MWt power was achieved, and satisfactory performance of the heat transfer systems was obtained. However, during the power ascension program, a number of reflector guide structure thermocouples were noted to be failing. These thermocouples are installed within the guide structure and the reflector segments to monitor thermal conditions in this area. Fifty thermocouples are installed in the guide structure, and ten of these are connected to the reactor safety system. The failure of a substantial number of these thermocouples required the development of replaceable instrumentation designed to obtain reliable temperature information from the reflector region. The left side of Figure 27 illustrates the thermocouple installations and routing of lead wire for the existing guide structure thermocouples, enclosed in a 1/16-inch sheath and imbedded in the aluminum guide structure. The sheathed portion extends out of the high-temperature region and terminates in the lower shield region with a connection to the glass insulated thermocouple leads.

The installation of additional thermocouples in the lower end of each reflector segment (shown on the right side of Figure 27) is now underway. The new thermocouples are ungrounded, with magnesium oxide insulation and a 1/16-inch-diameter 304 stainless steel sheath. There are no sharp bends in the sheathed portion of the thermocouple, and all of the glass insulated lead wire is accessible for inspection and repair, if required. The thermocouples can be replaced, if necessary, from the reflector drive cell. Three thermocouples are installed in each segment to provide two spares for each segment, which will reduce the need for cell entry should failure occur.

The sheathed portion of these thermocouples extends down through a guide tube in the hollow extension shaft to a connector just above the coupling nut on the drive shaft. The guide tube is continuous from the nickel segment to the shaft coupling nut, so that the thermocouples can be replaced, if necessary. In order to provide this feature, it was necessary to fabricate replacement extension shafts for each of the ten segments. Extension shafts were fabricated and delivered to the site, with the necessary tool and fixtures required to complete the modification.

The reactor vessel flow-level anomaly was corrected, following installation of the drywell into the center position in the reactor vessel. Tests completed after the installation of the drywell indicated that the vessel level did not fluctuate with variations in reactor flow, as was experienced previously. The level anomaly was determined to be the result of an abnormal reactor flow distribution caused by the temporary absence of the drywell and orifice from the center channel.

Excessive carryover from the refueling cell argon purifier was discovered during operation of the system, requiring entry into the cell for cleanup and removal of the residue. Design and procurement of a separator was initiated to eliminate gross carryover from the purifier.

The two remaining type II reactor level probes were completed and sent to the site. These probes will constitute backup spares for the level probes now in service.

A malfunction occurred in the reactor vessel through-head tip probe (type I). Following removal of the probe, the malfunction was determined to be caused by an internal open circuit condition. This was corrected, and the probe was returned to operation.

Figure 27. Reflector Thermocouple Modification

8. SEFOR OPERATION

8.1 EXPERIMENTAL PROGRAM

After installation of the two instrumented fuel assemblies (VNC-25 and VNC-29), the reactor vessel head was reinstalled and the through-head fuel was reloaded into the core.

The second fine Reflector No. 3 was calibrated differentially and the eight coarse reflectors were measured by "rod swapping" against either No. 3 or No. 8. This provided a complete calibration of the reflector system at 350°F for this core configuration (I-H or I-I are essentially the same).

That portion of test procedure TP I-D.1 (Flow Coefficient Measurement) above a flow rate of ≈ 4000 gpm was completed. The earlier attempt to complete these measurements was terminated because of low operating sodium level trips at the higher flow rates; i.e., more sodium was leaving the vessel than was being made up by the pump-around-pump at the < 1 gpm technical specification limit, for main primary flows above 4000 gpm, resulting in low sodium level trips. Loss of sodium from the vessel was attributed to the flow of sodium above the head shield plates and into the overflow because of the absence of either a shield plug or drywell in the center channel (see Section 8.2). In order to complete the test, special permission to raise the pump-around-pump limit was obtained. The results of the flow coefficient tests above 4000 gpm were essentially the same as those obtained at lower flow rates; i.e., no observable flow induced reactivity effect over the entire flow range, within the uncertainty of the measurement.

The center channel orifice and standoff were installed in the core, and the positioner-drywell combination was mounted on the reactor head. The measured reactivity effect from the installation of the drywell was $+2.6 \pm 1.4\%$ (temperature uncertainty).

Following completion of core loading to the full power configuration, and subsequent installation of the reactor head, diagnostic tests were performed which showed that the sodium level in the reactor vessel remained constant at all coolant flow rates. Previous tests, which had shown contradicting results, were reviewed and it was found that the previously observed level change at high flow rates had occurred prior to installation of the drywell in the center channel. Also, the orifice for the center channel was not installed on the limited number of occasions when the coolant flow rate was increased and level changes were observed. Thus, the level changes were

caused by coolant system operation with an intermediate configuration during the planned core loading procedure. Since tests performed with the final configuration showed no problem with level control, further corrective action is not required.

After replacement of the oscillator amplitude motor which had malfunctioned, the linear variable transducer (poison rod position indication) was recalibrated and the oscillator mechanism reassembled. The oscillator mechanism was transferred into the refueling cell and mounted on the positioner-drywell combination.

The venturi flowmeter, which is installed in the main primary coolant loop, was recalibrated (Δp sensing system) and flow and output data for the venturi and main EM flowmeter were taken over the full flow range at loop temperatures of $\approx 400^\circ\text{F}$. Previous calibration data had been taken during the hydraulic preoperation tests (March 28, 1969), but a failure and replacement of the oscillator-amplifier of the Barton differential pressure-sensing system required a calibration for the venturi data to be of value. In a comparison of the present and previous calibration data at 400°F , a disparity was noted with a magnitude of approximately 4% at full flow. The accumulation of additional data on both venturi and EM flowmeter is planned in an attempt to discover an explanation for the disparity.

A comparison of the data from the cross calibration of the venturi and EM flowmeter at 750°F showed the temperature dependent, nonlinear high flow rate (> 3000 gpm) behavior associated with large, center tap, EM flowmeters.⁹

The reactor was held critical during heatup from 350°F to 760°F to verify the temperature coefficient measured earlier. Good comparison was obtained with earlier data.

The temperature was stabilized temporarily at 550°F and the fine Reflector No. 8 was calibrated differentially, then heatup continued to 760°F with the reactor held critical. At 760°F both fine Reflectors No. 8 and No. 3 were calibrated differentially, which completed the anticipated calibrations of the fine reflectors over all conditions expected to be encountered in the experimental test program.

With the completion of the calibration of the fine reflectors at 760°F (oscillator poison rod calibrations), TPI-D.4 was initiated. This procedure included a static and a dynamic calibration. In the static calibration, the incremental reactivity worths of the poison rod were obtained from the movements of the calibrated fine reflector

required to maintain criticality as the poison rod was lowered into the core by the positioner mechanism. The measured integral worth of the poison rod with the tip inserted to the core centerline was 33¢. In the dynamic calibration, the poison rod was oscillated at a fixed frequency (0.24 Hz) and two amplitudes (~2 inches and 4 inches) at various locations in the core (4 inches, 6 inches, and 8 inches above the core centerline), and the core power response was measured. The differential values obtained from the dynamic calibration were in reasonable agreement with those from the static calibration. The oscillator experienced a number of difficulties, which were corrected, including: frequency of oscillation indicated by the torque transmitter and linear transducer and uniform speed control for the oscillator drive motor through the cycle.

After completion of TPI-D.4, the zero power frequency response (TPI-D.5) was completed, taking three full sets of data at all required frequencies.

While diagnosis and repair of the oscillator mechanism were in progress, reactor power was increased to 1 MW and testing resumed. The object of the testing at 1 MW was to investigate the first plant thermal performance and the low-power operating characteristics of both main and auxiliary air blast coolers, and to obtain the first thermal calibration of the nuclear instrumentation.

At a nominal power of 1 MW, a plant heat balance was obtained and radiation surveys made throughout the plant. Table 21 is the result of the plant heat balance. Considering the relative inaccuracies of low-power heat balances, the heat balances on the various components at the nominal 1

MW power level show good agreement. The reactor vessel instrumentation is considered to be the most accurate, since the temperature measurements are obtained with calibrated resistance temperature detectors (RTDs). Temperature measurements in other portions of the plant are "as received" chromel-alumel thermocouples with less accuracy and readout precision.

From the operation at 1 MW, it was determined that the auxiliary air blast cooler is capable of rejecting up to 750 kW(t) with natural air circulation through the cooler. The main air blast cooler is capable of rejecting at least 1 MW(t) (probably more) with natural air circulation through the cooler. The wide range monitors (WRM) and intermediate range monitors (IRM) were adjusted to correspond to the heat balance data at 1 MW. Prior to power increase, it was determined that the zero power critical point ($50 \times 10^{-5}\%$ WRM reading) corresponded to 600 watts, as indicated by the foil activation measurements. This value showed very good linear extrapolation agreement at 960 kW $85 \times 10^{-2}\%$ WRM reading).

Results from the radiation surveys made at 1 MW indicate that, with minor additions, the shielding of the plant will be adequate for 20 MW operation. Sheets of polyethylene have been fabricated for installation at the refueling cell crane bay access to reduce the neutron dose to less than 10 mRem at full power. A similar removable shield has been designed for the reflector drive room doorway.

When the zero power frequency response (TPI-D.5) was completed, further testing was discontinued until the reflector guide structure thermocouple malfunctions were corrected.

Although the two instrumented fuel assemblies had been installed in the core prior to the rise to 1 MW, the constant temperature reference junction was faulty and had been returned to the vendor for repair, so that no IFA data were taken during the 1 MW testing. The constant temperature reference junction has been subsequently received and connected. All temperature indications are functioning properly, but the flow indication from VNC-29 has been lost. It is postulated that the leads have leaked sodium, causing a short between the leads at the flowmeter, which results in a zero flow indication. A routine surveillance has been established to maintain current IFA functional status.

The Karlsruhe representative spent two weeks at SEFOR checking the automatic balanced oscillator instrumentation and inspecting for transportation damages. The test procedures for the balanced oscillator experiments of both the first and second kind (constant core average temperature and constant power) were drafted and issued for comment.

Table 21
SEFOR PLANT HEAT BALANCE
AT NOMINAL 1 MWt

	Main Loop MW(t)	Auxiliary Loop MW(t)	Total MW(t)
Reactor Vessel	0.92	0.04	0.96*
IHX Primary	0.86	0.04	0.90**
IHX Secondary	0.78	0.02	0.80**
Air Blast Cooler	0.80	0.08	0.88**

Average Wide Range Monitor Reading - 85×10^{-2}
Vessel $\Delta T = 45^\circ \text{F}$
Vessel Flow = 717 gpm

* Estimated Uncertainty = $\pm 10\%$

** Estimated Uncertainty = $\pm 15\text{-}20\%$ Due to Less Precise Instrumentation

The off-line Fourier analysis computer programs that had been written for use at the SEFOR Site were utilized to reduce the zero power frequency response data. These data will be utilized in the Nyquist diagram calculations during the power ascension phase for stability determinations. Nyquist plot predictions, using calculated transfer function data¹⁰, have been prepared using the Nyquist plot computer code. These predictions have been made for 5, 10 and 20 MW and are shown in Figures 28, 29, 30, and 31. Figure 28 shows the full Nyquist diagram obtained from theory. However, for practical purposes, the only portion of interest is the third quadrant up to and including the point at which the plot crosses the unit circle (indicated by the dashed box). This is the prediction portion shown in Figures 29, 30, and 31.

With testing temporarily discontinued, emphasis was shifted to preparation, planning, and checkout of equipment for the excursion tests. An out-of-cell preoperational test procedure was written and approved for the fast

8.2 OPERATION AND MAINTENANCE

The reactor was secured on March 20, 1970 in preparation for an outage to repair and/or replace the reflector guide structure thermocouples. The work is in progress at the end of this report period.

Operating Data

Reactor Critical	192.4 hours
Maximum Power Level	1.0 MW Nominal
Longest Continuous Run	76.9 hours

Reactor Scrams

Equipment	10
Personnel	2
Manual	<u>2</u>
Total	14

Two significant events were achieved in March: (1) the longest continuous run to date (76.9 hours) was sustained intentionally and was terminated at the completion of the program test procedure; (2) the reactor was operated at 1 MW with the air blast coolers in service for a new high power level for SEFOR.

Altogether 142 malfunctions were corrected and distributed as follows:

Mechanical	50
Electrical	30
Instrumentation	62
Total	142

Reactor operation was delayed by malfunctions of the reactor sodium level probes, a short to ground on electrical cables to sodium pipe heaters, malfunctions of the oscillator mechanism, and an accelerated rate of failure of the reflector guide structure thermocouples.

An increase in sodium temperature corrected the malfunction of one "J" probe (variable level indicator). Correction of the malfunction of the other "J" probe was accomplished by removal and cleaning of the probe. Correction of the malfunction of one through-head, operating level, low level probe was corrected by repairing an open electrical lead within the probe assembly.

An entry was made into the primary sodium equipment area with air breathing masks to identify the location of the short to ground on heater circuits 10D and 10E, and to measure the radiation field. The atmosphere was changed to air, the repairs were made, and the atmosphere returned to nitrogen.

A number of malfunctions of the oscillator were corrected. A difference in frequency of the oscillator, indicated by the torque transmitter and the linear transducer, was identified and corrected. Two set screws in the shaft coupling between the torque transmitter and oscillator wheel became loose. Because of the low torque requirements of the transmitter, the friction between the shaft and coupling was sufficient to provide an eccentric friction drive that produced a transmitter frequency 5% greater than produced by the oscillator mechanism. Relocation of the oscillator motor controller from the control room to the reactor building eliminated the resistance loss of the long cables and permitted proper control of motor speed throughout the cycle. The addition of counter balance weights for the poison rod further improved the wave form generated by the oscillator.

A design for replacement reflector guide structure thermocouples was completed, material procured and installation initiated.

Other significant modifications of equipment and maintenance activities included installation of:

- Shielding plugs in the transient and gamma detector tubes
- Additional evacuation alarms
- Main ABC remote vane control
- Communication station at the crane bay access
- Reference junction for instrumented fuel assemblies
- Emergency diesel enclosure door alarm
- Emergency diesel enclosure louvre position lights
- Emergency diesel overspeed trip alarm
- Reactor sodium overflow valve closed alarm
- Pump-around-pump flow alarms

Figure 28. Nyquist SEFOR Diagram Plot

Figure 29. Nyquist Plot Prediction for 5 MW-30% Flow

Figure 30. Nyquist Plot Prediction for 10 MW-50% Flow

Figure 31. Nyquist Plot Prediction for 20 MW-100% Flow

- Temperature monitor and alarm for gas-filled sodium piping and components
- Crane bay access neutron shielding and replacement of:
 1. Reactor building freon compressor valves and springs
 2. Freon compressor motor windings
 3. Emergency diesel battery charger transformer
 4. Sodium pipe heater circuit volt-pack core
 5. Freon condenser fan motor
 6. Argon blower belt sheaves

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