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**SOUTHWEST EXPERIMENTAL
FAST OXIDE REACTOR
DEVELOPMENT PROGRAM
TWENTY-FIFTH QUARTERLY REPORT
MAY-JULY 1970**

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

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AEC Research and
Development Report
August 1970

**SOUTHWEST EXPERIMENTAL FAST OXIDE
REACTOR DEVELOPMENT PROGRAM**

**Twenty-Fifth Quarterly Report
May-July 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 May 1 through July 31, 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 PuO_2 - 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).
 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).
 23. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Third Quarterly Report, November 1969–January 1970," GEAP-10010-23 (February 1970).
 24. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Fourth Quarterly Report, February–April 1970," GEAP-10010-24 (May 1970).
- 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).
 4. Prntsik, 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).
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).
30. Ikeuye, D. T. and Young, F. E., "Fabrication of Non-Fuel Core Components for SEFOR," GEAP-13595 (April 1970).
31. Meyer, R. A., et al., "Design and Analysis of SEFOR Core I," GEAP-13598 (June 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

Negotiations with Nuclear Fuel Services Incorporated (NFS) for new and replacement fuel rods were completed with the signing of Revision 4 to the original purchase order on June 19, 1970. Fuel fabrication activities were started in July 1970.

A quality assurance program was initiated by GE to monitor the fuel rod production at NFS. A GE Quality Control Engineer was assigned and has taken residence at NFS plant at Erwin, Tennessee.

A spare IFA flowmeter was tested to explain the malfunction of the VNC-29 flowmeter. The test did not indicate a common mode of failure for the IFA flowmeters.

Repair work was begun on the damaged rod G7-1 and preparations are being made to build a new rod to be used if the repair is unsuccessful.

2.3 TASK 10—STARTUP TESTING

Preliminary analysis of results obtained during initial operation at power levels up to 5 MW indicate that the observed reactivity effects and fuel temperature changes are consistent with the calculated Doppler reactivity coefficient

$$\left(\frac{dk}{dT}\right) \text{ of } -0.0085.$$

Thermocouples were used to measure fuel temperature changes at seven different core locations. These measurements were used to estimate an average core temperature for use with a simple logarithmic model to correlate changes in Doppler reactivity with fuel temperatures. The results obtained with this simple model and the predicted Doppler coefficient of -0.0085 are in good agreement with the measured "Doppler feedback" of -75 cents for a change in power level between zero and 5 MW at a constant coolant temperature of 760°F .

The ratios of the observed fuel temperature changes in the different core locations are in agreement with a power distribution that was obtained from two dimensional diffusion theory calculations by applying corrections for transport and fuel resonance shielding effects near the core surfaces.

A theoretical model for the inverse of the zero power reactivity transfer function was fitted to the experimental test data. The good agreement between prediction and experiment which was previously⁹ reported afforded the opportunity for this fit. The primary purpose of the analysis was to obtain a "best" analytical model for use during the later determination of reactivity feedback effects from the oscillator tests at power. Such a model is desirable since it is applicable at any frequency, while the experimental data can only be used at those frequencies where the zero power oscillator test was performed. A secondary purpose of the fitting was to obtain an estimate of the effective delayed neutron fraction β . The "best" estimate of β is 0.00327 ± 0.0001 . This is in good agreement with the previously calculated value of 0.00315 .

The power reactivity transfer function has been determined from the conventional oscillator test data taken at 2 MW. The data is now being analyzed to determine the reactivity coefficients and frequency response parameters at this power level.

The stroke of each fine reflector was reduced approximately 3 cm (out of a hundred cm stroke) when the reflector thermocouples were repaired. The fine reflectors were recalibrated to determine whether the slight reduction in stroke had altered previous calibration curves. Within the accuracy of the calibrations ($\approx 1\%$) no change was observed in shape of the worth versus position curves, and only a slight (1 to 1.5 cent) reduction in total worth resulted from the modification.

2.4 TASK 11—TRANSIENT TESTS

General Electric Co. has received authorization from the AEC to rent a fuel transfer cask from Atomics International. The cask will be used to ship used fuel rods from SEFOR to a "hot lab" for examination. The cask (B of E 1233) has been received, inspected and cleaned by GE, and is currently being stored at the SEFOR Site.

A draft of the license application including the structural and thermal analysis has been completed and sent to the AEC for approval.

The design of the fuel transfer equipment and the shielding associated with the fuel transfer has been completed. Fabrication of the fuel transfer equipment has been completed and the items were sent to the SEFOR Site for storage. Fabrication of the shielding is approximately 50% completed.

2.5 SEFOR SAFETY AUDIT

One change in the Technical Specifications was approved by the AEC-DRL and four other changes are under consideration.

Data from engineering tests of the reflector drives, FRED, and plant auxiliary systems were reviewed to assure proper operation of all equipment.

2.6 SEFOR SUPPORT

During this report period, reactor power operation was resumed following completion of the reflector guide structure thermocouple installation and satisfactory performance of the annual containment leak test. Reactor power was raised to 2 MW for the plant heat balance and then 5 MW power operation was achieved for continuation of the experimental testing program.

2.7 SEFOR OPERATION

During the plant outage that began on March 20, because of reflector guide structure thermocouple malfunctions, a replacement reflector segment temperature monitoring system, as auxiliary sodium leak detector system for the Auxiliary Primary IHX return to reactor vessel loop was installed, the annual inner and outer containment leak tests were performed and other equipment modifications and malfunction corrections were accomplished.

Reactor operation and experimental program testing was resumed on June 26, at power levels of 1, 2 and 5 MW. Preliminary results of tests indicated good agreement with those predicted and reactor power stability was demonstrated. Preparations for balanced oscillator and FRED tests were initiated.

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 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. Plans are being made to repair the IFA fuel rod damaged during fabrication.

3.3 CURRENT RESULTS

3.3.1 Fuel Rod Procurement

Negotiations with Nuclear Fuel Services (NFS) Erwin, Tennessee were completed on June 19, 1970 with the

signing of Revision 4 to the original purchase order. This revision provides that NFS will:

- fabricate 55 new fuel rods
- repair 22 He-leaking fuel rods
- replace 47 out-of-specification fuel rods
- supply 20.5 Kg of loose fuel pellets

Fuel fabrication activities were started at NFS. Materials for the inert parts of the fuel rods have been purchased and the preproduction activities such as weld qualification were started. Five lots of plutonium nitrate were received at NFS. The make up of the plutonium-uranium nitrate master mix solution was started.

Thirteen lengths of 1-1/6 inch diameter 304L stainless steel bar stock supplied by NFS, that will be used in the fabrication of fuel rod end plugs, were ultrasonically inspected at the GE San Jose plant. The objective of the ultrasonic inspection of this bar stock was to locate non-metallic inclusions in the bar that might result in a leak path through the wall of the machined end plug.

The ultrasonic inspection was done per GE Specification E5DYP3 which includes the fabrication of a standard defect bar from the material to be tested. One of

the bars had ultrasonic defect indications that were significant in comparison with the standard defect bar and it was quarantined so it would not be used for end plug fabrication. A section of this bar with the largest defect indication was cut off and examined metallographically to try to locate the defective area. Initial evaluation of the metallographic samples show strings of non-metallic inclusions that are heaviest in the defective areas located by the ultrasonic inspection.

3.3.2 Quality Assurance

A quality assurance program was started by GE to monitor the fuel rod fabrication effort at NFS. A quality assurance audit plan outline was published. This plan provides a method to audit the production process through the review of incoming material analyses and production records. Further verification of product quality is made through selection and analysis of samples during the preproduction and production phases of the order.

The quality assurance program was started and a QC Engineer who is the residence GE representative has taken residence at the NFS Erwin plant.

3.3.3 Instrumented Fuel Assemblies (IFA)

A test of a spare flowmeter was made at the IPO Shop in an attempt to discover the reason for malfunction of the VNC-29 flowmeter in SEFOR. The test temperature cycles were from 70° to 800°F and the helium leak check was continued through the test period. No leak was found and the test was concluded. The test indicates there is not a common defect in the flowmeter, which would cause a failure at the reactor temperature.

Before and after the test, the lead wire resistance and the magnetic flux field were checked and no change was observed.

Repair work was begun on the damaged fuel rod G7-1 and preparations are being made to build a new rod. This parallel path will be followed until the success or failure of the repair is known. The decision to start building a new rod was made to avoid losing time if the repair is unsuccessful. It is estimated the successfulness of the repair will be known by late September.

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

Analysis of all experimental data collected during performance of the experimental program at zero power was completed except for the zero power oscillator tests, and some of the measurements associated with the determination of the ratio (λ/β) of the prompt neutron lifetime to the delayed neutron fraction. A preliminary estimate of the power reactivity coefficient from data taken during initial operation at a power level of ≈ 1 MW compared favorably with the calculated coefficient.

4.3 CURRENT RESULTS

4.3.1 Reactivity Effects and Fuel Temperatures at Powers to 5 MW

Preliminary analysis of results obtained during initial operation at power levels up to 5 MW indicate that the observed reactivity effects and fuel temperature changes are consistent with the calculated Doppler reactivity coefficient

$$\left(\frac{dk}{dT}\right) \text{ of } -0.0085.$$

Thermocouples were used to measure fuel temperature changes at seven different core locations. These measurements were used to estimate an average core temperature for use with a simple logarithmic model to correlate changes in Doppler reactivity with fuel temperatures. The results obtained with this simple model and the predicted Doppler coefficient of -0.0085 are in good agreement with the measured "Doppler feedback" of -75% for a change in power level between zero and 5 MW at a constant coolant temperature of 760°F.

The ratios of the observed fuel temperature changes in the different core locations are in agreement with the relative power densities obtained from a calculated power distribution that included corrections to the diffusion theory calculations at the core surfaces to account for transport and fuel resonance shielding effects. The resonance corrections were calculated by using infinitely dilute, rather than shielded, Pu-239 fission cross sections at the core surface, and the transport corrections were obtained from a comparison of four energy group transport and diffusion theory calculations. Additional discussion of the corrections, as well as an illustration of this power distribution, is provided in reference 9.

The reflector positions, average core coolant temperatures, and power levels obtained during initial reactor operation up to 5 MW are listed in Table 1, along with the measured power reactivity effects corrected for coolant temperature differences. The power levels and average coolant temperatures in Table 1 were calculated from the observed primary system flows and temperatures shown in Table 2, while the zero power reference point was obtained by averaging three zero power measurements shown in Table 3.

The reactivity changes shown in Table 1 are composed of two* effects: the loss in reactivity due to the axial expansion of the fuel and clad as they are heated, and a loss in reactivity due to Doppler broadening of the nuclear cross sections. Because SEFOR has segmented fuel rods and dished fuel pellets, the former effect is small however, and the dominant component is the Doppler effect. The estimated Doppler effect is listed in Table 4. These values were obtained from Table 1 by correcting for the calculated fuel axial expansion reactivity effect of $-0.5\text{¢}/\text{MW}$.

A least squares fit of a parabola to the composite power reactivity data in Table 1 yielded the following relation between the reactivity, K , in cents, and the power, P , in megawatts.

$$K = -(18.35 - 0.55P)P$$

This relation will be compared with the results of the oscillator tests when they become available. The Doppler values in Table 4 are compared with calculations following the discussion of the fuel temperature measurements below.

Fuel temperatures were measured at seven different core locations. The temperature sensing devices are W-Re thermocouples (T/C's) that are described in references 1 and 2. The thermocouples are located along the center line of four different fuel rods in two instrumented fuel assemblies³ (IFA's). All of the fuel thermocouples are located at one of two axial positions; either 2-1/4 inches or 15-1/2 inches above the midplane of the 36 inch high core. Those near the midplane will be referred to as lower T/C's, while those 15-1/2 inches above the midplane will be referred to as upper T/C's. One IFA was located near the core center line, and one near the periphery as shown in Figure 1. As indicated in Figure 1, each IFA contains an "inner fuel rod" and an "outer fuel rod," with the inner rod being that closest to the core center line.

* It may be necessary to account for small additional effects due to changes in coolant inlet temperature and reflector temperature. Experiments to determine these effects are currently being analyzed.

The fuel temperature measurements from the seven T/C's are listed in Table 5. The eighth position, i.e., the upper T/C in the outer fuel rod of the peripheral IFA, was used as a "dummy thermocouple" in order to observe any spurious background signals;—there were none. As indicated in Table 5 and illustrated in Figures 2 and 3, all the fuel temperatures demonstrated an approximately linear dependence on power. The average increases per megawatt power change, i.e., the slopes of the straight lines in the figures, are tabulated in Table 5. The ratios of these slopes are compared in Table 6 with the ratios of local fuel rod power densities obtained from the calculated power distribution including corrections, that is described in reference 9.

In view of the linear power-temperature relationships noted above, the general agreement between the fuel temperature ratios and the power density ratios indicates that the actual power distribution is in agreement with the calculated one obtained by applying the resonance shielding and transport corrections discussed in reference 9. This distribution has a peak to average power ratio of about 1.7.

The actual local Pu enrichment was taken into consideration in determining the power density ratios shown in Table 6. The enrichment at all thermocouple locations in the peripheral IFA was 20.4% Pu/U+Pu, while the enrichment in the outer rod of the central IFA was 18.8% Pu at both thermocouple locations. The enrichments at the upper and lower locations of the inner rod of the central IFA were 18.1% and 17.6% Pu respectively.

In order to compare with the measurements, the temperatures at the fuel thermocouple locations were calculated using Schmidt's⁵ fuel thermal conductivity data for fuel temperatures less than 1100° and the following relationship⁶ for temperatures between 1100°F and 3000°F

$$K = 0.011 + [T(0.4848 - 0.4465D)]^{-1}$$

where K is in units of $\text{watts}/\text{cm}^2\text{-}^\circ\text{C}$, the temperature T is in degrees centigrade, and D is the initial fraction of theoretical pellet density. For the SEFOR IFA pellets $D = 0.937$. The clad to coolant heat transfer coefficient was taken as $5000 \text{ Btu}/\text{ft}^2\text{-hr-}^\circ\text{F}$ and the fuel to clad gap coefficient was adjusted to provide agreement between calculation and experiment.

Initial results indicate that a fuel to clad gap coefficient of about $200 \text{ Btu}/\text{ft}^2\text{-hr-}^\circ\text{F}$ gives calculated temperatures in reasonable agreement with the measurements. This is a reasonable⁷ value at low temperature for the He filled SEFOR fuel rods which have a cold diametral gap of 15 mils. The results of calculations

Table 1
POWER REACTIVITY DATA

Reflector No. 3		Reflector No. 8		Average Core Coolant Temperature ²	Temperature* Reactivity Correction	Power ³	Reactivity** Effects due to Power Changes ¹
Position (cm)	Worth (cents)	Position (cm)	Worth (cents)	(°F)	(cents)	(MW)	(cents)
0	0	—	52.62*	760.0*	0.0	0	0
0	0	37.61	51.58	744.0	-8.80	0.4	-7.76
0	0	41.04	58.27	745.5	-7.98	0.8	-13.63
0	0	48.16	72.22	763.2	1.76	1.0	-17.84
17.0	16.70	40.00	56.22	763.2	1.76	1.0	-18.54
16.76	16.37	47.00	69.97	759.1	-0.50	2.0	-34.22
42.20	61.14	47.00	69.97	760.8	0.44	5.0	-78.05

* Reactivity temperature corrections were made with a coefficient of -0.55 cents/ $^{\circ}$ F.

** This represents the change in core reactivity as the power is increased from zero and the average core coolant temperature is maintained at 760° F.

¹ The estimated standard deviation in the reactivity effect is ± 1.0 cents due to temperature deviations, with an additional $\sim 3\%$ uncertainty due to reflector interaction effects.

² The estimated standard deviation in temperature readings is $\pm 1.5^{\circ}$ F.

³ Preliminary estimate of the standard deviation in power is ± 0.2 MW or $\pm 10\%$, whichever is larger.

Table 2
HEAT BALANCE DATA

Main Primary System			Auxiliary Primary System			Power ³ (MW)	Average Core Temperature ² (°F)
Flow ¹ (gpm)	Inlet* Temperature ² (°F)	Outlet Temperature ² (°F)	Flow ¹ (gpm)	Inlet* Temperature ² (°F)	Outlet Temperature ² (°F)		
1987	741.2	746.2	185	747.2	748.4	0.4	744.0
1995	740.5	750.1	185	744.7	752.8	0.8	745.5
1998	757.2	769.0	185	758.7	771.7	1.0	763.2
2189	747.8	769.8	182	755.1	772.4	2.0	759.1
2326	734.1	787.1	180	743.3	786.3	5.0	760.8

* The recorded main primary inlet and auxiliary primary inlet temperatures have been decreased by 2° F and 1° F respectively to account for differences between inlet and outlet temperature readings at zero power.

¹ These are average values. Flow variations were of the order of 1%.

² The estimated standard deviation in temperature readings is $\pm 1.5^{\circ}$ F.

³ The standard deviation in power has yet to be determined. Preliminary estimates are ± 0.2 MW or $\pm 10\%$, whichever is larger.

Table 3
ZERO POWER CRITICAL POINTS

Date	Reflector No. 3		Reflector No. 8		Reactor Inlet ⁺ Temperature ¹ (° F)	Relative Reactivity* at 760° F (cents)
	Position (cm)	Worth (cents)	Position (cm)	Worth (cents)		
6/30/70	0	0	35.74	48.00	751	52.95
7/1/70	0	0	37.91	52.15	760	52.15
7/8/70	0	0	29.70	36.82	731	52.77
						Average = 52.62

* Reactivity temperature corrections made with a coefficient of -0.55 cents/° F.

+ The recorded panel temperature readings have been increased by one degree to account for an observed offset from the DAS readings used in the other tables.

¹ The estimated standard deviation in temperature readings is $\pm 1.5^\circ$ F.

Table 4
**PRELIMINARY ESTIMATE OF MEASURED
DOPPLER REACTIVITY EFFECTS**

Power (MW)	Estimated Doppler Effect* (cents)
0.0	0.0
0.4	-7.6
0.8	-13.2
1.0	-17.3
1.0	-18.0
2.0	-33.2
5.0	-75.6

* These values include corrections for the calculated fuel and fuel clad axial expansion effect of -0.5 cents/MW, and corrections for changes in core average coolant temperature. Small additional corrections may be required to account for changes in reflector temperature and core inlet temperature. The measurement uncertainties are indicated in Table 1.

FIGURE 1 Instrumented Fuel Assembly (IFA) Locations in Assembly I-I

Table 5
IFA FUEL ROD TEMPERATURES

Fuel Temperature Rise Above the Sodium Temperature**								
Power (MW)	Central IFA				Peripheral IFA			
	Inner Fuel Rod		Outer Fuel Rod		Inner Fuel Rod		Outer Fuel Rod	
	Upper T/C* (°F)	Lower T/C (°F)	Upper T/C (°F)	Lower T/C (°F)	Upper T/C (°F)	Lower T/C (°F)	Upper T/C+ (°F)	Lower T/C (°F)
0	0	0	0	0	0	0	—	0
0.4	52	82	—	85	35	68	—	56
0.8	75	135	—	137	61	104	—	90
1.0	108	166	105	176	77	133	—	116
2.0	197	—	192	329	140	246	—	208
5.0	471	822	477	864	—	636	—	529
Average Increase per Megawatt	94.4	165	95.8	172	75.0	127	—	106

* These are average values. This T/C was noisy with 20 to 30°F temperature fluctuations. The other T/C's demonstrated little or no fluctuations.

+ This is a dummy thermocouple (T/C) that is used to account for any spurious signals.

** The average sodium temperature ranged between ~ 740 and 760°F.

Table 6
RELATIVE FUEL ROD TEMPERATURES AND POWER DENSITIES

Relative Ratios	Central IFA				Peripheral IFA			
	Inner Fuel Rod		Outer Fuel Rod		Inner Fuel Rod		Outer Fuel Rod	
	Upper T/C	Lower T/C	Upper T/C	Lower T/C	Upper T/C	Lower T/C	Upper T/C	Lower T/C
Average Measured Fuel Temperature Rises	0.57	1.00*	0.58	1.04	0.45	0.77	—	0.64
Fuel Rod Power Densities Based on Calculations	0.60	1.00*	0.60	1.02	0.41	0.72	0.37	0.63

* Normalization point—The power density at this point is 80.5% of the peak power density in a normal standard fuel rod at the core center.

FIGURE 2 Fuel Temperature Changes in Central Instrumented Fuel Assembly

FIGURE 3 Fuel Temperature Changes in Peripheral Instrumented Fuel Assembly

using this gap coefficient are shown as a function of power density in Figure 4 and are compared with measured temperatures from two of the T/C's in Figure 5. Better agreement between calculation and measurement may be obtained by use of a model for the fuel-clad gap coefficient which increases with power density to account for the expected reduction in gap size with increasing fuel temperature. For the present calculation, the peak fuel density in normal standard fuel rods was taken as 150 W/cm^3 at 20 MW and the sodium temperature was taken to be 770°F .

The calculated central fuel temperatures in Figure 4 can be compared with any of the measured temperatures in Table 5 by determining the local power density at the thermocouple location from the following relation

$$D = \frac{150}{20} (0.805) P R$$

where D is the desired power density in W/cm^3 ; P is the power level in MW, R is the appropriate power density ratio from row two of Table 6; $150/20$ is the ratio of the peak core power density to the total reactor power, and 0.805 is the factor in the footnote of Table 6.

Since the calculated and measured temperatures were comparable, the calculated curves were used to estimate the average fuel temperature (the "average" temperature cannot be measured) and the average temperature change was used to estimate the Doppler reactivity to obtain a rough comparison with experiment. More detailed calculations are now being performed with the WEDOP⁸ program to obtain better estimates of reactivity-temperature effects. For this comparison the core average temperature increase per MW power change was estimated from Figure 4 by using the relation discussed above to determine the appropriate power density. For a peak to average power ratio of 1.7 the "core average rod" temperature rise above the sodium temperature at 5 MW is about 385°F , or the average increase is 77°F/MW . This value was used with the calculated Doppler coefficient to estimate the Doppler power reactivity effects from the following simple formula

$$K = \frac{-0.0085}{0.00327} (100) \ln \left[1 + \frac{77P}{1220} \right]$$

where K is the reactivity in cents, -0.0085 is the calculated Doppler coefficient; 0.00327 is the estimated effective delayed neutron fraction obtained from the zero power oscillator tests described below; P is the power in megawatts and 1220 is the average coolant temperature in

degrees Rankine. A plot of this equation is shown with the estimated Doppler reactivity feedback data in Figure 6. As indicated by the figure the agreement is quite good. The more detailed calculations which are now being performed should provide better agreement between measured and calculated temperatures and should provide a better basis for comparison of the measured and calculated Doppler reactivity effects.

4.3.2 SEFOR Zero Power Oscillator Tests

4.3.2.1 DESCRIPTION OF EXPERIMENT

The zero power oscillator test was performed by oscillating a poison (boron-carbide) rod in the central drywell to produce sinusoidal oscillations in the reactivity and the neutron flux. The time dependent rod position, and the signals from two of the operational neutron monitors located below the vessel (and approximately 5 ft below the core) were recorded on magnetic tape with the data acquisition system. The sampling rate was controlled by the oscillator rod position so that 32 nearly equally spaced samples were recorded in each cycle of oscillation.

During the experiment, the amplitudes of the power oscillations were always less than 20% of the mean value, and the mean was maintained at a low level ($\sim 1 \text{ KW}$) so that power dependent feedback effects were negligible.

The oscillator rod was positioned so that the lower end of the 26 inch long poison section was at an indicated mean height of 5.5 inches above the core midplane. This location was selected because previous⁹ measurements had indicated a nearly linear dependence of reactivity on rod position in this region of the core. In order to limit the amplitude of the power oscillations, the rod was oscillated with an amplitude that ranged between ~ 4 inches or less at high frequencies and $\sim 1/4$ to $1/3$ inch at low frequencies. Eleven different frequencies of oscillation ranging between $\sim 0.24 \text{ cps}$ and 0.0018 cps were used, and three different sets of data covering the entire frequency range were obtained.

The nominal frequencies of oscillation are listed in Table 7 and the amplitude of the poison rod oscillations for each set of measurements are shown in Table 8. For reference purposes, the data tape and file numbers are also included in Table 8. In addition to the measurements shown in the table, background "noise" measurements at zero amplitude were obtained at each frequency for all three sets of data.

FIGURE 4 Calculated Increase in Fuel Rod Temperature

FIGURE 5 Comparison of Measured and Calculated Temperature in Two Fuel Thermocouples

FIGURE 6 Power Reactivity Feedback — Corrected for Fuel Expansion

Table 7
FREQUENCIES OF OSCILLATION

Designation	Radians/Second	Cycles/Second
F2	1.56	0.25
F3	0.98	0.16
F4	0.61	0.097
F5	0.35	0.056
F6	0.22	0.035
S1	0.13	0.021
S2	0.078	0.0125
S3	0.049	0.0078
S4	0.031	0.0049
S5	0.018	0.0029
S6	0.011	0.00175

Table 8
OSCILLATOR DATA

Set 1 (Tape No. 55)			Set 2 (Tape No. 26)			Set 3 (Tape No. 53)		
Frequency	Rod Amplitude (inches)	File Number	Frequency	Rod Amplitude (inches)	File Number	Frequency	Rod Amplitude (inches)	File Number
F2	3.00	12	F2	3.9	13	F2	1.6	2.
F2	3.96	13	F2	3.35	14	F3	1.6	3
F3	3.89	14	F3	3.35	15	F3	0.68	4
F4	3.27	15	F4	3.35	16	F4	0.69	5
F5	3.11	16	F4	2.3	17	F4	1.24	6
F6	2.50	17	F5	2.3	18	F5	1.24	7
S1	1.85	18	F6	2.3	19	F6	1.24	8
S2	1.58	19	F6	1.5	20	F6	0.70	9
S3	0.96	20	S1	1.56	21	S1	0.71	10
S4	0.74	21	S2	0.78	23	S2	0.70	11
S4	0.74	22	S3	0.78	24	S3	0.71	12
S5	0.46	23	S3	0.35	1*	S4	0.71	13
S6	0.32	24	S6	0.34	27	S4	0.44	14
			S6	0.36	25	S5	0.46	15
						S6	0.24	16

*From Tape No. 53.

4.3.2.2 DESCRIPTION OF ANALYSIS

Most of the steps involved in the analysis have been described in detail in Reference 10, and will be only briefly discussed here. The equations and procedures used for the least squares analysis will be described in some detail however, as they differ from the expressions in the referenced document.

In order to determine the transfer function, the mean value of the flux signals, and the harmonic components of the flux and oscillator rod position signals were determined from the recorded experimental data. The amplitude of the rod oscillation was then converted to a reactivity oscillation by multiplying the displacement with the constant conversion factor of 2.51 cents/inch that had been determined from the previously performed⁹ static worth versus rod position measurements. Finally, the zero power reactivity transfer function was calculated at all frequencies by dividing the fundamental component of the flux signal with the product of the mean flux and the fundamental component of the reactivity oscillation. The resulting transfer functions were fit with the theoretical model.

In order to eliminate the effects of drift, particularly in the flux oscillations at low frequencies, the procedure described by Persson¹¹ was used to determine the fundamental component of the signals. It was found that use of this method with at least six (or interger multiples of six) successive half cycles of data was generally sufficient to eliminate the effects of drift.

The contributions of the second and third harmonics to the flux signals was determined by a conventional Fourier analysis. These contributions were small and they were neglected in the analysis.

As previously reported⁹, analysis of the background data showed that the background noise is essentially zero, and that it does not contribute to the fundamental component of the signal.

4.3.2.3 FITTING OF THE DATA

The mathematical model for the inverse, $Z(iw)$, of the zero power transfer function can be written as follows:

$$Z(iw) = iw \left[\frac{\ell}{\beta} + \frac{1}{\beta} \sum_j f_j \sum_k \frac{\beta_{jk}}{\lambda_{jk} + iw} \right]$$

$$= iw\ell/\beta + \sum_j f_j T_j/\beta$$

where T_j is defined by this expression and where

- i = the square root of minus one
- w = the frequency in radians per second
- ℓ = the prompt neutron lifetime
- β = the effective delayed neutron fraction
- f_j = the fraction of prompt neutrons produced by fissionable isotope j
- β_{jk} = the effective delayed neutron fraction for isotope j and delay group k
- λ_{jk} = the decay constant for isotope j and delay group k .

In SEFOR there are four fissionable isotopes that contribute to the delayed neutrons. These are U-238, Pu-239, Pu-240, and U-235. The contribution from Pu-241 is very small and was neglected. Most (~97%) of the delayed neutrons are produced by U-238 and Pu-239. The characteristics of the zero power transfer function are thus primarily determined by the relative delayed neutron contributions of these two isotopes. If the small contributions from Pu-240 and U-235 are treated as known quantities, the above expression, can be written as

$$Z = q_1 [T_1 + T_{34}] + q_2 [T_2 + T_{34}]$$

$$\text{where } T_{34} = \frac{f_3}{\alpha} T_3 + \frac{f_4}{\alpha} T_4$$

and where the insignificant contribution of $iw\ell/\beta$ in the frequency range of interest (0.01 to 1.5 radians per second) has been neglected.

In the above equation, the indices 1, 2, 3, 4 denote the isotopes U-238, Pu-239, Pu-240, and U-235 respectively; $\alpha = 1 - f_3 - f_4 = f_1 + f_2$; and $q_1 = f_1/\beta$ and $q_2 = f_2/\beta$. The above expression was fit to the experimental values of the inverse, $Z(iw_m)$, of the measured transfer function by minimizing the following sum S with respect to the parameters A_1 and A_2 .

$$S = \sum_m \left\{ \frac{\sigma_o^2}{\sigma_m^2} [X_m - A_1 R_{1m} - A_2 R_{2m}]^2 + \frac{\sigma_o^2}{\delta_m^2} [Y_m - A_1 I_{1m} - A_2 I_{2m}]^2 \right\}$$

In this sum X_m , R_{1m} , and R_{2m} are the real parts of Z , $T_1 + T_{34}$, and $T_2 + T_{34}$ respectively at frequency w_m ; Y_m , I_{1m} , and I_{2m} are the corresponding imaginary components; σ_0 is a normalization constant; and σ_m and δ_m are the estimated standard deviation in X_m and Y_m respectively.

The parameters A_1 and A_2 were determined from the fit. These parameters are the same as q_1 and q_2 of the previous expression except for a normalization constant $C = A_j/q_j$ which has been introduced to normalize the experimental data and to reduce the uncertainty introduced during the conversion of oscillator rod position to reactivity.

The equations for the two parameters are as follows:

$$A_j = \sum_{k=1}^2 Q_{jk} P_k \quad \text{for } j = 1, 2$$

where Q is the inverse of a T matrix whose elements are defined by

$$T_{jk} = \sum_m \frac{\sigma_0^2}{\sigma_m^2} R_{jm} R_{km} + \sum_m \frac{\sigma_0^2}{\delta_m^2} I_{jm} I_{km}$$

and P is a vector with elements defined by the following equation.

$$P_j = \sum_m \frac{\sigma_0^2}{\sigma_m^2} X_m R_{jm} + \sum_m \frac{\sigma_0^2}{\delta_m^2} Y_m I_{jm}$$

The estimated standard error σ_1 and σ_2 in the parameters A_1 and A_2 were also determined from the fit by using the following expression

$$\bar{\sigma}_j = \sigma \sqrt{Q_{jj}}$$

$$\text{where } \sigma = \sqrt{\frac{S}{2M-2}}$$

S is the sum defined above and M is the number of measurements in a given set.

The fractions f_1 and f_2 were determined from the values A_1 and A_2 according to the following expression for $j = 1, 2$.

$$f_j = \alpha A_j / (A_1 + A_2)$$

The effective delayed neutron fraction was determined from

$$\beta = \sum_j f_j \sum_k \beta_{jk}$$

and the normalizing factor C was determined from

$$C = (\beta/\alpha) (A_1 + A_2)$$

The estimated standard errors σ_β in β , σ_f in f_1 and f_2 (the error in f_1 is the same as in f_2), and σ_c in c were determined from the following expressions:

$$\sigma_f = \frac{\sigma}{A_1 + A_2} \sqrt{f_2^2 Q_{11} - 2 f_1 f_2 Q_{12} + f_1^2 Q_{22}}$$

$$\sigma_\beta = \sigma_f \left| \beta_1 - \beta_2 \right|$$

where β_1 and β_2 are defined below.

$$\sigma_c = \sqrt{Q_{11} E^2 - 2 E G Q_{12} + G^2 Q_{22}}$$

with E and G defined by

$$E = \alpha \beta_1 + (f_3 \beta_3 + f_4 \beta_4) / \alpha$$

$$G = E + \alpha (\beta_2 - \beta_1)$$

where the delayed neutron fraction β_j for each isotope ($j = 1, 2, 3, 4$) is defined by

$$\beta_j = \sum_k \beta_{jk}$$

The delayed neutron parameters given by Keepin¹² were used in the analysis. In order to obtain the effective delayed neutron fractions, Keepin's values were modified by calculated effectiveness factors. The factors for the four isotopes were calculated under the assumption that the delayed neutron energy spectra were the same for all isotopes. The energy spectra for the first four groups was taken from Figure 4-5 of Reference 12 and the energy spectrum for the last two groups was assumed to be a composite of the other four spectra. The relative effectiveness factors (i.e., the ratio of the effective delayed neutron fraction to the actual delayed fraction) for groups 1 through 6 that were obtained from a perturbation calculation using the BISYN¹³ code are 0.8906, 0.9146, 0.9033, 0.9130, 0.9116 and 0.9116 respectively. The values $f_3 = 0.019$ and $f_2 = 0.008$ that were used in the analysis were obtained from calculated core average reaction rates.

4.3.2.4 RESULTS

The three sets of data in Table 8 were analyzed. Since the signals from two different flux detectors (denoted by A and B in Table 9) were included in each set, two "measured" power reactivity transfer functions at up to 15 frequency points were available from each of the three sets. These six separate groups of data were fitted in two ways. In one fitting, the standard deviations, σ_m and δ_m , which had been estimated from the "spread" in the fundamental components of the signals were used to obtain pointwise weight functions,

$$\sigma_o^2/\sigma_m^2 \text{ and } \sigma_o^2/\delta_m^2.$$

A second fitting of the six groups was performed with a constant weight, since it was expected that the weights would be relatively constant, and the actual weights did not, in general, exhibit a great variance. For the flux channels the "spread" in the cosine and sine components of the signal from cycle to cycle were of the order of 0.1% of the mean flux value, while the induced flux oscillations varied between 2% and 10% of the mean value. The "spread" in the cosine and sine components of the oscillator rod position signal were negligible.

The results of the fits using the two types of weights are shown in Table 9. There seemed to be little difference in the two methods, and a weighted average of all values (the weights being proportional to the inverse square of the estimated standard errors) were used to obtain the values at the bottom of Table 3. The averaged value of β , 0.00327, is in good agreement with the calculated value of 0.00315. The average values for f_1 and f_2 were 0.118 and 0.855 respectively. The corresponding calculated values were 0.103 and 0.870. The average value of C was 0.9174. If the static rod worth calibration value of 2.51 cents/inch were also the correct calibration for this dynamic measurement, C would be unity. Since it is not, the dynamic worth of the rod is 9% higher than the static worth, and to correctly normalize the dynamic reactivity data a conversion factor of $2.51/0.9174 = 2.74\text{¢/inch}$ should be used.

It should be noted that the values of β , f_1 , and f_2 that are shown in Table 9 are those values that provide a best fit to the SEFOR oscillator data based on the calculated relative effectiveness factors, and the assumption that the model and the parameters in the model (the β 's and λ 's) are correct. The estimated standard errors which are shown therefore provide a measure of the goodness of fit, but may be misleading as to the actual uncertainty in the effective delayed neutron fraction and in the fraction of fissions produced by the two major fissionable isotopes.

The normalized experimental data are compared with the "best" calculated curves in Figures 7 through 14. The delayed neutron parameters which will be used in the analysis of the data at power are listed in Table 10, and the values of this transfer function and its inverse are tabulated at several frequencies in the range of interest in Table 11. The values of the measured transfer functions are listed in Tables 12 through 14.

4.3.3 Conventional Oscillator Tests

The first series of the at-power conventional oscillator tests have been carried out at 2 MW with a coolant flow of 2000 gpm (40% of the maximum value). The analysis of the data indicates that at this power and flow, the primary feedback effects are due to the change in the temperature of the fuel; as the change in coolant temperature is very small.

The amplitude of the power-reactivity transfer function at 2 MW is shown in Figure 15 and the phase in Figure 16.

As with the zero power oscillator tests, three series of measurement have been carried out at eleven frequencies and the data from two power channels have been recorded and analyzed.

The feedback kernel will be determined and a least square fitting of a model to the measured data will provide a value for the power coefficient at 2 MW.

4.3.4 Reflector Recalibration

The stroke of each fine reflector was reduced approximately 3 cm (out of a hundred cm stroke) when the reflector thermocouples were repaired.

The fine reflectors were recalibrated to determine whether the slight reduction in stroke had altered previous calibration curves. Within the accuracy of the calibrations (~1%) no change was observed in shape of the worth versus position curves, and only a slight (1 to 1.5 cent) reduction in total worth resulted from the modification.

The results of the recalibration measurements using period measurements are shown in Tables 15 and 16 and in Figures 17 and 18. These results can be compared with the previous calibration of reflector No. 8 in Assembly I-F and of No. 3 in Assembly I-H that are given in Reference 9.

Reflector No. 3 was calibrated with reflectors No. 5 and 10 down and with reflector No. 6 at 42.9 cm. Reflector No. 8 was also calibrated with reflector No. 5 and 10 down and reflector No. 6 was at 40.8 cm.

Table 9
FITTED PARAMETERS

Set Number	Flux Detector	Weights	Prompt Neutron Fraction for U-238	Prompt Neutron Fraction for Pu-239	Normalization Constant	Effective Delayed Neutron Fraction
			f_1	f_2	C	β
1	A	Pointwise	0.1224	0.8510	0.9342	0.00332
1	B	Pointwise	0.1255	0.8478	0.9386	0.00336
2	A	Pointwise	0.1212	0.8522	0.9210	0.00331
2	B	Pointwise	0.1258	0.8475	0.9172	0.00336
3	A	Pointwise	0.1128	0.8605	0.8977	0.00321
3	B	Pointwise	0.1136	0.8598	0.8943	0.00322
1	A	Constant	0.1231	0.8503	0.9367	0.00333
1	B	Constant	0.1225	0.8509	0.9340	0.00332
2	A	Constant	0.1213	0.8520	0.9216	0.00331
2	B	Constant	0.1142	0.8591	0.9177	0.00323
3	A	Constant	0.1016	0.8718	0.8942	0.00308
3	B	Constant	0.1137	0.8597	0.8954	0.00322
Weighted Average			0.118 $\pm 0.010^*$	0.855 $\pm 0.010^*$	0.9174 $\pm 0.037^*$	0.00327 $\pm 0.00010^*$

* Estimated standard errors obtained from the fit. These values do not reflect possible additional uncertainties introduced in the calculation of the effectiveness factors, nor the uncertainties in the actual isotopic β 's and λ 's.

Table 10
RECOMMENDED SEFOR DELAYED NEUTRON PARAMETERS

i	U-238		Pu-239		Pu-240		U-235	
	$F_1 = 0.118$		$F_2 = 0.855$		$F_3 = 0.019$		$F_4 = 0.008$	
	$\beta_i \times 10^5$	$\lambda_i(\text{sec})$	$\beta_i \times 10^5$	$\lambda_i(\text{sec})$	$\beta_i \times 10^5$	$\lambda_i(\text{sec})$	$\beta_i \times 10^5$	$\lambda_i(\text{sec})$
1	17.1	0.0132	6.95	0.0129	6.68	0.0129	21.7	0.0127
2	185.5	0.0321	52.22	0.0311	66.40	0.0313	124.8	0.0317
3	216.6	0.139	39.84	0.134	46.16	0.135	108.8	0.115
4	524.2	0.358	61.08	0.331	85.00	0.333	238.2	0.311
5	303.6	1.41	19.14	1.26	30.99	1.36	74.8	1.40
6	101.2	4.02	6.47	3.21	7.02	4.04	15.2	3.87
Total	1348.2		185.70		242.25		583.5

FIGURE 7 Imaginary Part of the Inverse of the Zero Power Transfer Function Using Detector A

FIGURE 8 Imaginary Part of the Inverse of the Zero Power Transfer Function Using Detector B

FIGURE 9 Real Part of the Inverse of the Zero Power Transfer Function Using Detector A

FIGURE 10 Real Part of the Inverse of the Zero Power Transfer Function Using Detector B

FIGURE 11 Amplitude of the Zero Power Transfer Function Using Detector A

FIGURE 12 Amplitude of the Zero Power Transfer Function Using Detector B

FIGURE 13 Phase of the Zero Power Transfer Function Using Detector A

FIGURE 14 Phase of the Zero Power Transfer Function Using Detector B

Table 11
SEFOR ZERO POWER TRANSFER FUNCTION G_0

Frequency (radians/sec)	Real Part of the Inverse of G_0 (cents/% power)	Imaginary Part of the Inverse of G_0 (cents/% power)	Amplitude of G_0 (% power/cents)	Phase of of G_0 (degrees)
0.00400	0.005760	0.04369	22.69	-82.49
0.00500	0.008794	0.05390	18.31	-80.73
0.00630	0.01348	0.06657	14.72	-78.56
0.00800	0.02061	0.08200	11.83	-75.89
0.0100	0.03008	0.09850	9.710	-73.02
0.0125	0.04296	0.1167	8.042	-69.79
0.0160	0.06191	0.1380	6.611	-65.84
0.0200	0.08364	0.1573	5.612	-62.00
0.0250	0.1095	0.1755	4.834	-58.04
0.0315	0.1397	0.1922	4.209	-54.00
0.0400	0.1728	0.2070	3.708	-50.14
0.0500	0.2045	0.2196	3.333	-47.04
0.0630	0.2379	0.2327	3.005	-44.37
0.0800	0.2742	0.2476	2.707	-42.08
0.100	0.3111	0.2626	2.456	-40.16
0.125	0.3522	0.2777	2.230	-38.26
0.160	0.4028	0.2929	2.008	-36.02
0.200	0.4531	0.3035	1.834	-33.81
0.250	0.5067	0.3094	1.684	-31.41
0.315	0.5640	0.3093	1.555	-28.74
0.400	0.6224	0.3019	1.446	-25.88
0.500	0.6735	0.2892	1.364	-23.24
0.630	0.7212	0.2721	1.297	-20.67
0.800	0.7650	0.2527	1.241	-18.28
1.00	0.8017	0.2342	1.197	-16.20
1.25	0.8350	0.2155	1.160	-14.47
1.60	0.8683	0.1945	1.124	-12.62
2.00	0.8949	0.1750	1.097	-11.06
2.50	0.9179	0.1555	1.074	-9.614
3.15	0.9379	0.1357	1.055	-8.234

Table 12
MEASURED ZERO POWER TRANSFER FUNCTION
Set 1

Frequency (radians/sec)	Amplitude of Zero Power Transfer Function (% power/cent reactivity)		Phase of Zero Power Transfer Function (degrees)	
	Detector A	Detector B	Detector A	Detector B
0.0112	9.202	9.206	-71.26	-71.30
0.0178	6.048	6.022	-62.44	-62.46
0.0306	4.318	4.341	-53.30	-53.31
0.0306	4.324	4.339	-53.23	-53.74
0.0490	3.383	3.375	-46.19	-46.28
0.0780	2.714	2.726	-41.22	-41.11
0.1289	2.195	2.196	-36.93	-37.08
0.2231	1.737	1.748	-31.88	-31.82
0.3518	1.482	1.484	-26.92	-26.82
0.6071	1.285	1.288	-20.87	-20.83
0.9708	1.170	1.171	-16.44	-16.59
1.533	1.093	1.098	-13.02	-13.01
1.560	1.109	1.112	-13.15	-13.03

Table 13
MEASURED ZERO POWER TRANSFER FUNCTION
Set 2

Frequency (radians/sec)	Amplitude of Zero Power Transfer Function (% power/cent reactivity)		Phase of Zero Power Transfer Function (degrees)	
	Detector A	Detector B	Detector A	Detector B
0.0112	9.075	9.065	-69.11	-69.17
0.0112	9.183	9.187	-69.93	-69.36
0.0488	3.415	3.401	-45.67	-47.78
0.0491	3.401	3.407	-46.47	-46.42
0.0779	2.814	2.796	-41.41	-41.29
0.1292	2.207	2.222	-37.25	-37.04
0.2225	1.779	1.787	-32.10	-32.18
0.2231	1.763	1.773	-32.17	-31.93
0.3521	1.510	1.517	-26.95	-27.01
0.6103	1.311	1.315	-20.70	-20.93
0.6091	1.292	1.300	-20.79	-20.99
0.9769	1.194	1.197	-16.42	-16.40
1.556	1.109	1.119	-12.82	-13.03
1.558	1.118	1.111	-12.96	-13.03

Table 14
MEASURED ZERO POWER TRANSFER FUNCTION
Set 3

Frequency (radians/sec)	Amplitude of Zero Power Transfer Function (% power/cent reactivity)		Phase of Zero Power Transfer Function (degrees)	
	Detector A	Detector B	Detector A	Detector B
0.0112	9.189	9.249	-69.02	-68.81
0.1783	—	6.285	—	-62.30
0.0308	4.387	4.365	-52.33	-53.16
0.0307	4.364	4.385	-53.41	-53.51
0.0490	3.333	3.398	-46.38	-46.31
0.0781	2.768	2.785	-41.35	-41.31
0.1291	2.248	2.265	-36.83	-37.15
0.2246	1.800	1.803	-32.01	-31.88
0.2248	1.788	1.791	-31.82	-31.92
0.3555	1.535	1.537	-26.85	-26.87
0.6143	1.331	1.343	-20.78	-20.95
0.6133	1.337	1.342	-21.13	-20.81
0.9812	1.230	1.234	-16.22	-16.62
0.9815	1.216	1.217	-16.88	-16.55
1.558	1.140	1.139	-12.88	-12.74

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FIGURE 15 Amplitude of Power-Reactivity Transfer Function at 2 MW

FIGURE 16 Phase of the Power-Reactivity Transfer Function at 2 MW

Table 15
REFLECTOR 3 CALIBRATION – ASSEMBLY I-I
(370° F)

Reflector Number 3

From (cm)	To (cm)	Doubling Time (sec)	Reactivity (cents)
00.16	10.11	82.8	7.643
10.11	17.95	69.0	8.841
17.95	25.07	58.4	10.06
25.07	30.06	78.8	7.955
30.06	35.14	69.5	8.791
35.14	40.06	68.9	8.851
40.06	44.97	66.8	9.068
44.97	49.94	66.0	9.154
49.94	55.13	63.8	9.399
55.13	59.64	81.8	7.719
59.64	66.09	56.5	10.316
66.09	72.05	74.0	8.365
72.05	78.62	81.4	7.75
78.62	87.06	86.8	7.356
87.06	95.17	168.3	4.177

Table 16
REFLECTOR 8 CALIBRATION – ASSEMBLY I-I
(370° F)

Reflector Number 8

From (cm)	To (cm)	Doubling Time (sec)	Reactivity (cents)
00.42	9.51	89.9	7.148
9.51	17.08	76.8	8.122
17.08	23.51	71.6	8.588
23.51	28.96	75.3	8.251
28.96	34.06	73.7	8.394
34.06	38.76	76.5	8.148
38.76	43.42	75.5	8.234
43.42	48.05	75.9	8.199
48.05	52.59	82.5	7.667
52.59	57.48	75.8	8.208
57.48	64.01	56.3	10.345
64.01	71.47	55.8	10.415
71.47	79.99	61.4	9.683
79.99	88.63	100.6	6.513
88.63	95.27	275.0	2.674

FIGURE 17 Reflector No. 3 Calibration in Assembly I-I (370°F)

FIGURE 18 Reflector No. 8 Calibration in Assembly I-I (370°F)

5. TASK II—TRANSIENT TESTS

5.1 PURPOSE

The purpose of Task II 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 cask B of E 1233 which is intended to be used for the shipment of selected SEFOR fuel rods to a "hot lab" for destructive examination. The cask has been received and inspected by GE and all deficiencies have been corrected. The inside of the cask has been cleaned and the outside painted. The cask has been shipped to the SEFOR Site where it is being stored.

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 sent to these organizations for approval. The license application contains a description of the cask, the procedures to follow when handling the fuel and sufficient analysis to show the conditions described in 10CFR71 are met.

The design and fabrication of the equipment required for the fuel transfer operation has been completed and sent to the SEFOR Site for storage. This equipment includes the transfer tube and seals between the refueling cell valve and transfer cask, the set of canisters used to ship a single failed fuel rod and the canister used to ship three nonfailed fuel rods.

The design of the shielding around the vertical transfer valve in the SEFOR Refueling Cell is completed. The shielding is designed as a semi-permanent 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

Questions from the AEC concerning the cask license application were answered.

Fabrication of the shielding for the vertical transfer valve has begun. This shielding consists of lead filled containers that fit around the valve. The containers have been built and preparations are being made to fill them with lead.

6. SEFOR SAFETY AUDIT

The installation of the replaceable thermocouples in the reflector segments was completed and the changes in the Technical Specifications to permit their use in the safety system were approved by the AEC-DRL. Reactor operation up to 5 MW has shown that these thermocouples respond as expected. None of these thermocouples have failed, but the failure rate for thermocouples originally installed in the guide structure continues to be high.

Other Technical Specification changes have been proposed and are presently being reviewed by the AEC-DRL. These changes involve: (1) an increase in the allowable pump-around flow limit to improve system operation and eliminate numerous low flow alarms; (2) a revision of the equations used for gaseous radwaste discharge limits, to

clarify the limits; (3) a revision of portions of the Administrative Section; (4) revisions required to permit use of guinea pig rods under the innermost ports throughout the planned test program.

The reflector drive system was tested after the new thermocouples were installed in the reflector segments, and the results of these tests were reviewed to determine that the modifications had no effect on this system.

The results of the FRED checkout tests were reviewed to ascertain that this device would operate properly.

Engineering surveillance data from plant auxiliary systems was reviewed to provide assurance that these systems are operating properly.

7. SEFOR SUPPORT

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

During May, the reactor remained shut down for completion of the reflector guide structure thermocouple modification work. During this period, the modification and the addition of thermocouples to the replacement extension shafts for each of the ten reflector segments was completed. Details of the installation work are given in the SEFOR Quarterly Progress Report No. 24, GEAP 10010-24. The installation, assembly and checkout activities connected with the reflector thermocouple installation was completed on schedule. Following this work, the containment building was prepared for the containment leak test. Following satisfactory completion of this task, the reactor systems were ready for the power ascension phase.

As a part of the preparation for the power ascension, one of the necessary steps was to determine the quantity of NaK remaining in the refueling cell argon purification system and from this determination, verify the remaining cleanup capability available during reactor operation. This task was accomplished by draining NaK from the refueling cell purification system into drums so that the liquid quantity could be measured accurately. It was determined that of the total of 4100 pounds NaK originally available for purification, it was possible to transfer only approximately 400 - 500 lbs from the tank. This information indicated that the useful life remaining in the bubbler was very short. Design effort has been initiated to procure and install a replacement purification system. The purification system design basis is as follows:

- a) Two smaller size units are placed in parallel such that one can be taken off stream for removal or maintenance while the other unit is on stream.
- b) A backflow tank is provided which can contain the entire quantity of NaK in one of the purification units.

- c) A separator is provided downstream to remove potential entrainment.

A flowsheet illustrating the design of the replacement purification system is shown in Figure 19. Design drawings are currently in the design process and critical materials placed on order.

Additional sodium leak detectors were added to the auxiliary primary loop to provide backup for leak detectors which are currently installed. These leak detectors are installed in regions which would become inaccessible following reactor power operation. Leak detectors are designed in such a manner that the detector itself can be replaced from outside of the shielding surrounding the reactor vessel.

In order to provide additional backup for the reactor vessel through-head tip probe, (Type I), an additional probe is being fabricated. The backup probe is similar in design to the currently installed tip probe, and will provide a ready replacement in the event that malfunctions occur in the existing probe.

Replacement reactor/vessel head bolts were fabricated which are designed to eliminate problems associated with galling of threads previously experienced. The new replacement bolts have a heavier nitrided case which is designed to minimize galling. Procedural controls have been developed for installing the head bolts which reduce the potential for damage. Prototype bolts and procedure were mocked up in San Jose prior to final fabrication and delivery of the reference bolt design.

Updating of drawings associated with various portions of the plant was initiated. Plant modifications were reviewed to determine the extent of updating necessary. P&ID's were updated and electrical drawings are in the process of updating. Mechanical drawings are being reviewed for possible revision.

FIGURE 19. Refueling Cell Argon Purification System

8. SEFOR OPERATION

8.1 EXPERIMENTAL PROGRAM

At the close of the last reporting period, preparations were under way to check the Fast Reactivity Excursion Device (FRED) out-of-refueling-cell. These preparations were completed, the test procedure prepared and approved, and the FRED mounted on a suitable test stand. The out-of-refueling-cell testing was completed. The nominal ejection time for 20" of travel from liftoff (the distance to travel from the core centerline to upper fuel boundary) is 101 milliseconds when the accumulator is charged to the design pressure of 1500 psig, which compares favorably to the 99 millisecond time obtained in San Jose tests.

When the reflector guide structure thermocouple replacement had been completed and the reactor control and instrumentation returned to a fully functional state, provisions were made to perform an "in situ" FRED test with the device installed on the reactor vessel head. This test provided a complete simulation of the FRED operation under typical transient operation environment (exclusive of Argon in the refueling cell). Requirements for mechanical manipulation during installation and removal, operations familiarization, and installed electrical and hydraulic connection verification were also obtained. Test procedures were written, approved and the test performed. Results of these tests are being evaluated and compared to design prototype testing carried out earlier in the program in San Jose, California. The test results are being analyzed to obtain distance traveled as a function of time. The in-cell and out-of-cell test results compare favorably.

Two discrepancies were noted during the in-cell testing. First, the piping connecting the transfer chamber (out-of-cell) with the accumulator (in-cell) has a larger volume than calculated which results in a 2000-psig transfer chamber pressure providing only ≈ 1400 psig accumulator pressure rather than the 1500 psig design objective. The second discrepancy involves a difference in the method of grounding the circuit which provides signals from the liftoff switches in the FRED and the input circuits of the Data Acquisition System (DAS). It is anticipated that an increase in the allowable pressure in the transfer chamber will be made to increase the pressure in the accumulator to the design value of 1500 psig, and a circuit modification will be made to obtain compatibility between the FRED signals and the DAS input requirements.

Preparations for the absolute fission rate measurements continued. The test procedure for the performance of the solid state track recorder (SSTR)* absolute fission

rate measurements was in preparation. Source foils shown in Table 17 were received at SEFOR. The source foils are 3-mil-thick nickel discs on which the respective isotope has been electro-deposited. An Americium-241 source was obtained from the National Bureau of Standards for use as a calibration source for absolute alpha counting of the foils. The absolute alpha counting will be performed by the Chemistry Department of the University of Arkansas. The calibration source will be used to calibrate their counting equipment and will be performed in the near future.

To provide additional information for interpretation of the B/l noise measurements performed during the zero power tests, an additional calibration test was conducted which exposed the He^3 chambers to a "white noise" neutron source. By slightly modifying the transient detector chamber carriages, it was possible to insert an He^3 chamber into the transient fission detector tubes. This locates the detector (He^3) in a region of highly thermalized neutrons which approach a "white noise" neutron source. Procedures for the measurements were completed and the detectors placed in the tubes at approximately the centerline of the core. Just prior to resuming the power ascension program measurements of chamber current and voltage plateaus at the conditions shown in Table 18 were recorded and transmitted to Sunnyvale for analysis.

At the conclusion of the reflector guide structure thermocouple replacement, mechanical performance and nuclear verification tests of the reflector system tests were completed. The mechanical test was, essentially, the same as the original preoperation test prior to reactor loading. The nuclear verification consisted of comparisons of critical reflector configurations and calibrations of the two fine reflectors prior to and after the modification.

The mechanical checkout was completed satisfactorily. The critical conditions are shown in Table 19 prior to and after the modifications. Figures 20 and 21 illustrate the calibration comparisons of the two fine reflectors. Within the uncertainty of the measurements, no detectable changes were observed in integral or differential worths of the two reflectors.

Also, in the interim of plant shutdown, the transient detector instrumentation was checked. Two transient amplifiers were returned to the vendor for repair. A transient test plan was prepared to provide guidance in the preparation of the transient test procedures. The familiarization transient tests procedure has been drafted and is being reviewed. The subprompt transient tests procedure is being drafted.

*Gold, Raymond; and Armani, R. J., "Absolute Fission Rate Measurements with Solid-State Track Recorders," Nuclear Science and Engineering, 34, pp 13-32, 1968.

Table 17
SSTR* ABSOLUTE FISSION RATE
SOURCE FOILS

Isotope	Isotope Surface Density - μ gm/cm ²
Pu ²³⁹	5, 10, 15 and 20
Pu ²⁴⁰	20, 40, 60 and 80
U ²³⁸	150, 250 and 400
U ²³⁵	5, 10, 15 and 20

* Solid State Track Recorder

Table 18
CONDITIONS FOR "WHITE NOISE"
CALIBRATION OF He³ DETECTOR

Reactor Power WRM-%*	Noise Recording	DC Current Amperes	Plateau
Shutdown	Yes	0.45×10^{-9}	
13×10^{-6}	Yes	0.27×10^{-8}	
62.5×10^{-6}	Yes	1.29×10^{-8}	
20.5×10^{-5}	Yes	3.88×10^{-8}	
53×10^{-5}	Yes	1.02×10^{-7}	Yes
25×10^{-4}	Yes	0.44×10^{-6}	Yes
60.5×10^{-4}	Yes	1.14×10^{-6}	
24×10^{-3}	Yes	0.4×10^{-5}	Yes
Detector Voltage Off			
Shutdown	Yes	0	
24×10^{-3}	Yes	0	

* Wide Range Monitor (WRM) - 100% $\approx$ 20 MW_t

Table 19
CRITICAL CONDITION COMPARISON BEFORE AND AFTER
REACTOR GUIDE STRUCTURE T/C MODIFICATION

Temperature-°F	#5	#10	#3	#8(cm)	Temperature Corrected #8(cm)	Difference (cents)
Pre 399 *	Down	Down	Down	58.80		
					49.50	2.7 ± 2.5 cents**
Post 379 *	Down	Down	Down	50.97		

* ± 2.5°F

** $\pm \sqrt{2 \times 2.5^\circ \times .7 \text{ cents}/^\circ \text{F}} = 2.5 \text{ cents}$

The Automatic Balanced Oscillator Control Equipment was inspected and a partial checkout completed by a Karlsruhe design representative. Several items had been damaged in transit and were returned to the vendors (in USA) for repair. The Balanced Oscillator (manual and automatic) test procedures for both types of tests, constant power and constant temperature, were written and distributed for comment. A major share of the comments have been received and are in the process of being included in the procedure.

Preoperation checkout of the balanced oscillator pump control unit was completed. Performance of the control unit was satisfactory except at the highest anticipated oscillation rate of 0.033 Hz. At that frequency, an irregular sinusoidal flow oscillation was observed, i.e. high content of frequencies other than the fundamental including harmonics. The wave shapes were analyzed for higher harmonic content and are shown in Table 20. The effects of the irregularity on flow control are being evaluated.

Shortly after the two instrumented fuel assemblies (IFA's) were installed in January, the flowmeter in the center IFA (VNC-29) failed to function. The malfunction was believed to be due to in-leakage of sodium shorting the signal leads. However, to ensure that the highly unlikely possibility of a flow blockage would not go undetected, if it existed, procedures were prepared which required close scrutiny of the temperature data at 0.5 MW intervals as the reactor was raised from zero power to 5 MW(t). The pro-

cedures also obtained the static data at 1 MW at a core average temperature of $\approx 760^\circ \text{F}$. The IFA comparison at selected power levels between 0 and 5 MW(t) are shown in Table 21. The temperature rise across both IFA's at all power levels was comparable with the rise across the reactor vessel (and hence proportional to the rise across the core) and exhibited no evidence of flow blockage.

Test data was gathered at 1, 2, and 5 MW(t). At 2 MW(t), static tests and conventional oscillator tests were performed. The analyses of these data are discussed in subsection 4.3.3. The 2 MW(t) Nyquist plot was prepared from the conventional oscillator data and is illustrated in Figure 22. It is noted that the phase margin where the Nyquist plot crosses the unit circle is 83.4° (see Figure 23), which indicates that at 2 MW, SEFOR is highly stable.

At the 5 MW(t) power level, tests were performed at $\approx 2500 \text{ gpm}$ (50% flow) and 760°F core average. These tests consisted of static test, conventional oscillator test, and reflector temperature coefficient test. Plant radiation measurements were also made at these conditions.

When the 50% flow measurements at 5 MW(t) were completed, the flow was increased to full flow (5000 gpm), and the static and conventional oscillator tests were again performed. Static tests at full flow were repeated a second time to verify the proper functioning of the DAS during the first measurements. The flow was reduced to $\approx 1500 \text{ gpm}$ (30% flow) and 760°F core average, and static tests were performed. Oscillator tests at this condition were delayed because of malfunction of the DAS.

FIGURE 20 Integral Reactivity Worth Curve Comparison Pre and Post Reflector Guide Structure T/C Modification Reflector No. 3

FIGURE 21 Integral Reactivity Worth Curve Comparison Pre and Post Reflector Guide Structure T/C Modification Reflector No. 8

Table 20
BALANCED OSCILLATOR PUMP CONTROL
FLOW OSCILLATION HARMONIC CONTENT
MAIN PRIMARY PUMP

Case	Frequency Hz	2nd Harmonic %	3rd Harmonic %	Remark
I	0.0045	3	2.95	Harmonics
II	0.0045	4.1	2.8	Are In
III	0.012	3.4	3.5	Random
IV	0.012	2.0	3.7	Fashion
V	0.033	4.5	7.4	
VI	0.033	3.7	5.2	

Table 21
IFA POWER ASCENSION TEMPERATURE
RISE COMPARISON

Reactor Power	Coolant Temperature Rise - °F VNC-29	Coolant Temperature Rise - °F VNC-25	Reactor Vessel Coolant Temperature Rise - °F
1.0	13.3	14.5	15.1
1.5	17.4	18.1	16.9
2.0	20.9	21.5 *	19.6
3.5	36.9	25.0	35.4
4.0	42.9	28.4	41.8
5.0	54.7	34.4	41.0

* Above this power level, the inlet T/C for VNC-25 appeared to develop a false junction measuring some intermediate condition and are unreliable with respect to temperature difference.

FIGURE 22 SEFOR Nyquist Diagram for 2 MW, 50% Flow (Third Quadrant Only)

FIGURE 23 SEFOR Nyquist Diagram—Unit Circle Intercept Phase Margin Degrees

Nyquist plots have been prepared for the 5 MW(t) data and are illustrated in Figures 24 through 26. The phase margin where the Nyquist plot intersects the unit circle ranges from 74.5 to 77°. This large phase margin is indicative of a very stable reactor system. It should also be noted that the plots are in good agreement with predictions (see Figures 28 and 29 of GEAP-10010-24). Although linear extrapolations of the phase margins are not consistent with values predicted by calculations, Figure 23 indicates the phase margin at 10 MW(t) from linear extrapolation of measured data at 2 and 5 MW(t). The indicated value at 10 MW(t) by extrapolation is $\approx 60^\circ$.

The power ascension test phase had been augmented with a procedure, Primary Shield and Bottom Shield Temperature Monitoring, TP No. II-10, which acquires shielding temperatures at various power levels. This information will be analyzed to determine the adequacy of the shield cooling and design calculations.

Progress on the Experimental Program was limited by malfunctions of the oscillator, oscillator angular position indication system, oscillator positioner, and the Data Acquisition System.

The inner and outer containment leak tests were performed at a nominal 10 psig. The results were as follows:

	% Pressure Decay in 24 hours	
	Allowable	Measured
Inner Containment	16.5	6.91 $\pm$ 0.39
Outer Containment	1.4	0.46 $\pm$ 0.15

8.2 OPERATION AND MAINTENANCE

The outage for installation of Reflector Guide Structure thermocouples and the Annual Containment leak tests, which began on March 20, was terminated on June 26, 1970. The major effort during the outage was directed toward the installation of replacement temperature monitoring systems for the Reflector Guide Structure. The replacement systems, which provide the ability to replace from the Reflector Drive Room any thermocouple that malfunctions, monitors the nickel temperature at the lower end of the reflector segment and the nitrogen temperature at the outlet of each reflector segment bay. Either set of these temperatures is related to the reflector guide structure temperatures, which were connected to the reactor safety system in a two-out-of-ten trip logic. Installation was accomplished at this time because of the high rate of failure of the guide structure thermocouples that had been experienced, and the difficulty that would be encountered

in the installation after the components had become highly radioactive during power operation. Radiation levels encountered during the installation were very low and contamination nonexistent except for a slight amount on the nickel surfaces. A change in the Technical Specifications was approved to permit the use of the replacement temperature monitoring systems in the safety system.

A replacement sodium leak detector system was installed to supplement the six detectors on the Auxiliary Primary IHX return to reactor vessel loop that are connected to the reactor safety system. Since the originally installed detectors are within the primary pipe tunnel and may become inaccessible for repair because of core radiation after power operation, each detector of the replacement system is replaceable from the primary sodium equipment room.

Preventive maintenance was performed on the reflector valve packs and hydraulic drives including disassembly, inspection, cleaning, replacement of "O" rings and seals, and pressure testing of components. Following re-assembly of the reflector drive system, a thorough pre-operation test was conducted to demonstrate the operability of the system.

Two attempts were made to measure the quantity of NaK remaining in the argon bubbler by transferring to drums and weighing were only partially successful. A total of approximately 600 pounds could be transferred at the pressures used. An additional 800 pounds of NaK were procured and charged into the bubbler to permit continuance of operation until a replacement purifier can be designed and fabricated.

The reactor vessel thru-head port cover seals and bolts were replaced with new components of improved design that will permit remote removal and installation.

Other significant modifications of equipment and maintenance activities included:

- Preventive maintenance on all inner containment mechanical and electrical equipment, including improvement of nitrogen blower bases, motor shaft alignment and replacement of rough bearings; and work on:

- Argon blower shaft seal and oil reservoir

- Argon Vent Vacuum pumps flow control valves

- IFST rupture disc/pressure relief valve remote vent valve

- IFST gasket seal

- In-core thermocouples outer vessel head connectors

- FRED electrical cable and penetration wiring

FIGURE 24 SEFOR Nyquist Diagram for 5 MW 100% Flow (Third Quadrant Only)

FIGURE 25 SEFOR Nyquist Diagram for 5 MW 50% Flow (Third Quadrant Only)

FIGURE 26 SEFOR Nyquist Diagram for 5 MW(t) 30% Flow (Third Quadrant Only)

He³ chamber support structure in transient detector holes

Gaseous radwaste compressors and batch tank gaskets

Man Access Suit exhaust system, neck rings, and docking system

IFA connector vent lines

Freon compressors and motors

Argon compressor low pressure trip

Pump-around-pump high sodium level trip

Argon batch tank header pressure alarm

Refueling Cell Vertical Transfer port/fuel shipping cask shielding

Mechanical and electrical drawings update

Oscillator, position servo system, and oscillator positioner motor

Data Acquisition System, magnetic tape units, and shaft encoder

Auxiliary Intermediate Heat Exchanger dome dye penetrate check

Liquid Argon supply rerouted through the supply NaK bubbler

Reactor Overflow Valve closure alarm

Refueling Cell/Nitrogen Zone intercell leakage

A total of 70 equipment modifications were completed and 148 malfunctions corrected. The malfunctions were distributed as follows;

Mechanical	64
Electrical	50
Instrumentation	34
	<u>148</u>

During the outage, the following tests were accomplished:

Provisional Test Procedure No. 21, "Auxiliary Primary Pump Performance Test"

Provisional Test Procedure No. 27, "Magnetic Flow Meter Calibration"

Provisional Test Procedure No. 31, "Balanced Oscillator Checkout"

Pre-operational Test Nos. 24/29, "Reflector Drive System Control and Instrumentation System (Rev.)"

Pre-operational Test No. 58, "FRED Checkout" (Out-of-cell and in-cell)

After reactor operation and experimental program testing were resumed on June 26, 1970, the following Argon-Nitrogen Inter-cell Leakage tests were performed:

Provisional Test Procedure No. 13, "Reflector Calibration"

Test Procedure II-1, "Radiation Survey Plan" at 2 MW and 5 MW.

Test Procedure II-2, "Auxiliary ABC and Heat Balance" at 1 MW

Test Procedure II-3, "Nuclear Instrumentation and Heat Balance Comparison" at 2 MW and 5 MW

Test Procedure II-4, "Static Tests" at 2 MW and 5 MW

Test Procedure II-5, "Conventional Oscillator Tests" at 2 MW and 5 MW

Test Procedure II-8, "Measurement of Reflector Temperature Coefficient" at 5 MW

Test Procedure II-10, "Monitoring of Primary Shield Temperature and Bottom Shield Temperatures" at 5 MW

Numerous License Test Procedures.

Reactor Operations Data

	This Quarter	To Date
Critical	333.9	— hours
Energy Generated	1218.0	1230.8 MWH(t)
Maximum Power Level	5	5 MW(t)
Startups	22	357
Scrams	5	79

AEC License Examinations were administered to Operations Personnel during the week of July 20. Five candidates took the Senior Operator License examination, and two took the Operator License examination.

Personnel from Brookhaven National Laboratory re-examined a number of selected fuel rods using more refined techniques to determine plutonium content.

The plant radiation survey at 25% power (5 MW) was completed. Other than at a few locations which will require minor additions of shielding materials, the plant shielding was verified to be adequate for full power operation. Sheets of polyethylene and borated polyethylene were added at the Crane Bay Access Hatch and the Reflector Drive Room Door to reduce the neutron leakage to acceptable levels. Some additional gamma shielding may be required at 20 MW outside the primary sodium equipment room door, in the pipe trench at the west wall of the Refueling Cell, and at the Crane Bay Access.

Cover gas samples were obtained at the 5 MW power level and the isotopic constituent identified as Argon-41. No trace of fission gas activity was observed. Preliminary results indicate general agreement with the predicted range of cover gas activity.

Samples of the nitrogen atmosphere were obtained and analyzed with the gamma spectrometer to provide an indication of the isotopic activity. Argon-41 was identified. Evaluation of the radioactive decay on the nitrogen monitor recorders indicated, in addition, a half life near that of nitrogen-16.

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