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**SOUTHWEST EXPERIMENTAL
FAST OXIDE REACTOR
DEVELOPMENT PROGRAM
TWENTY-SIXTH QUARTERLY REPORT
AUGUST - OCTOBER 1970**

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

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SOUTHWEST EXPERIMENTAL FAST OXIDE
REACTOR DEVELOPMENT PROGRAM

Twenty-Sixth Quarterly Report
August–October 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 August 1 through October 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 $\text{PuO}_2\text{--U-238 O}_2$, 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).
1. "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).
  25. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Fifth Quarterly Report, May–July 1970," GEAP-10010-25 (August 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. 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).
17. McVean, H. 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
7.0	Fuel Handling
8.0	Core Design

2.2 TASK 6—REACTIVITY CONTROLS REQUIRED FOR EXPERIMENTATION

Detail design of the poison rod for the fast reactivity excursion device (FRED) is proceeding following worth measurement of the SEFOR oscillator rod.

2.3 TASK 9—FUEL FABRICATION

Fabrication of fuel rods at Nuclear Fuel Services (NFS) continued throughout the report period. The pre-production weld and fuel pellet samples required were completed with satisfactory results and NFS started the production phase.

Quality assurance activities included the evaluation of preproduction samples, an audit of NFS' production facilities prior to production, and the continuous audit by the GE Resident Quality Control Engineer.

Repair work was successfully completed on damaged instrumented fuel rod G7-1.

Tests and evaluation of the performance of the two instrumented fuel assemblies installed in the SEFOR reactor were conducted at a reactor power of 5 MW.

2.4 TASK 10—STARTUP TESTING

In subsequent publications a new value of the effective delayed neutron fraction β , and a new inhour equation will be used for SEFOR. The new quantities will provide consistency between static and oscillator test results, as well as a more accurate evaluation of reactivity effects that are measured and calculated in units of dollars. The old conversions are used for all data tabulated in this report, and for all comparisons of calculated and measured results throughout the remainder of this report. The new value of β will be 0.00327. This is 3.8% larger than the previously used calculated value of 0.00315. Those previously and currently reported *calculated* reactivity effects

that are expressed in units of dollars or cents should thus be decreased by 3.8% to reflect the change in β . The magnitude of previously and currently reported *measured* reactivity effects should be decreased by 2.3% to account for the change in the inhour equation. These two changes cause a 1.5% increase in the magnitude of previously and currently reported ratios of measured to calculated reactivity effects. Specifically, the estimated Doppler $T \frac{dk}{dT}$ of -0.0080 that is discussed in the present report should be increased 1.5% in magnitude to a value of -0.0081 to reflect the new conversion formulas.

After initial operation at 10 MW, the core loading was changed to provide additional reactivity to offset the reactivity feedback introduced by operation at higher power levels. The change in loading was made on August 21, 1970 by replacing a B₄C rod in core locations W3.0-S2.0E with a stainless steel rod. The replacement resulted in an increase in core reactivity of 36.1 cents.

The SEFOR reflector temperature coefficient of reactivity was measured at a power level of 5 MW. The observed change in reactivity due to reflector temperature changes in excess of 100°F was 1.1 cents. The magnitude of this reactivity change is only slightly larger than the previously estimated standard deviation ($\pm 0.5\%$) in reactivity measurements, while the normal change in reflector temperature between zero power and 5 MW is less than 100°F. The reflector temperature coefficient may thus, for all practical purposes, be treated as zero at power levels ≤ 5 MW. The reflector temperature coefficient was measured again at higher power levels and this data is currently being analyzed. Preliminary analysis indicates the effects at higher power are also small.

Previous measurements⁽³⁾ of the oscillator rod incremental reactivity worth using two different measuring techniques differed in some instances by as much as 10%. In order to check these measurements, the oscillator rod worth has been remeasured using yet a third technique. A re-evaluation of the previous measurements and a comparison of the previous and more recent measurements indicates that the apparent differences between the various measurements is not statistically significant.

Results of power, reactivity, and fuel temperature measurements obtained from static and oscillator tests during reactor operation at power levels up to 17.2 MW during the power ascension phase of the experimental program are currently being analyzed. The analysis to date shows good agreement between calculation and experiment and indicates that the SEFOR Doppler coefficient is -0.0080. The preliminary estimate of the standard deviation in this value is ± 0.0015 .

Analysis has been completed of additional tests on the He-3 chambers to investigate previous test results from

the noise measurements to determine λ/β . The additional analysis tends to verify the previously determined value of $\lambda/\beta = 0.205 \times 10^{-3}$ sec.

2.5 TASK 11—TRANSIENT TESTS

The irradiated fuel shipping cask was licensed by the AEC during the reporting period and a permit for transporting the loaded cask was issued by the U.S. Department of Transportation.

The design and fabrication of the fuel transfer equipment and shielding has been completed.

2.6 SEFOR SAFETY AUDIT

Data on the reflector temperatures, the natural circulation tests, reactor coolant outlet temperature following a scram, guinea pig rod geometry, and power coefficient was reviewed for its safety significance.

2.7 SEFOR SUPPORT

The replacement argon purification system was fabricated and shipped to the site. Water flow tests were completed which demonstrated the adequacy of the NaK bubbler tank gas distribution.

Performance measurements were completed for the intermediate heat exchangers, and the heat transfer coefficients compared with predicted values.

Fabrication of a backup reactor vessel sodium level tip probe was completed to the point of final inspection and code testing.

2.8 SEFOR OPERATIONS

Group II Power Ascension Tests were completed thru 17.5 MW. Tests at 19 MW and 20 MW (full power) were delayed until AEC approval was received for changes required to the Technical Specifications which involve criteria for identification of operation anomalies, operation with Guinea Pig rods above 17.5 MW and criteria for operation with known fuel leaks.

Group III Static Tests at power levels up to 16 MW, and manual balanced oscillator tests (first type) at 10 MW were completed. Fuel surveillance tests were performed after 5, 10, 15, and 17.5 MW operation. Diagnostic tests were performed to evaluate the difference in power level as indicated by heat balances and the neutron instrumentation.

Plant and reactor availability for performance of the Experimental Program increased to 71% for the quarter. Forty-two (42) reactor startups were made and 6606 MW hours of energy were generated to increase the accumulated energy generated to 7944 MW hours.

Plant radiation surveys were made and reactor cover gas radioactivity was monitored.

3. TASK 6—REACTIVITY CONTROLS REQUIRED FOR EXPERIMENTATION

3.1 TASK 6.1—FAST REACTIVITY EXCURSION DEVICE (FRED)

outside the reactor refueling cell confirmed its performance characteristics.

3.1.1 Purpose

The purpose of the FRED is to initiate reactor power excursions under controlled conditions. The device developed and fabricated under this task will be used in SEFOR to carry out the planned excursions during the experimental program.

3.1.2 Past Accomplishments

The FRED design, fabrication, and testing has been completed. The device was shipped to SEFOR, and tests

3.1.3 FRED Poison Rod

The experimental program requires poison slugs with three different reactivity worths for the FRED tests. The detail design of the three poison rods are proceeding following the reactivity measurement of the SEFOR oscillator rod.

The FRED poison rods are being designed to enable reactivity adjustment and remote disassembly, should either prove necessary. For the latter purpose, the fuel rod assembly-disassembly equipment in the refueling cell will be modified to accommodate the FRED poison rods as well as the fuel rods.

4. TASK 9—FUEL FABRICATION

4.1 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.

4.2 PAST ACCOMPLISHMENTS

The complete fuel-rod order had been received at SEFOR in previous quarters. A revision in the original purchase order was signed in the previous quarter providing for fabrication by Nuclear Fuel Services (NFS) of new fuel rods, repair of leaking fuel rods, replacement of out-of-specification fuel rods, and supply of additional fuel pellets. A quality assurance program has been started to monitor the fuel rod fabrication at NFS, with a GE engineer in residence at NFS. Five instrumented fuel assemblies (IFA's) have been completed and shipped to the site. Two IFA's have been inserted in the reactor.

The complete core component and fuel-rod order was received at SEFOR in previous quarters.

4.3 CURRENT RESULTS

4.3.1 Fuel Rods

4.3.1.1 Fuel Rod Procurement

Nuclear Fuel Services (NFS) is fabricating 102 additional fuel rods and repairing 22 helium leaking rods under Revision 4 to the original purchase order for fuel rods.

4.3.1.2 Repair of Fuel Rods

The repair of 20 of the 22 He-leaking fuel rods was started. The repair of two of the fuel rods was postponed because of mechanical damage to the rods that could not be corrected by replacing the end plugs. The repair was accomplished by cutting off and re-welding end plugs that were suspected of leaking. A problem was encountered with contamination on the outside of the rods because of insufficient decontamination prior to welding but vigorous decontamination reduced this to acceptable levels. Final processing to complete the repair consists of passivating and final inspection.

4.3.1.3 Fabrication of New Fuel Rods

The makeup of the plutonium-uranyl nitrate master-mix solution was completed and analyzed by NFS. The tests results were satisfactory and the master mix was released for production. A sample of the master mix was sent to GE for independent analysis. The result of this check analysis is reported below in the section on Quality Assurance.

Preproduction activities in the pellet fabrication line were completed. Co-precipitation of the master-mix was

started and a limited quantity of mixed oxide powder was produced. Parameters were established for production with the equipment and preproduction fuel pellets were produced and submitted to GE for approval. Production approval was given by GE (results of the fuel pellet sample analysis is reported under Quality Assurance) and full production of fuel pellets was started. To date three pellet batches have been processed through the line.

The procurement of inert hardware for the fuel rods was completed except for tubing. Rejection of tubing lengths by ultrasonic inspection for wall thickness made it necessary to reorder tubing.

Weld qualification was completed for the end plug weld and the center spacer electron beam weld and these welds have been released for production.

The UO₂ insulator pellets that will be used for the new fuel rods were produced when the initial fuel rods were fabricated. Samples from the insulator pellet lots to be used were submitted to GE for analysis and approval.

4.3.2 Quality Assurance

The GE Quality Assurance Audit Plan for the SEFOR fuel rod production was published. The NFS fabrication activities were audited by the GE Resident Engineer during the fuel pellet preproduction and the fuel rod repair. In addition, subcontractors who are performing work for NFS on fuel rod items were audited. Orbit Industries, Cleveland, Ohio was visited by the GE Resident Engineer to review their procedures for inspecting the 316 stainless steel tubing. A GE representative was present at Buehler Corp., Indianapolis, Indiana when some of the center spacer electron beam qualification welds were made and evaluated.

A quality systems engineer from the GE Quality Assurance organization audited the records and files being maintained by the GE Resident Engineer.

4.3.2.1 Round Robin Analysis

A round robin chemical analysis program was initiated at the start of production activities for fuel rod fabrication. Samples of a synthetic master-mix solution and fuel pellets were analyzed by the laboratories at GE and NFS to assure that reasonable agreement would be obtained for the attributes listed in the fuel pellet specification. The results of these analyses are shown in Table 1 and indicate good agreement between the two laboratories.

4.3.2.2 Master-Mix Analysis

A sample of the production master-mix solution (No. 256-B-110) was submitted to GE for an independent analysis of the plutonium and uranium content. The GE and NFS results given in Table 2 show good agreement with the value calculated from the separate solutions used in the master-mix makeup.

Table 1

SEFOR FUEL ROD PRODUCTION GE-NFS ROUND-ROBIN CHEMICAL ANALYSIS RESULTS

Sample		Results			
		General Electric		Nuclear Fuel Services	
1.	Synthetic Master Mix				
a.	Grams/Pu gram solution	0.02476 0.02491 0.02487 0.02497	0.02490 0.02479 0.02478 0.02473	Avg 0.02484	0.02473 0.02475 0.02475
					Avg 0.02473
b.	Grams U/gram solution	0.08842 0.08878 0.08796	0.08838 0.08839 0.08827	Avg 0.08837	0.08749 0.08776
					Avg 0.08768
2.	(Pu-U)O ₂ Fuel Pellets				
a.	O/M Ratio	1.9914 1.9902			1.990
b.	Pu Total Pu + U	0.2676 0.2673	0.2669 0.2683	Avg 0.2677	0.2704 0.2699 0.2700
					0.2700
					Avg 0.2700
c.	Moisture, ppm	< 1, 3			
d.	Total gas microliters/gram	66.3	69.1 69.4	(1350°C) (1650°C)	72
e.	Impurities, ppm				
	H	—			5.8
	B	<1	<1		<0.2
	N	—			16
	C1	11			<5
	F	19			<5
	Al	50	50		846
	Ag	<0.1	<0.1		0.8
	Ca	<25	<25		7
	Cd	<2	<2		<0.2
	Cr	14	14		20
	C	161	141		146
	Cu	30	30		22
	Fe	200	150		144
	Mg	4	4		12
	Mn	10	10		8
	Mo	<10	<10		5
	Na	<15	<15		110
	Ni	<24	<24		41
	Pb	<50	<50		2
	Si	12	8		57
	Sn	<1	<1		<2
	Ti	—	—		<250
	V	<3	<3		<5
	Zn	<10	<10		27
	Li	<10	<10		
	Be	<0.3	<0.5		Total Metallic Impurities
	Co	<1	<1		1559 ppm
	P	<50	<50		

Table 2
ANALYSIS OF SEFOR MASTER MIX SOLUTION 256-B-110

<u>Attribute</u>	<u>NFS Analysis</u>	<u>GE Analysis</u>
Grams Pu/gram solution	0.03597 ⁽¹⁾	0.03591 ⁽²⁾
Grams U/gram solution	0.13796 ⁽¹⁾	0.13865 ⁽¹⁾
$\frac{\text{Pu}}{\text{Pu+U}}$ ratio	0.2068 ⁽³⁾	0.2057 ⁽³⁾

(1) Average of six readings

(2) Average of seven readings

(3) Based on weights (NFS) and assays (ARHCO & NBL) of Pu & U solutions used in the master-mix make up, the $\frac{\text{Pu}}{\text{Pu+U}}$ ratio is 0.2046.

4.3.2.3 Fuel Pellet Preproduction Samples

Ten preproduction fuel pellets submitted by NFS as part of the preproduction phase were received at GE and evaluated against the fuel pellet specification. The results of the analyses performed by GE and NFS, and the comparison against the specification are shown in Table 3. The results were satisfactory and the fuel pellet production was approved.

4.3.2.4 UO₂ Insulator Pellet Samples

The UO₂ insulator pellet samples submitted by NFS were received at GE and analyzed. The results of the analysis and the comparison against the insulator pellet specification is shown in Table 4. The results were satisfactory and the insulator pellet lots were approved.

4.3.2.5 End Plug Weld

Ten end plug weld samples submitted by NFS as part of the preproduction phase of fuel rod fabrication were evaluated against the requirements of the fuel rod specification. The weld samples met the requirements of the specification and the end plug weld parameters represented by the weld sample were approved for production.

4.3.2.6 Center Spacer Weld

Evaluation of samples from the first two qualification runs made on the center spacer electron beam weld revealed cracks in one of the dual welds at the tube to spacer interface. The cracks appeared to be caused by the clearance between the tube inside diameter and the center spacer diameter. The next qualification run was made with a single electron beam weld that consumed the lip of the spacer. Metallographic examination of these welds indicated that no cracks were present and the weld width met the structural requirement of the weld. A single electron beam center spacer weld was approved for production of fuel rods.

4.3.3 Instrumented Fuel Assemblies

The repair work on the damaged fuel rod (G7-1) discussed in the preceding report was successfully completed. Support effort for this repair included a stress analysis of the modified design and welding process development. The electrical resistance readings of the thermocouples and the resistance of the insulation on the bare wire thermocouple (non-grounded) indicated a successful repair. A temporary guard will be installed during the remaining assembly operations to prevent high stress in the repair braze joint as a result of handling. The parallel path effort of building a new fuel rod (needed if the repair effort failed) has been cancelled as a result of the successful repair.

Work on completing the two fuel rods (one repaired) was scheduled, with completion of these two rods expected in November.

Diagnostic tests were performed on the sensors in the two IFA's which are in the core in order to establish the credibility of the data and to determine possible corrections of malfunctions. The integrity of the IFA system cabling as well as the data acquisition system (DAS) hardware/software was verified by several tests. The circuit resistances of the installed IFA's (VNC 25 and VNC 29) were measured, and a comparison with the previous recorded values indicated some of the sensor circuit resistances have increased significantly. Two fuel thermocouples showed open circuits (VNC 29 Pin Pair TU, VNC 25 Pin Pair RS), and two showed slightly increased resistance at an undetermined location (VNC 29 Pin Pair EF, and RS). The slightly increased electrical resistance did not appear to produce anomalous temperature data.

The sensor millivolt signals were recorded on a multi-point recorder and were found to correspond to the values being recorded digitally and typed out by the DAS.

The electrical resistance of the sensors in the stored, unused IFA's (VNC 24, 26, 28) were accurately measured (± 0.1 ohm) and recorded for use after reactor installation.

Table 3
SEFOR FUEL PELLETS
PREPRODUCTION PELLETS
GE Specification 22A1545 Rev. 4

G.E. RESULTS

<u>Specification Requirement</u>	<u>Pellet No.</u>									
	1	2	3	4	5	6	7	8	9	10
Diameter										
0.874" to 0.876"	0.8757	0.8756	0.8754	0.8756	0.8757	0.8749	0.8753	0.8754	0.8755	0.8755
Length 5/8"±3/32"										
0.531" to 0.719"	0.6507	0.6434	0.6548	0.6565	0.6653	0.6653	0.6623	0.6680	0.6560	0.6556
Dish Depth										
0.012" to 0.018"	0.0121	0.0124	0.0129	0.0125	0.0118	0.0132	0.0124	0.0122	0.0131	0.0127
I.D. Marking (V)	ok	ok	ok	ok	ok	ok	ok	ok	ok	ok
Surface Appearance										
Photograph	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓
Comp. with Crack Std.	ok	ok	ok	ok	ok	ok	ok	ok	ok	ok
Density 9.99 to 10.40 g/cc	10.31	10.31	10.35	10.36	10.30	10.33	10.42	10.32	10.32	10.32
Total Gas <50 µL/g	50.7	27.1	20.2	19.2	34.1	21.1	14.8	48.1	25.5	28.5
Moisture 30 ppm Max	<1	1	<1	2	<1	<1	<1	<1	<1	<1
O/M 1.97 to 1.99	1.9815	1.9790	1.9807	—	—	—	—	—	—	—
<u>Pu239+241</u> Pu+U 18.7±0.3%	18.75	18.72	18.95	—	—	—	—	—	—	—
Pu Total										
% Pu of net weight*†	18.00	17.97	18.19	—	—	—	—	—	—	—
U Total										
% U of net weight†	69.76	69.80	69.56	—	—	—	—	—	—	—
U235										
0.204%±0.01 of Total U	0.21	0.21	0.21							
Homogeneity										
First Sample % Pu	18.13	18.21	18.19	The Pu concentration in a 0.1 gram <u>SAMPLE</u> taken at any region of the pellet shall not deviate more than 1% from the average Pu concentration in the pellet.						
Second Sample % Pu	18.14	18.25	18.18							

<u>Impurities</u>		<u>1</u>	<u>2</u>	<u>3</u>
	<u>PPM Max</u>			
H	5	3.85	2.07	1.48
B	10	<1	<1	<1
N	200	<200	<200	<200
C	250	25	22	27
C1	25	<2	3	<2
F	25	2	2	1

Table 3 (Continued)

G.E. RESULTS					
Specification Requirement	Pellet No.				
	1	2	3		
<u>All Other Elements</u>					
<u>2500</u>					
Ag	<0.1	<0.1	<0.1		
Al	<4	<4	<4		
B	<1	<1	<1		
Be	<0.5	<0.5	<0.5		
Ca	<25	<25	<25		
Cd	<2	<2	<2		
Co	<1	<1	<1		
Cr	24	24	24		
Cu	11	16	20		
Fe	200	220	220		
Mg	4	4	4		
Mn	4	4	4		
Mo	<10	<10	<10		
Ni	40	40	40		
Pb	<1	<1	<1		
Si	<6	<6	<6		
Sn	<1	<1	<1		
V	<3	<3	<3		
Zn	<10	<10	<10		
P	<50	<50	<50		
Li	<10	<10	<10		
Na	<15	<15	<15		
<u>Characteristics</u>	<u>NFS Results</u>				
<u>Density</u>					
Whole Pellet	10.23, 10.38, 10.28 g/cc				
Within Pellet	<u>Whole</u>	<u>Center</u>			
	10.23	10.17			
	10.38	10.35			
	10.28	10.13			
	10.37	10.17			
	10.38	10.28			
	10.32	10.14			
Single Phase of Mixed Oxide	Acceptable				
O/M Ratio	1.987	1.984 before outgassing			
	1.980	1.976 after outgassing			
<u>Pu total</u>			<u>Center</u>	<u>Edge</u>	
<u>Pu+U</u> Ratio	0.2050	0.2051	0.2051	0.2050	0.2052

Table 3 (Continued)

<u>Characteristics</u>	<u>NFS Results</u>	
Homogeneity	Acceptable	
Volatiles	0.033 cc/gram	
Landwidth	<u>Letter Side</u>	<u>Blank Side</u>
	0.043; 0.044"	0.039; 0.041"
	0.043; 0.039"	0.039; 0.044"
	0.041; 0.043"	0.037; 0.041"
Impurities (ppm)		
H - 2.6	Cr - 150	Sn - 3
B - <0.25	Cu - 11	Ti - <250
N -	Fe - 215	V - <5
C - 53	Mg - 23	Zn - 35
Cl - 6	Mn - 8	
F - <5	Mo - 6	
Al - 73	Na - 10	
Ag - 0.5	Ni - 100	
Ca - 30	Pb - <5	
Cd - 0.2	Si - 45	

*Check Analysis-Round-Robin Sample

Rerun of Round-Robin Sample for Pu = 23.33% of net weight.

Original Analysis of Round-Robin

Sample; Pu = 23.43% of net weight

†NBL also performed Pu and U analyses on samples from these three pre-production pellets, their results are:

	<u>Pellet No.</u>		
	<u>1</u>	<u>2</u>	<u>3</u>
% Pu of net weight	18.14	18.18	18.16
	18.14	18.16	18.15
%U of net weight	69.98	69.96	70.02
	70.00	69.94	70.01
	69.98	70.02	

Table 4
SEFOR FUEL RODS
UO₂ INSULATOR PELLET PRODUCTION SAMPLES

Attribute	Specified GE 22A1583 Rev 2	NFS Analysis				NFS Overcheck	GE Analysis	
		Lot 1		Lot 2			Pellet #3	
Density	9.983–10.42 g/cc	Ave. 10.15 Range 9.97 10.32	Ave. 10.27 Range 10.17 10.38			Ave. 10.42 10.46 10.34	10.46	
O/M Ratio	2.00 –2.03	2.029 2.029	2.029 2.034		Lot 1-2.010 Lot 2-2.013		2.00	
U-235/U Ratio	0.22 ± 0.01	0.228	0.228				0.21	
Impurities*								
A1	500	24	20	8	6		<4	
B	6	<0.2		<0.2			<1	
C	150	<25		<25		Lot 1 59 Lot 2 71	39	
Cd	20	<0.01		<0.01			<2	
Ca	25	<20		<20			<25	
C1	25	<10		<10		Lot 1 4 Lot 2 3	3	
Cr	500	9	6	5	3		<3	
Cu	500	1		1			<<0.5	
F	75	10		10			4	
Fe	500	26	23	24	18		<14	
Pb	500	<1		<1			<1	
Mg	25	9	8	7	6		<2	
Ni	500	18	10	26	22		<24	
Si	100	95	80	75	65		60	
N	200	<10		<10			<200	
Na	500	<40		<40			<15	
Ag	500	<0.1		<0.1			<0.1	
Sn	500	<1		<1			<1	
V	500	<2.5		<2.5			<3	
Zn	500	<25		<25			<10	
Total Impurities	<2500						<238.1	

*Maximum parts per million parts uranium

Table 4 (Continued)

<u>Attribute</u>	<u>Specified GE Drwg 159A2750</u>	<u>NFS Inspection</u>			<u>GE Evaluation</u>	
		<u>Lot</u>	<u>Ave</u>	<u>Range</u>		
Diameter	0.882" to 0.884"	1	0.8832	0.8841/0.8827	Pellet 1	0.8830"
		2	0.8828	0.8835/0.8820	2	0.8827"
					3	0.8834"
Length	0.312" to 0.438"	1	0.3578	0.3683/0.3508	Pellet 1	0.3543"
		2	0.3547	0.3671/0.3424	2	0.3497"
					3	0.3586"
Dish Depth	0.007" to 0.017"	1(A)	0.012	0.010/0.015	Pellet 1 Top	0.0077"
		(B)	0.013	0.010/0.015	Bot	0.0109"
		2(A)	0.012	0.010/0.014	Pellet 2 Top	0.0092"
		(B)	0.012	0.010/0.014	Bot	0.0099"
					Pellet 3 Top	0.0101"
					Bot	0.0081"
Weight	Not Specified	Not Measured			Pellet 1	36.62 g
					2	35.66 g
					3	37.12 g

5. TASK 10—STARTUP TESTING

5.1 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.

5.2 PAST PERFORMANCE

Analysis of all experimental data collected during performance of the experimental program at zero power was completed. Preliminary analysis of results obtained during initial operation at power levels up to 5 MW indicated that the observed reactivity effects and fuel temperature changes are consistent with the calculated Doppler reactivity coefficient ($T \frac{dk}{dT}$) of -0.0082.

5.3 CURRENT RESULTS

5.3.1 New Reactivity Conversion Formulas for SEFOR

In subsequent publications a new value of the effective delayed neutron fraction β , and a new inhour equation will be used for SEFOR. The new quantities are based on the analysis of the zero power oscillator tests. They will provide consistency between static and oscillator test results, as well as a more accurate evaluation of reactivity effects that are measured and calculated in units of dollars. The old conversions are used for all data tabulated in this report, and for all comparisons of calculated and measured results throughout the remainder of this report. The adjustments required for use of the new conversion formulas are described in the subsequent paragraph.

The new value of β will be 0.00327. This is 3.8% larger than the previously used calculated value of 0.00315. Those previously and currently reported *calculated* reactivity effects that are expressed in units of dollars or cents should thus be decreased by 3.8% to reflect the change in β . The magnitude of previously and currently reported *measured* reactivity effects should be decreased (see below) by 2.3% to account for the change in the inhour equation. These two changes cause a 1.5% increase in the magnitude of previously and currently reported ratios of measured to calculated reactivity effects. Specifically, the estimated Doppler $T \frac{dk}{dT}$ of -0.0080 that is discussed in the present report should be increased 1.5% in magnitude to a value of -0.0081 to reflect the new conversion formulas.

These new conversion formulas are based on the fitted delayed neutron parameters obtained from the analysis of the zero power oscillator tests reported in reference 1. The estimated value of β obtained from this analysis is 0.00327. This is 3.8% higher than the previously

used calculated value of 0.00315. Since calculated reactivity effects must be divided by β to convert to units of dollars, previously calculated values that are expressed in dollar units may be updated by decreasing their magnitude 3.8%. Calculated reactivity values which are expressed in absolute or percentage units are unaffected by the change in β .

All measured reactivity values in SEFOR are reported in units of dollars or cents. These values are based, either directly, or indirectly through the calibrated reflector control, on the relationship between measured asymptotic reactor period and reactivity. This relationship, the inhour equation, involves the same delayed neutron parameters as the zero power reactivity transfer function. The parameters in the fit of the theoretical model to the measured zero power reactivity transfer function are tabulated in reference 1. Those parameters were used in the inhour equation⁽²⁾ to compare with the previously⁽²⁾ used parameters that were based on calculated neutron contributions from the fissionable isotopes. The resulting relationships between asymptotic period and reactivity are compared in Table 5.

All of the reactivity measurements in SEFOR, including the calibrations of reflector segments which are used in turn to measure reactivity changes, are made^(1,2) with reactor periods between ~80 and ~450 seconds, with the majority of these having periods of ~100 seconds with a measurement accuracy of ~1%. The data in Table 5 show that for a 100 second period, the new conversion formula gives a reactivity that is 2.3% less than the old. The tabulated values also show that the ratio of the reactivity obtained from the new inhour equation to the reactivity obtained from the old equation is essentially constant (within 0.2%) over the range of periods between ~80 and ~500 seconds. The constancy of this ratio allows all previously reported measured reactivity values to be updated by applying only one reduction factor of 2.3%. It also allows the previously determined relationships between reactivity and reflector position to be used for future measurements provided the reactivity values are reduced by 2.3%.

When the new value of β and the new inhour equation are used to convert a measured reactivity in dollars to a reactivity in absolute units, the number is ~1.5% larger, as indicated by the last column of Table 5 for a 100 second period, than was previously the case. The last column also illustrates the increase necessary to update the magnitude of previously reported ratios of measured to calculated reactivity effects.

5.3.2 Core Loading

After initial operation at 10 MW, the core loading was changed to provide additional reactivity to offset the reactivity feedback introduced by operation at higher power levels. The change in loading was made on August 21, 1970

Table 5
COMPARISON OF NEW AND OLD SEFOR
REACTIVITY CONVERSION FORMULAS

$$\beta_{\text{new}} = 0.00327$$

$$\beta_{\text{old}} = 0.00315$$

Period (sec)	Reactivity (ρ)*		$K_{\text{new}}/K_{\text{old}}$	$K_{\text{new}}\beta_{\text{new}}/K_{\text{old}}\beta_{\text{old}}$
	K_{new}	K_{old}		
50	14.360	14.669	0.9789	1.016
60	12.643	12.920	0.9786	1.016
70	11.306	11.558	0.9782	1.015
80	10.233	10.464	0.9779	1.015
90	9.3512	9.5648	0.9777	1.015
100	8.6125	8.8110	0.9775	1.015
200	4.8382	4.9546	0.9765	1.014
300	3.3727	3.4552	0.9761	1.013
400	2.5902	2.6541	0.9759	1.013
500	2.1029	2.1551	0.9758	1.013
600	1.7702	1.8142	0.9757	1.013

*Calculated from inhour equation using the new and old delayed neutron parameters respectively.

by replacing a B₄C rod in core location W3.0-S2.0E with a stainless steel rod. The replacement resulted in an increase in core reactivity of 36.1 cents. This difference (58.06 - 21.96 = 36.10 cents) is evidenced by the average values in Tables 6 and 7. The new core loading was designated Assembly I-J. This loading is illustrated in Figure 1. The previous loading, Assembly I-I, is illustrated in reference 1.

5.3.3 Reflector Temperature Reactivity Coefficient at 5 MW

The SEFOR reflector temperature coefficient of reactivity was measured at a power level of 5 MW. The observed change in reactivity due to reflector temperature changes* in excess of 100°F was 1.1 cents. The magnitude of this reactivity change is only slightly larger than the previously⁽²⁾ estimated standard deviation ($\pm 0.5\%$) in reactivity measurements, while the normal change in reflector temperature between zero power and 5 MW is less than 100°F. The reflector temperature coefficient may thus, for all practical purposes, be treated as zero at power levels ≤ 5 MW. The reflector temperature coefficient was measured again at higher power levels and this data is

currently being analyzed. Preliminary analysis indicates the effects at higher power are also small.

5.3.3.1 Reflector Temperature Measurements and Analysis

To enable the determination of the Doppler coefficient, a number of experiments are performed in SEFOR to determine the relationship between reflector position (reactivity) and plant variables such as vessel inlet temperature, flow, etc. The measurements described here were performed to determine the relationship between reactivity and the temperature of the radial reflectors. In these experiments the reflector temperature was first increased by decreasing the gas coolant flow around the reflectors. The reflector temperature was then decreased by increasing the gas flow to its original value. During the measurements a fine reflector position was adjusted to maintain a constant flux level. The average variation in flux level was held to 1 part in 700, while the variation in average core coolant (sodium) temperature during the measurements was held to less than 1°F. The small reactivity corrections for coolant temperature variations were accounted for. The flux variations should produce a negligible effect.

*These temperature changes were measured by thermocouples located near the center of a raised coarse reflector.

Table 6
LOW POWER (0.06 MW) CRITICAL POINTS IN ASSEMBLY I-I

Date	Reflector #3		Reflector #8		Reactor Coolant [†] Temperature (°F)	Relative Reactivity* at 760°F and Zero Power (%)
	Position (cm)	Worth (%)	Position (cm)	Worth (%)		
8/6/70	0	0	33.97	44.66	735	57.31
8/14/70	0	0	41.01	58.21	760	57.11
8/16/70	0	0	39.56	55.36	750	59.76
Average =						58.06*

*Reactivity temperature corrections were made with a coefficient⁽¹⁾ of $-0.55\%/^{\circ}\text{F}$, and a reactivity correction of 1.10 cents to the zero power point was made with a power reactivity coefficient at low power of $-18.4\%/ \text{MW}$. The estimated uncertainty in the reactivity is $\pm 1.5\%$.

[†]The estimated uncertainty in temperature is $\pm 2.5^{\circ}\text{F}$.

Table 7
LOW POWER (0.06 MW) CRITICAL POINTS IN ASSEMBLY I-J

Date	Reflector #3		Reflector #8		Reactor Coolant [†] Temperature (°F)	Relative Reactivity* at 760°F and Zero Power (%)
	Position (cm)	Worth (%)	Position (cm)	Worth (%)		
8/23/70	0	0	18.56	18.95	752.4	22.03
8/23/70	0	0	18.23	18.48	751.7	21.95
8/25/70	0	0	21.22	22.90	761	21.25
8/30/70	0	0	20.30	21.51	756	22.61
Average =						21.96*

*Reactivity temperature corrections were made with a coefficient⁽¹⁾ of $-0.55\%/^{\circ}\text{F}$ and a reactivity correction of 1.10 cents to the zero power point was made with a power reactivity coefficient at low power of $-18.4\%/ \text{MW}$. The estimated uncertainty in the reactivity is $\pm 1.5\%$.

[†]The estimated uncertainty in temperature is $\pm 2.5^{\circ}\text{F}$.

Figure 1. Core Loading for Assembly I-J

The averages of data taken at the lowest and highest equilibrium reflector temperature readings are shown in Table 8. The data in set number 1 are the averages of five successive measurements with the data acquisition system static test program* between the times of 1344 and 1425 on 7/9/70. Data in set 2 are the averages of three readings at about 0035 on 7/10/70 and the data in set 3 are averages of three successive readings at 0250 on 7/10/70. The long wait between equilibrium conditions was necessitated by the slow response of reflector temperature to changes in gas coolant flow. The time dependent response of thermocouple No. 56 between steady state values of 290°F and 149°F is shown in Figure 2 for a rapid increase from 40% to 100% coolant gas flow at time zero. The response to a decrease in flow was even slower.

The reflector temperature readings shown in Table 8 were taken from two thermocouples located near the center of coarse reflector number six. This reflector was maintained in the completely raised position during the measurements and the thermocouples were in the vicinity of the peak reflector gamma and neutron flux. As a result of their location these thermocouples are among those most responsive to power and gas coolant flow changes, and the observed temperatures were indicative of the highest temperatures in the reflector region.

The data of Table 8 indicate that the core reactivity decreases very slightly with increasing reflector temperature, and that the effect is about -0.01% /°F when the temperature is monitored by these particular thermocouples. The measured reactivity is of the same magnitude as the previously⁽²⁾ estimated standard deviation ($\pm 0.5\%$) in reactivity measurements at zero power however, and since the normal change in reflector temperature between zero power and 5 MW is 60°F or less, the reflector temperature coefficient may be treated as zero at power levels ≤ 5 MW. Preliminary analysis of more recent measurements indicates that the reflector temperature coefficient of reactivity is also small at higher power levels.

5.3.4 Oscillator Rod Worth Measurements

Previous measurements⁽³⁾ of the oscillator rod incremental reactivity worth using two different measuring techniques differed in some instances by as much as 10%. In order to check these measurements, the oscillator rod worth has been remeasured using yet a third technique. A re-evaluation of the previous measurements and a comparison of the previous and more recent measurements indicates that the apparent differences between the various measurements is not statistically significant.

The two earlier rod worth measurements included a "dynamic calibration" in which the rod worth was determined from a comparison of the measured and theoretical zero power reactivity transfer functions; and a "reflector-swap calibration" where the worth was determined by compensating the reactivity effects of changes in oscillator rod position with movement of a previously calibrated reflector. In the third (recent) set of measurements, the asymptotic neutron flux doubling times resulting from prescribed changes in oscillator rod position were measured and converted into asymptotic periods and then into reactivities with the SEFOR in-hour equation of reference 2.

The results of the third set of measurements using the "period technique" are shown in Table 9. They are compared with the previous reflector swap measurements and with a calculation in Figure 3. The calculated curve is a reactivity perturbation calculation obtained using the BISYN⁽⁴⁾ flux synthesis program and is normalized to a value of 34.0¢ at the zero[†] position. The incremental worths, cents/inch, for the different techniques with the lower tip of the rod at an indicated position of 5.5 inches are listed in Table 10. One of the incremental worths was determined directly⁽¹⁾ from a fit to the zero power reactivity transfer function, and the others were obtained using a four point interpolation formula with the reflector swap and asymptotic period measurements. In the previously reported⁽³⁾ comparison, only the average of the values for the reflector swap measurements, or, 2.51 cents per inch, was compared with the worth obtained from the oscillator tests.

As the data in Table 10 indicates, the average incremental worth from the reflector swap and asymptotic period measurements is about 7% less than the incremental worth obtained from the oscillator tests. The average values agree within the limits of the indicated uncertainties however, and there is apparently no significant difference between the incremental worth obtained from the different measurement techniques.

5.3.5 Fuel Temperature Measurements

Fuel temperatures have been measured at up to seven different core locations at power levels up to 17.2 MW. The early fuel temperature measurements at power levels up to 5 MW were used to normalize the model used for fuel temperature calculations. This was accomplished by adjusting the fuel to clad heat transfer coefficient to provide agreement between measured and calculated temperatures. The normalized model was then used to predict fuel temperatures at higher power levels. As previously indicated

*The data for reflector thermocouple No. 60 were taken from Test Plan Procedure II-8 data sheets for all three sets of measurements.

†When the indicated position of the oscillator rod is zero, the actual position of the lower tip of the oscillator rod is estimated to be 1-7/8 inches below the core midplane. This places the lower end of the B₄C region of the oscillator rod 5/16 inches below the core midplane.

Figure 2. Response of a Reflector T/C to a Rapid Increase in Coolant Flow

Table 8
REFLECTOR TEMPERATURE-REACTIVITY MEASUREMENTS

Data Set Number	Average Core Temperature (°F)	Reflector No. 3		Core Temperature Reactivity** Correction to 760° F (¢)	Reflector Thermocouple No. 56 (°F)	Reflector Thermocouple No. 60 (°F)	Relative*** Reactivity (¢)
		Position (cm)	Worth† (¢)				
1	760.4	42.20	-61.14	+0.22	152	130	+0.11 ± 0.5*
3	759.9	42.17	-61.08	-0.06	149	128	-0.11 ± 0.5*
2	760.8	42.94	-62.56	+0.44	290	252	-1.09 ± 0.5*

*This is the previously estimated standard deviation in a reactivity measurement at zero power. Data are now being collected to obtain an estimate of the deviation at power.

†This is the negative reactivity which would be inserted if the reflector were moved to its completely lowered position.

**Core temperature reactivity corrections made using a reactivity coefficient of -0.55¢/°F.

***The relative reactivity was adjusted to make the average of the values for set one and three equal to zero.

Table 9
OSCILLATOR ROD WORTH FROM PERIOD MEASUREMENTS

Change in Oscillator Rod Position		Doubling Time (sec)	Incremental Reactivity Change (¢)	Oscillator Rod Position	Cumulative Reactivity Worth (¢)
From (in.)	To (in.)				
0.04	2.50	85.7	7.43	20.00	0.00
2.50	5.00	95.5	6.80	13.00	-3.83
5.00	7.50	104.6	6.30	10.00	-8.44
7.50	10.00	130.4	5.22	7.50	-13.66
10.00	13.00	150.5	4.61	5.00	-19.96
13.00	20.00	185.5	3.83	2.50	-26.76
				0.04	-34.19

Figure 3. Oscillator Rod Reactivity Worth vs. Position

Table 10
DIFFERENTIAL OSCILLATOR ROD WORTH
WITH TIP AT 5.5 INCHES

<u>Measurement</u>	<u>Incremental Worth</u> (ϕ /in.)
Reflector Swap** (inserting rod)	2.68
Reflector Swap** (withdrawing Rod)	2.35
Asymptotic Period	2.61
Average	$2.55 \pm 0.20^*$
Zero Power Transfer Function ⁽¹⁾	2.74 ± 0.11

*This is 1.96 times the square root of one sixth of the sum of the squares of the individual deviations from the mean and is essentially the estimated standard deviation in the mean for a system with only two degrees of freedom.

**The experimental data is tabulated in reference 3.

by the preliminary analysis in reference 1, only one fuel to clad heat transfer coefficient was required to adequately describe all the early fuel temperature measurements at the seven core locations. The normalization of the fuel temperature calculational model and the comparison of predicted and measured temperatures at higher power levels are discussed below. The use of this model to determine the power-fuel temperature relationship necessary for a comparison of measured and calculated Doppler reactivity effects is discussed in Section 7.

5.3.5.1 Fuel Temperature Instrumentation

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 5 and 6. The thermocouples are located along the center line of four different fuel rods in two instrumented fuel assemblies⁽⁷⁾ (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 are referred to as lower T/C's, while those 15-1/2 inches above the midplane are 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. Each IFA contains an "inner fuel rod" and an "outer fuel rod", with the inner rod being that closest to the core center line.

5.3.5.2 Normalization of the Model

The early fuel temperature measurements at power levels up to 5 MW were previously reported in reference 1.

As discussed in reference 1, the differences between the measured fuel and sodium temperatures at all locations were observed to vary linearly with power, and the magnitudes of the local changes at the different T/C locations were correlated with the local power densities at those locations. This indicated that one and the same model could be used to calculate all fuel temperatures in the core. The normalization of the model to the early 5 MW data is described below.

Using the best estimate of fuel thermal conductivity, power density, rod dimensions, thermal properties of the clad and coolant, etc., the fuel to clad heat transfer coefficient was adjusted to provide agreement between calculated central fuel rod temperatures and the temperatures measured during initial operation at power levels up to 5 MW. The fuel to clad head transfer coefficient, rather than some other parameter, was used to normalize the data, because it is the one temperature-influential-parameter in SEFOR that is least amenable to independent calculation or measurement.

The fuel thermal conductivity used in the normalization employed Schmits⁽⁸⁾ data for fuel temperatures less than 1100°F, and the following relationship⁽⁹⁾ for temperatures between 1100°F and 3000°F.

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

where K is in units of watts/cm²·°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 Btu/ft²·hr·°F, and the power distribution discussed in reference 1 was used. This distribution has a core peak to average power ratio of 1.7 and a core peak power density of 150 W/cm³ at 20 MW. The local power density at a given thermocouple location was determined from the following relation

$$\rho = \frac{150}{20} (0.805)PR$$

where ρ is the desired power density in W/cm³; P is the power level in MW; R is the appropriate power density from row two of Table 14 of reference 1; 150/20 is the ratio of peak core power density to the total reactor power; and 0.805 is the factor in the footnote of Table 14 of reference 1. This relationship is illustrated in Figure 4.

The values for the fuel to clad heat transfer coefficient obtained from analysis of the initial data through 7/11/70 at power levels to 5 MW can be expressed by the following polynomial

$$h = 130 + 1.75 \rho + 0.0277 \rho^2$$

Figure 4. Local IFA Fuel Rod Power Densities at the T/C Locations

where h is in units of $\text{Btu/ft}^2\text{-hr-}^\circ\text{F}$ and ρ is in W/cm^3 . The coefficient obtained using this expression deviates from the expected coefficient by no more than 20% at power densities above ~ 30 watts/cc, but is 40% lower than expected at zero power. Fuel rod temperatures calculated using this expression with the fuel thermal conductivity, etc. described above are illustrated in Figure 5. Since the average fuel temperature at higher power levels is primarily determined by the thermal conductivity rather than the fuel-cladding gap coefficient (e.g. the difference between average and surface temperature in Figure 5 is only dependent on fuel conductivity), and since the fitted gap coefficient did not deviate greatly from the expected value, it was used in the fuel temperature and Doppler models described below.

5.3.5.3 Comparison of Measured and Calculated Temperatures

The measured increases in fuel temperature with power are tabulated in Table 11. As indicated by the data in the table, one T/C (#17) was erratic after initial operation at 2 MW, and another T/C (#3) was erratic after initial operation at 5 MW. All of the data in Table 11 was taken with an average core sodium temperature of $\sim 760^\circ\text{F}$, except for the few measurements which are noted in the table.

Five of the fuel thermocouples are compared with the temperatures calculated using the model described above in Figures 6 through 10.

Good agreement between calculated and measured temperatures was observed for the thermocouples in the central IFA up to power levels of 10 MW or higher. Above 10 MW T/C #2 became somewhat erratic as indicated by the data, while T/C #4 agreed with calculations for initial operation up to ~ 15 MW, but began to indicate lower fuel temperatures thereafter. T/C #1 has indicated slightly lower temperatures than predicted above the 5 MW power level. Subsequent to initial operation at 5 MW, Figures 9 and 10 show that fuel thermocouples #18 and 20 in the peripheral IFA increased above the predicted temperatures (even while the power level remained constant) and then, except for some lower temperatures for T/C #18 near 13 MW, remained above the calculated values. Reasons for this behavior are being investigated.

5.3.6 Static Test Results

5.3.6.1 Power Reactivity Measurements to 17.2 MW

The power levels, average coolant temperatures, and reflector positions obtained at average coolant temperatures of $\sim 760^\circ\text{F}$ and power levels up to 17.2 MW during the performance of the static tests in the initial power ascension phase of the SEFOR experimental program⁽¹⁰⁾ are listed in Table 12. Also shown in Table 12 are the reactivity effects due to power changes at a constant average coolant temperature of 760°F . These tabulated reactivity values have been corrected for the variations in average sodium

temperature using previously measured reactivity temperature coefficients and an approximation to the theoretical dependence of the average core coolant temperature reactivity coefficient on reactor power. At 760°F coolant temperature, the correction can be written as $-0.36 - 0.19 (1220)/(1220+65P)$ cents/ $^\circ\text{F}$, where the power level, P , is in units of MW. The first term in this expression is the expected non-Doppler contribution while the latter term approximates the Doppler effect observed (and calculated) at power levels up to 17.2 MW. Some of the values in Table 12 are averages of measurements performed over a time interval of several days. The individual measurements from which these averages were obtained are listed in Tables 6, 7, 13, 14, and 15. The thermal power levels and average coolant temperatures were calculated using the actual flow and temperature data in Table 16.

A comparison of the reactivity at zero power in early July (Table 13) with that in early August (Table 6) indicates a loss in reactivity during this period of $\sim 5.5 \pm 1.5$ cents. The calculated loss due to burnup (2180 MW hr) during this interval is 3.5 cents and the estimated loss due to Pu-241 decay contributes an additional ~ 1 cent. The measured and calculated changes at zero power are thus in acceptable agreement. The other change in the zero power critical condition on 8/21/70 is due to the core loading change discussed in Section 5.3.2. (See Tables 6, 7, and 12).

Although all the static test points in Table 12 were obtained at essentially the same average core coolant temperature and power level, the coolant inlet temperatures (flows) were not always the same (see Table 16). To determine the reactivity effect of a change in sodium inlet temperature while maintaining the power and average coolant temperature constant, two flow conditions were obtained (~ 1500 gpm on July 23 and 5000 gpm on July 24) at a power level of 5 MW, and again at 10 MW (~ 2500 gpm on Aug. 14 to ~ 5000 gpm on Aug. 15) as indicated in Table 16. As the corresponding reactivity data in Table 12 indicate, any reactivity effect associated with this change is less than the estimated standard deviation in reactivity ($\pm 1.5\%$ or 4% whichever is larger), and may be neglected.

The data in Table 12 also indicate that the integral power reactivity effect may be decreasing slightly with time. This is the behavior which would be expected were the fuel temperatures to decrease slightly as the new fuel pellets begin to crack and expand to fill the fuel-cladding gap as a result of operation at higher power levels. Such behavior would cause a slight change in the power-fuel temperature relation, and thus a small change in power-reactivity effect, even though the fuel temperature-reactivity relation remained unchanged.

5.3.6.2 Measured Doppler Power Reactivity Effects

The reactivity changes shown in Table 12 are composed of two effects: the loss in reactivity due to the axial expansion of the fuel and clad as it is heated, and a loss in reactivity due to Doppler broadening of the nuclear

Figure 5. Fuel Rod Temperature vs. Power Density

cross sections. Because SEFOR has segmented rods and dished fuel pellets, the former effect is small however and the dominant component is the Doppler effect. The "experimental" Doppler components of the power reactivity effects are listed in Table 17. These values were obtained from Table 12 by correcting for the calculated^(10,11) fuel and clad axial expansion reactivity effect of $-0.5\text{¢}/\text{MW}$.

5.3.6.3 Measured Temperature Reactivity Effects at Constant Power

The reactivity effect due to a change in average coolant temperature while holding the power constant was

measured at two different power levels during the power ascension phase of the experimental program. The measured results are tabulated in Table 18.

The at-power results in Table 18 show that the power calculated from a thermal heat balance and the power indicated by the neutron instrumentation (the wide range monitors - WRM's) tended to diverge as the coolant temperature was decreased below $\sim 700^\circ\text{F}$. Part of this difference is attributable to the increase in the number of neutron mean free paths between the core and the detectors because of the increase in sodium density as the temperature is lowered. Calculations indicate that the WRM readings

Table 11
IFA FUEL ROD TEMPERATURES

Fuel Temperature Rise Above Sodium Coolant Temperature of ~760°F

Date (1970)	Thermal Power (MW)	Central IFA				Peripheral IFA		
		Inner Fuel Rod		Outer Fuel Rod		Inner Fuel Rod		Outer Fuel Rod
		Upper T/C #1 (°F)	Lower T/C #2 (°F)	Upper T/C #3 (°F)	Lower T/C #4 (°F)	Upper T/C #17 (°F)	Lower T/C #18 (°F)	Lower T/C #20 (°F)
7/2	0.4	52	82	-	85	35	68	56
7/2	0.8	75	135	-	137	61	104	90
7/2	1.0	108	166	105	176	77	133	116
7/3	2.0	197	-	192	329	140	246	208
7/9	5.0	471	822	477	864	-	636	529
7/9	5.1	475	816	487	870	-	641	541
7/10	5.2	507	824	487	862	-	640	540
7/10	5.1	507	824	487	860	-	646	544
7/10	5.2	495	822	477	875	-	643	534
7/11	5.0	466	760	446	808	-	608	506
7/23	5.0	472	804	-	830	-	702	574
7/24	5.2	448	737	-	753	-	662	540
7/27*	5.2	453	750	-	769	-	710	564
7/27*	5.4	467	780	-	799	-	734	585
7/27**	5.4	463	775	-	792	-	752	598
7/28†	5.8	503	842	-	861	-	835	658
8/6	2.2	199	294	-	317	-	344	282
8/7	5.2	470	790	-	803	-	808	664
8/14	4.9	454	770	-	773	-	831	678
8/14	6.0	554	962	-	964	-	991	832
8/14	7.0	640	1123	-	1044	-	1119	966
8/14	8.0	724	1289	-	1364	-	1254	1090
8/14	9.1	824	1490	-	1578	-	1360	1236
8/14	10.1	902	1671	-	1748	-	1506	1343
8/15	10.1	884	1435	-	1774	-	1669	1317
8/17	9.9	872	1390	-	1772	-	1704	1327
8/18††	10.0	841	1329	-	1772	-	1646	1279
8/23	5.1	436	651	-	-	-	967	754
8/29	10.1	898	1365	-	1756	-	1753	1427

*Average core sodium temperature is ~600°F.

**Average core sodium temperature is ~500°F.

†Average core sodium temperature is ~400°F.

††Average core sodium temperature is ~560°F.

Table 11 (Continued)

Fuel Temperature Rise Above Sodium Coolant Temperature of ~760° F

Date (1970)	Thermal Power (MW)	Central IFA			Peripheral IFA	
		Inner Fuel Rod		Outer Fuel Rod	Inner Fuel Rod	Outer Fuel Rod
		Upper T/C #1 (° F)	Lower T/C #2 (° F)	Lower T/C #4 (° F)	Lower T/C #18 (° F)	Lower T/C #20 (° F)
8/29	11.2	973	1525	1834	1728	1520
8/29	12.5	1076	2021	2306	1691	1643
8/29	13.2	1145	2144	2487	1554	1736
8/29	14.0	1213	2292	2834	1499	1825
8/29	15.5	1383	2559	2761	1820	1957
8/29	14.8	1329	2351	2703	2095	1903
8/30	14.8	1343	1337	1957	2099	1920
8/30	15.1	1352	1331	1239	2262	1948
9/4	9.9	852	847	1491	1690	1426
9/5	10.1	828	840	1564	1720	1425
9/5	15.0	1305	1325	1934	2214	1969
9/5	15.0	1290	1315	1818	2170	1960
9/5	17.2	1612	1340	1805	1898	2177
9/6	17.2	1561	1307	1270	2251	2182

Figure 6. Fuel Temperature vs. Power (T/C #1)

Figure 7. Fuel Temperature vs. Power (T/C #2)

Figure 8. Fuel Temperature vs. Power (T/C #4)

Figure 9. Fuel Temperature vs. Power (T/C #18)

Figure 10. Fuel Temperature vs. Power (T/C #20)

Table 12
POWER REACTIVITY DATA

Date (1970)	Reflector #3		Reflector #8		Average Core Coolant Temperature (°F)	Thermal Power (MW)	Neutron Instrumentation (MW)	Reactivity Change* at 760° F Average
	Position (cm)	Worth (¢)	Position (cm)	Worth (¢)				Coolant Temperature (¢)
6/30-7/8	0	0	-	52.62†	760.0†	0	-	0
7/2	0	0	37.61	51.58	744.0	0.4	0.5	- 7.8
7/2	0	0	41.04	58.27	745.5	0.8	0.9	-13.6
7/2	0	0	48.16	72.22	763.2	1.0	1.1	-17.9
7/2	17.0	16.70	40.00	56.22	763.2	1.0	1.1	-18.6
7/3	16.76	16.37	47.00	69.97	759.1	2.0	2.0	-34.2
7/9-7/24	-	60.82†	47.00†	69.97†	760.0†	5.1†	5.1†	-78.2†
8/6-8/16	-	-	-	58.06†	760.0†	0	-	0
8/6	18.79	19.24	47.00	69.97	759.6	2.2	2.0	-31.4
8/7	43.70	64.02	47.00	69.97	760.1	5.2	5.1	-75.9
8/14	42.32	61.37	47.01	69.99	758.1	4.9	5.0	-74.3
8/14	50.83	77.70	47.01	69.99	763.7	6.0	6.1	-87.8
8/14	57.43	89.60	47.01	69.99	765.2	7.0	7.2	-98.9
8/14	62.11	97.72	47.01	69.99	760.6	8.0	8.1	-109.3
8/14	70.00	110.23	47.07	70.11	760.2	9.1	9.3	-122.2
8/14-8/17	70.00†	110.23†	-	78.92†	760.0†	10.0†	10.2†	-131.1†
8/23-8/30	0	0	-	21.96†	760.0†	0	-	0
8/23	40.00	56.91	30.82	38.84	759.7	5.1	5.2	-74.0
8/29	40.00	56.91	58.63	91.70	758.3	10.1	10.2	-127.5
8/29	40.00	56.91	64.00	100.78	759.6	11.2	11.1	-135.9
8/29	40.20	57.30	70.00	110.02	758.0	12.5	12.4	-146.3
8/29	43.35	63.35	70.00	110.02	757.4	13.2	13.2	-152.3
8/29	46.89	70.17	70.00	110.02	757.5	14.0	14.0	-159.4
8/29	53.48	82.59	70.00	110.02	759.4	15.5	15.3	-170.9
8/29	51.42	78.90	70.00	110.02	759.5	14.8	14.8	-167.2
8/30	50.19	76.50	70.00	110.02	756.2	14.8	14.9	-166.4
8/30	51.19	78.38	70.00	110.02	758.9	15.1	15.2	-170.0
9/4	57.50	89.72	40.00	56.22	762.2	9.9	10.0	-122.9
9/5	58.39	91.29	40.00	56.22	760.4	10.1	10.1	-125.4
9/5	50.57	77.21	70.00	110.02	756.1	15.0	15.1	-167.5
9/5	50.81	77.66	70.00	110.02	756.2	15.0	15.1	-184.3
9/5	61.00	95.85	70.00	110.02	759.1	17.2	17.4	-184.3
9/6	60.70	95.34	70.00	110.02	757.7	17.2	17.3	-184.5

*Reactivity-temperature corrections made using a reactivity coefficient of $\left[\frac{-0.36-0.19(1222)}{1220+65P} \right] \text{ ¢/° F at power level P (MW) ...}$

†Average values obtained from Tables 4, 7, 13, 14, and 15.

Table 13
ZERO POWER CRITICAL POINTS

Date	Reflector #3		Reflector #8		Reactor Inlet [†] temperature (°F)	Relative Reactivity* at 760°F (%)
	Position (cm)	Worth (%)	Position (cm)	Worth (%)		
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 were made with a previously measured⁽¹⁾ coefficient of $-0.55\%/^{\circ}\text{F}$. The estimated uncertainty in reactivity is $\pm 1.5\%$.

[†]The recorded panel temperature readings have been increased by one degree to account for an observed offset from the data acquisition system readings used in the other tables. The estimated uncertainty in temperature is $\pm 2.5^{\circ}\text{F}$.

should decrease $\sim 6.5\%$ at a constant thermal power due to an increase in sodium density of 6% as the temperature is lowered from 760°F to 400°F . The other contributor to the observed trend is the influence of reflector configuration on the neutron detection rate in the WRM's. This effect was noted⁽²⁾ during the initial core loading, where the WRM count rates were observed to decrease by $\sim 40\%$ more than the counts at the core center, when all 10 reflectors were completely lowered. If the total fission rate had remained constant, it is expected that the effect would have been less than this (the estimated value is ~ 28 rather than 40%), since the ratio of total power to in-core detector count rate is influenced by the flux shape changes resulting from changes in reflector configuration. For the change in reflector configuration between the 5 MW condition at a 760°F coolant temperature and that at a 400°F coolant temperature the WRM count rates would be expected to decrease by $\sim (1.5/10) 28 \sim 4\%$ due to reflector effects alone.*

5.3.6.4 Comparison of Measured and Calculated Doppler Reactivity Effects

The data in Table 17 demonstrate the dependence of the measured Doppler reactivity changes on the reactor power level. Since the Doppler reactivity effect is caused by changes in fuel temperature rather than the power per se, the relationship between total power and local fuel temperatures throughout the core must be known before the measured Doppler power-reactivity changes can be compared to the calculated values. To make this comparison, the fuel

thermal conductivity, fuel-to-clad gap coefficient, power distribution, etc., that are discussed in Section 5.3.5 were used in the WEDOP⁽¹²⁾ program to calculate local fuel temperatures throughout the core for various total reactor power levels. The local reactivity effect introduced by local fuel temperature changes was then calculated by the WEDOP⁽¹²⁾ program under the assumption that the local change in reactivity is proportional to the logarithm of the local temperature change. (A logarithmic total change corresponds to a differential change having a constant value for $T \frac{dk}{dT}$.) Since previous analyses⁽¹¹⁾ have shown

that a power squared weighting is a good approximation to the more accurate flux-adjoint weighting, the total core Doppler reactivity changes were then calculated using a power-squared weighting of the local reactivity effects.

The reactivity changes for a Doppler $T \frac{dk}{dT}$ of -0.0080 that were obtained from the WEDOP⁽¹²⁾ calculations are compared with the measured Doppler reactivity effects in Figure 11. It is seen that the agreement between measured and calculated effects as a function of reactor power is quite good.

The WEDOP⁽¹²⁾ program was also used to calculate the Doppler component of the change in reactivity that occurs when the coolant temperature is changed while the power level is held constant. These calculated results were combined with the previously⁽²⁾ calculated non Doppler coolant coefficient of $-0.36\%/^{\circ}\text{F}$ and are compared with the measured results in Table 19. As with the power-

*Note that reflector #10 was completely lowered and #3 was completely lowered from a mid position.

Table 14
5 MW POWER REACTIVITY DATA (7/9/70-7/23/70)

Date (1970)	Reflector #3		Reflector #8		Average Coolant Temperature (°F)	Thermal Power (MW)	Neutron Instrumentation (MW)	Reactivity Change* at a 760°F Average Coolant Temperature (¢)
	Position (cm)	Worth (¢)	Position (cm)	Worth (¢)				
6/3-7/8	0	0	-	52.62†	760.0†	0		0
7/9	42.20	61.14	47.00	69.97	760.8	5.0	5.1	-78.1
7/9	42.20	61.14	47.00	69.97	760.4	5.1	5.1	-78.3
7/10	42.94	62.56	47.00	69.97	760.8	5.2	5.1	-79.5
7/10	42.17	61.08	47.00	69.97	759.9	5.1	5.1	-78.5
7/10	42.22	61.18	47.00	69.97	759.8	5.2	5.1	-77.7
7/11	41.00	58.84	47.00	69.97	759.8	5.0	4.9	-76.1
7/23	41.75	60.27	47.00	69.97	758.7	5.0	5.1	-78.2
7/24	42.08	60.91	47.00	69.97	758.5	5.2	4.9	-79.0
						5.1	5.1	-78.2

*Temperature-reactivity corrections made using a reactivity coefficient of 0.51¢/°F. The estimated standard deviation in these reactivities is ±1.5¢.

†Average values obtained from Table 13.

Table 15
10 MW POWER REACTIVITY DATA (8/14-8/17)

Date (1970)	Reflector #3		Reflector #8		Average Coolant Temperature (°F)	Thermal Power (MW)	Neutron Instru- mentation (MW)	Reactivity Change* at a 760°F Average Coolant Temperature (¢)
	Position (cm)	Worth (¢)	Position (cm)	Worth (¢)				
8/6-8/16	-	-	-	58.06 [†]	760.0 [†]	0	-	-
8/14	70.00	110.23	52.68	80.89	760.8	10.1	10.3	-132.7
8/15	70.00	110.23	51.89	79.40	759.9	10.1	10.3	-131.6
8/17	70.00	110.23	50.47	76.68	759.7	9.9	10.1	-129.0
						10.0	10.2	-131.1

*Temperature-reactivity corrections made using a reactivity coefficient of 0.48¢/°F. The estimated standard deviation reactivity is ± 1.5¢.

[†]Average values obtained from Table 6.

reactivity measurements noted above, the agreement between calculations using a Doppler $T \frac{dk}{dT}$ of -0.0080 and experiment is good. In these cases, however, the Doppler effect is less than half the total measured change, and the comparison of results is more dependent upon the magnitude of the non Doppler contribution.

5.3.7 Oscillator Tests to 10 MW

5.3.7.1 Measured Transfer Functions

Oscillator experiments are performed at several power levels and coolant flows to demonstrate reactor stability, to provide data for comparison with the power coefficient determined from the static tests, and to provide for the determination of the frequency response of the fuel. Oscillator tests have been performed at zero power, and at nominal power levels of 2 MW, 5 MW, and 10 MW. The zero power oscillator tests have been previously analyzed and the good agreement between calculated and measured results is described in reference 1.

Up to eleven different frequencies of oscillation were used in each set of oscillator data at a given power level and coolant flow, and the complete range of frequencies was repeated 2 or 3 times. The neutron flux signals were recorded on two data channels (#27 and #28). These channels demonstrated similar behavior and in most cases only one data channel was used in the analysis. The oscillator experiments were performed during the first extended period of operation at a given power level, and they were then repeated during the second period of extended operation at the same power level. The former measurements are designated as Series A while the latter are designated as Series B. The nominal frequencies of oscillation used for the experiments are listed in Table 20,

while the amplitude of oscillation and the labels for the magnetic tapes on which the data is recorded, are listed in Tables 21A through 21J.

The power reactivity transfer functions measured at power levels of 2, 5 and 10 MW are tabulated in Tables 22 through 42 and the at-power data is compared with the zero power reactivity transfer functions in Figures 12 through 17.

In all of these measurements the lower end of the oscillator rod was at a mean position of 5.5 inches above the core mid plane. The neutron flux and oscillator rod position were measured as a function of time, and the rod position was converted to reactivity using the conversion factor of 2.74 cents per inch that was determined from analysis⁽¹⁾ of the zero power oscillator tests.

5.3.7.2 Theoretical Model

The oscillator test data is analyzed by comparing theoretical models with the measured frequency dependent reactivity feedback kernel, and the frequency response of the coolant temperature to power transfer functions. The "measured" reactivity feedback kernel, $h(iw)$, at frequency w is determined from the zero power reactivity transfer function G_0 , and the at-power reactivity transfer function, G , at frequency w through the following relation:

$$h(iw) = \frac{1}{P} \left[\frac{1}{G_0(iw)} - \frac{1}{G(iw)} \right]$$

In this relation P is the mean power level at which the oscillator test was performed, while the other notation is given above.

The theoretical model for the frequency dependent reactivity feedback is described in reference 10 and may be written as follows:

$$h(i\omega) = \alpha F_{av}(i\omega, \gamma, \nu) + [C_f F_s(i\omega, \gamma, \nu) + C_q] U(i\omega)$$

In this model α is the power reactivity coefficient at a constant average coolant temperature; γ and ν are parameters in the frequency response function F_{av} that describes the response to a power oscillation of the average fuel temperature, and in the function F_s that describes the response of the average fuel temperature to a coolant temperature oscillation. Mathematical models for F_{av} and F_s are given in reference 10. An approximation for F_{av} is $1/(1+i\omega\tau)$ where $\tau = \gamma + \nu^2/8$ is the fuel time constant. The constants C_f and C_q are the fuel temperature and coolant temperature reactivity coefficients, and U is the measured average coolant temperature response to a power oscillation.

As noted in reference 10, the reactivity energy coefficient η can be determined from the fitted parameters according to the following expression

$$\eta = \frac{\alpha - C_p}{(\gamma + \nu^2/8) A_f}$$

where $C_p = -0.5\text{¢}/\text{MW}$ is the calculated non-Doppler power reactivity coefficient and where $A_f = 0.943$ is the fraction of the energy absorbed in the fuel.

5.3.7.3 Measured Feedback and Coolant Temperature Oscillations

To fit the theoretical model to the data, measured values of h , and U are required at a number of frequencies ω . The experimental values of h are tabulated in Tables 22 through 42 and illustrated in Figures 18 through 23. These were determined from the measured at-power reactivity transfer functions and the calculated zero power reactivity function that was determined⁽¹⁾ from a fit to the zero power experimental data. The data indicates that coolant flow has little or no effect on reactivity feedback and it also indicates that the uncertainty in the phase angle of h becomes very very large at the higher frequencies. This large uncertainty in phase at higher frequencies is to be expected since the zero power and at-power reactivity transfer functions are nearly identical at the higher frequencies and h , which is related to the difference between the two, is nearly zero. Consequently, small variations in the measurements cause large variations in the phase of h .

The measured values of U were taken as zero at 2 MW since the coolant temperature oscillations at this power level were very small ($\lesssim 1^\circ\text{F}$). At higher power levels the coolant temperature oscillations are still small, but the magnitudes of the oscillations vary somewhat from one thermocouple to another. The influence of these variations and the effect of the coolant temperature oscillations on the fitted parameters is being investigated.

5.3.7.4 Preliminary Comparison of Measured and Theoretical Feedback Kernels

The power reactivity coefficients, α , the fuel time constants $(\gamma + \nu^2/8)$, and the energy coefficient η , that have been determined from analysis to date of the oscillator test data is shown in Table 43. The coolant flow appears to have no influence on the fitted parameters so only one value is listed for each power level. Calculated values of $C_q = 0.36\text{¢}/\text{MW}$ and $C_f = -0.15\text{¢}/^\circ\text{F}$ were used in the analysis. The effect of using different calculated coefficients, as well as the influence of the magnitude and phase of the coolant temperature power transfer functions on the fitted parameters is still being investigated. Preliminary indications are that the uncertainties in the tabulated values of Table 43 are of the order of $\pm 15\%$.

5.3.8 Comparison of Static and Oscillator Test Results with Calculations

A weighted least squares fit of the measured power-reactivity data in Table 12 to a second order polynomial was used to determine the power reactivity coefficients at various power levels up to 17.2 MW. The resultant coefficients are compared in Table 44 with the coefficients obtained from analysis of the oscillator tests. The table shows the general agreement between static and oscillator test results. In view of the good agreement between the calculated and measured static test results (discussed in Section 5.3.6), and the consistency between results obtained using different measurement techniques, the best estimate of the SEFOR Doppler $T \frac{dk}{dT}$ must be taken as -0.0080 .

The presently estimated standard deviation in the Doppler coefficient is ± 0.0015 . This confidence interval was taken from the square root of the sum of the squares of the following estimated standard deviations: 15% in fuel thermal conductivity — this causes $\sim 15\%$ uncertainty in fuel temperature at higher power levels, an additional 10% uncertainty in the fuel-temperature power relation due to the large uncertainty in the model for the fuel to clad gap coefficient, 5% uncertainty in reactivity measurements and a 5% uncertainty in absolute power level. It is expected that more detailed analysis may reduce the magnitude of the uncertainty band by $\sim 25\%$.

5.3.9 Prompt Neutron Lifetime Measurements

Analysis has been completed of additional tests on the He-3 chambers, which were performed to investigate two unexplained effects⁽²⁾ in the measurements of β/λ . These two effects are:

- (1) A difference of about 10% for β/λ from measurements at different neutron-flux levels was observed.
- (2) The contribution of uncorrelated noise in the one-detector experiment was lower than expected.

It was anticipated that these effects might be caused by saturation effects in the Helium-3 ionization chambers, combined with a decrease in the cut-off frequency.

In order to check this, the following chamber characteristics were measured at conditions close to those of the previous measurements:

- (1) The dependence of the direct current (d.c.) of the chamber on the applied voltage was determined.
- (2) The frequency response of the chamber was obtained by measuring the auto-correlation function in white neutron noise for the same d.c. levels and voltage which were used during the previous reactor noise measurements.

Both characteristics were measured with the detector located in the serpentine ring outside the core, and behind the Ni-reflectors, because white neutron noise was expected at this position.

The direct current versus voltage relation shows a plateau as expected (width ~ 300 volts), but the plateau is shifted to higher voltages with increasing reactor power (see Figure 24). This result indicates that the chambers must have been affected by saturation effects during the noise measurements No. 2 and 3 listed in Table 6-1 of reference 2. This saturation effect is expected to influence the absolute values of the correlation function but should not affect the experimental results for β/ℓ unless the frequency response of the chambers changes at the same time. However, differences were also observed in the power spectral densities $\phi(\omega)$ measured at different d.c. values ($\bar{i}$), which indicates that the saturation has changed the frequency response of the chamber. Figure 25 shows the power spectral densities measured at $\bar{i} = 0.1 \mu\text{A}$ and $\bar{i} = 4.0 \mu\text{A}$. The curves of additional power spectral density measurements with values for $\bar{i}$ between 0.1 and 4 μA lie between the two plots of Figure 25. But measurements with $\bar{i} < 0.1 \mu\text{A}$ indicate no further shift of $\phi(\omega)$ to larger values than those of curve A. Therefore, it can be concluded that for $\bar{i} \leq 0.1 \mu\text{A}$ the frequency response of the He-3 chambers has not been deteriorated by saturation effects. This explains the previously observed⁽²⁾ decrease of the measured β/ℓ with increasing reactor power level (i.e. higher d.c. currents) but proves also that the measurement No. 1 reported in reference 2 ($\ell/\beta = 0.205 \times 10^{-3}$ sec) is not erroneous because of saturation effects.

The relatively large decrease of all measured $\phi(\omega)$ curves in the region of the reactor cut-off frequency $\omega = 4800$ rad/sec may be due to either one of the following reasons

- (1) The frequency spectrum of the neutron noise outside the Ni reflector in the serpentine ring is not "white" for the chambers as expected.
- (2) The neutron noise was "white" but the measured power spectral density reflects the frequency response of the chamber. If this assumption turns out to be true, a big correction (of about 30%) would be necessary for the previously measured β/ℓ value.

The expectation of white neutron noise in the Ni reflector was based on the calculated detector efficiency of 3×10^{-5} (counts per fission in the reactor) and an estimated decrease in the ratio of correlated to uncorrelated noise by a factor of 100. If, however, the detector efficiency is considerably larger than 3×10^{-5} (by a factor of 10 or more) the observed shape of $\phi(\omega)$ could be explained by correlated neutrons from the core.

The assumption of a considerably higher detector efficiency could also explain the fact that the auto-correlation functions measured with the chambers in the core center showed no contribution of uncorrelated events. Though there is no obvious reason for a considerably higher detector efficiency than 3×10^{-5} , this assumption might be justified because it explains both facts: the large decrease of all the curves in Figure 25 in the region of the cut-off frequency of the reactor and the absence of an uncorrelated part in the auto-correlation functions measured in the core center. From this assumption it follows that the previously measured⁽²⁾ value $\ell/\beta = 0.205 \times 10^{-3}$ sec is correct. However, it has not been conclusively demonstrated that the chamber frequency response is not significantly affecting this value of ℓ/β . It must be noted that similar He-3 chambers were used in reactor noise measurements in the SEOFR ZPR-III critical assembly. In those measurements, the He-3 chambers (with 10 atmosphere of He-3 pressure) were shown to have no effect on the ℓ/β value obtained. The chambers used in SEFOR were identical to those used in ZPR -III, except that 6 atmospheres He-3 pressure was used in SEFOR, which is expected to improve the chamber frequency response.

Table 16
HEAT BALANCE DATA

Date (1970)	Main Primary System			Auxiliary Primary System			Thermal Power (MW)	Average Coolant Temperature (°F)
	Inlet* Temperature (°F)	Outlet* Temperature (°F)	Flow (gpm)	Inlet* Temperature (°F)	Outlet* Temperature (°F)	Flow (gpm)		
7/2	741.2	746.2	1987	747.2	748.4	185	0.4	744.0
7/2	740.5	750.1	1995	744.7	752.8	185	0.8	745.5
7/2	757.2	769.0	1998	758.7	771.7	185	1.0	763.2
7/3	747.8	769.8	2189	755.1	772.4	182	2.0	759.1
7/9	734.1	787.1	2326	743.3	786.3	180	5.0	760.8
7/9	733.7	786.4	2354	743.2	786.0	179	5.1	760.4
7/10	733.8	787.4	2347	740.7	786.0	181	5.2	760.8
7/10	732.8	786.6	2336	738.9	784.8	180	5.1	759.9
7/10	734.2	788.8	2342	741.4	786.0	173	5.2	761.7
7/11	744.0	775.5	3991	749.4	775.1	164	5.0	759.8
7/23	719.9	797.3	1512	724.9	794.3	182	5.0	758.7
7/24	745.3	771.6	5000	744.3	773.1	151	5.2	758.5
7/27	579.5	628.7	2480	578.9	625.0	153	5.2	604.0
7/27	579.1	628.3	2594	578.4	624.2	150	5.4	603.6
7/27	475.8	525.5	2516	479.4	519.2	152	5.4	500.6
7/28	376.8	426.8	2587	383.7	421.1	150	5.8	401.8
8/6	747.6	770.2	2412	751.0	769.6	151	2.2	759.0
8/7	734.2	786.3	2427	730.9	782.9	149	5.2	760.1
8/14	734.1	783.0	2449	727.3	777.4	171	4.9	758.1
8/14	734.0	794.1	2455	729.0	787.1	148	6.0	763.7
8/14	730.1	800.9	2458	726.2	793.2	145	7.0	765.2
8/14	720.8	801.0	2464	720.3	792.3	146	8.0	760.6
8/14	714.5	806.4	2443	715.3	796.7	146	9.1	760.2
8/14	710.0	812.2	2456	711.8	800.3	145	10.1	760.8
8/15	734.5	785.6	5000	732.7	779.5	147	10.1	759.9
8/17	699.7	820.7	1996	697.6	808.2	147	9.9	759.7
8/18	504.1	615.7	2050	503.5	602.8	147	10.0	559.4
8/23	735.3	784.0	2483	736.0	784.6	237	5.1	759.7
8/29	709.8	807.1	2505	711.7	802.2	210	10.1	758.3
8/29	723.0	796.9	3715	713.6	792.7	209	11.2	759.6
8/29	716.7	799.7	3703	714.5	794.6	212	12.5	758.0
8/29	713.7	801.3	3710	714.0	796.1	212	13.2	757.4
8/29	710.7	804.3	3703	714.6	798.8	211	14.0	757.5

Table 16 (Continued)

Date (1970)	Main Primary System			Auxiliary Primary System			Thermal Power (MW)	Average Coolant Temperature (° F)
	Inlet* Temperature (° F)	Outlet* Temperature (° F)	Flow (gpm)	Inlet* Temperature (° F)	Outlet* Temperature (° F)	Flow (gpm)		
8/29	708.4	810.7	3748	709.1	803.6	210	15.5	759.4
8/29	710.1	809.2	3703	710.1	802.5	209	14.8	759.5
8/30	695.4	817.2	2981	699.3	808.9	208	14.8	758.2
8/30	697.3	820.7	2986	702.0	812.3	217	15.1	758.9
9/4	715.5	809.1	2534	717.9	804.5	248	9.9	762.2
9/5	735.4	785.6	5000	733.6	784.1	230	10.1	760.4
9/5	695.3	817.1	3008	699.6	808.5	213	15.0	756.1
9/5	706.6	806.1	3737	707.8	799.3	208	15.0	756.2
9/5	713.9	804.5	4748	716.7	798.0	218	17.2	759.1
9/6	712.5	802.9	4760	717.0	797.3	220	17.2	757.7

*To account for observed differences between inlet and outlet temperature readings at zero power the tabulated temperatures have been adjusted from the actual recorded value as follows: between 7/2/70 and 7/11/70 the main inlet temperature was decreased 2°F and the auxiliary inlet temperature was decreased 1°F; electronic amplifiers were adjusted on 7/23; between 7/23 and 8/7 the auxiliary outlet temperature was increased 1°F while the main outlet temperature was increased 2°F between 7/27 and 8/7; between 8/14 and 9/6 the main outlet temperature was increased 0.5°F; the auxiliary outlet was decreased 2.1°F between 8/14 and 8/18 and was increased 1.3°F between 8/23 and 9/6.

Table 17
DOPPLER COMPONENT OF MEASURED
POWER-REACTIVITY EFFECTS

Date (1970)	Thermal Power (MW)	Doppler Effect* (ρ)
7/2	0.0	0.0
7/2	0.4	- 7.6
7/2	0.8	- 13.2
7/2	1.0	- 17.4
7/2	1.0	- 18.1
7/3	2.0	- 33.2
7/9-7/24	5.1	- 75.6
8/6-8/16	0.0	0.0
8/6	2.2	- 30.2
8/7	5.2	- 73.3
8/14	4.9	- 71.9
8/14	6.0	- 84.5
8/14	7.0	- 95.0
8/14	8.0	-105.3
8/14	9.1	-117.1
8/14-8/17	10.0	-126.1
8/23-8/30	0.0	0.0
8/23	5.1	- 71.5
8/29	10.1	-122.5
8/29	11.2	-130.3
8/29	12.5	-140.1
8/29	13.2	-146.1
8/29	14.0	-152.4
8/29	15.5	-163.2
8/29	14.8	-159.8
8/30	14.8	-158.9
8/30	15.1	-159.4
9/4	9.9	-118.0
9/5	10.1	-120.3
9/5	15.0	-159.6
9/5	15.0	-160.0
9/5	17.2	-175.7
9/6	17.2	-175.9

*Obtained from the last column of Table 16 by accounting for the calculated⁽¹¹⁾ fuel-fuel clad axial expansion reactivity effect of $-0.5\text{¢}/\text{MW}$. A preliminary estimate of the standard deviation in the Doppler reactivity values is $\pm 2.0\text{¢}$ or 5% whichever is larger. The preliminary estimate of the standard deviation in power is $\pm 0.3\text{ MW}$ or 5%, whichever is larger, for power levels $\geq 2\text{ MW}$.

Table 18

COOLANT TEMPERATURE REACTIVITY DATA AT A CONSTANT POWER

Date (1970)	Reflector #3		Reflector #8		Thermal Power (MW)	Neutron Instrumentation (MW)	Average Coolant Temperature (°F)	Reactivity** Change at a Constant Thermal Power (¢)
	Position (cm)	Worth (¢)	Position (cm)	Worth (¢)				
7/9-7/24	-	60.82*	47.00*	69.97*	5.1*	5.1*	760.0*	0*
7/27	35.53	46.76	0	0	5.2	4.9	604.0	85.3
7/27	36.80	49.12	0	0	5.4	5.1	603.6	85.6
7/27	40.00*	55.14	48.00†	70.36	5.4	4.9	500.6	142
7/28	0†	0	51.52†	71.03	5.8	5.1	401.8	202
8/14-8/17	70.00*	110.23*	-	78.92	10.0	10.2*	760.0*	0*
8/18	0	0	53.77	82.93	10.0	9.5	559.4	106

*Average Values obtained from Tables 14 and 15.

**Power reactivity corrections were made using a reactivity coefficient of $-13¢/\text{MW}$ at 5 MW.

† Reflector #10, which as a reactivity worth of 133¢, was completely lowered at the two lower temperatures, and was completely raised at the higher temperatures.

Table 19

MEASURED AND CALCULATED REACTIVITY EFFECTS AT CONSTANT POWER

Power (MW)	Average Core Coolant Temperature (°F)	Calculated* Non-Doppler Effect (ρ)	Calculated* Doppler Effect (ρ)	Total Calculated* Reactivity Change (ρ)	Total Measured** Reactivity Change (ρ)
0	760	0.0	0.0	0	0
0	650	39.6	24.0	64	63
0	550	75.6	48.0	124	123
0	350	147.6	104.0	252	254
5	760.0	0.0	0.0	0.0	0.0
5	604.0	56.2	27.3	83.5	85.3
5	603.6	56.3	27.6	83.9	85.6
5	500.6	93.4	47.6	141	142
5	401.8	129.0	69.1	198	202
10	760.0	0.0	0.0	0.0	0.0
10	559.4	72.2	33.8	106	106

*The calculated total non-Doppler effect is a constant $-0.364/^\circ\text{F}$. The Doppler contribution was calculated with the WEDOP⁽¹²⁾ program using a Doppler $T \frac{dk}{dT}$ of -0.0080 .

**The measured values at power were taken from Table 18 while the zero power values were taken from Reference 2.

Table 20

NOMINAL 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 21A
OSCILLATOR DATA

2 M/W, Series A, 40% Flow

Set #1 (Tape #62)			Set #2 (Tape #62)			Set #3 (Tape #62)		
Frequency	Rod Amplitude (inches)	File No.	Frequency	Rod Amplitude (inches)	File No.	Frequency	Rod Amplitude (inches)	File No.
F2	2.0	7	F2	2.0	21	S5	0.67	37
F3	2.0	8	F3	2.0	22	S4	0.67	38
F4	2.0	9	F4	2.0	23	S3	0.67	39
F4	1.5	10	F4	1.5	24	S2	0.67	40
F5	1.5	11	F5	1.5	25	S1	0.67	41
F6	1.5	12	F6	1.5	26	F6	0.67	42
F6	1.0	13	F6	1.0	27	F6	1.0	43
S1	1.0	14	S1	1.0	28			
S2	1.0	15	S2	1.0	29			
S3	1.0	16	S2	0.8	30			
S3	0.6	17	S3	0.8	31			
S4	0.6	18	S3	0.6	32			
S5	0.6	19	S4	0.6	33			
S6	0.6	20	S5	0.6	34			
			S5	0.6	35			
			S6	0.6	36			

Table 21B
OSCILLATOR DATA

2 MW, Series B, 40% Flow

Set 1 (Tape #93)			Set 2 (Tape #93)			Set 3 (Tape #93)		
Frequency	Rod Amplitude	File No.	Frequency	Rod Amplitude	File No.	Frequency	Rod Amplitude	File No.
F2	2.0 inches	12	F3	2.0 inches	27	F2	2.0 inches	40
F3	2.0	13	F4	2.0	28	F3	2.0	41
F4	2.0	14	F4	1.5	29	F4	2.0	42
F4	1.5	15	F5	1.5	30	F4	1.5	43
F5	1.5	16	F6	1.5	31	F5	1.5	44
F6	1.5	17	F6	1.0	32	F6	1.5	45
F6	1.0	18	S1	1.0	33	F6	1.0	46
S1	1.0	19	S2	1.0	34			
S2	1.0	20	S3	1.0	35			
S3	1.0	21	S3	0.6	36			
S3	0.6	22	S4	0.6	37			
S4	0.6	23	S5	0.6	38			
S5	0.6	24	S6	0.6	39			

Table 21C
OSCILLATOR DATA

5 MW, 30% Flow

Series A Set 1 (Tape #79)			Series A Set 2 (Tape #79)			Series B Set 1 & 2 (Tape #78)		
Frequency	Rod Amplitude	File No.	Frequency	Rod Amplitude	File No.	Frequency	Rod Amplitude	File No.
F2	1.2 inches	16	F2	1.2 inches	33	F2	1.2 inches	45
F3	1.2	15	F3	1.2	32	F3	1.2	46
F4	1.2	14	F4	1.2	31	F4	1.2	47
F5	1.2	13	F5	1.2	30	F5	1.2	48
F6	1.2	12	F6	1.2	29	F6	1.2	49
S1	1.2	17	S1	1.2	28	S1	1.2	50
S2	1.2	18	S2	1.2	27	S2	1.2	51
S3	1.2	19	S3	1.2	26	S3	1.2	52
S4	1.2	20	S3	1.2	25	S4	1.2	53
S5	1.2	21	S4	1.2	34	S5	1.2	54
S6	1.2	22	S5	1.2	24	S6	1.2	55
			S6	1.2	23	S5	1.2	56
						S3	1.2	57
						S1	1.2	58

Table 21D
OSCILLATOR DATA

5 MW, 50% Flow

Series A (Tape #72)			Series B Set #1 (Tape #78)			Series B Set #2 (Tape #78)		
Frequency	Rod Amplitude	File No.	Frequency	Rod Amplitude	File No.	Frequency	Rod Amplitude	File No.
S1	1.2 inches	13	F2	1.2 inches	3	F2	1.2 inches	14
F2	1.2	14	F3	1.2	4	F4	1.2	15
S2	1.2	15	F4	1.2	5	F6	1.2	16
F3	1.2	16	F5	1.2	6	S2	1.2	17
S4	1.2	20	F6	1.2	7	S4	1.2	18
F6	1.2	21	S1	1.2	8	S6	1.2	19
S6	1.2	25	S3	1.2	10			
S3	1.2	26	S4	1.2	11			
S4	1.2	27	S5	1.2	12			
			S6	1.2	13			

Table 21E
OSCILLATOR DATA

5 MW, 100% Flow, Series A

Set #1 (Tape #73)			Set #2 (Tape #73)		
Frequency	Rod Amplitude	File No.	Frequency	Rod Amplitude	File No.
F2	1.2 inches	23	F3	1.2 inches	32
F3	1.2	19	F4	1.2	33
F4	1.2	12	F5	1.2	34
F5	1.2	16	F6	1.2	35
F6	1.2	15	S1	1.2	25
S1	1.2	24	S2	1.2	26
S2	1.2	22	S3	1.2	27
S3	1.2	21	S4	1.2	28
S4	1.2	20	S5	1.2	29
S5	1.2	17	S6	1.2	37
S6	1.2	14			

Table 21F
OSCILLATOR DATA

5 MW, 100% Flow, Series B

Set 1 (Tape #78)			Set 2 (Tape #78)		
Frequency	Rod Amplitude	File No.	Frequency	Rod Amplitude	File No.
F2	1.2 inches	24	F2	1.2 inches	35
F3	1.2	25	F4	1.2	36
F4	1.2	26	F6	1.2	37
F5	1.2	27	S2	1.2	38
F6	1.2	28	S4	1.2	39
S1	1.2	29	S6	1.2	40
S2	1.2	30			
S4	1.2	32			
S5	1.2	33			
S6	1.2	34			

Table 21G
OSCILLATOR DATA

10 MW, 50% Flow, Series A

Set 1 (Tape #26)			Set 2 (Tape #26)		
Frequency	Rod Amplitude	File No.	Frequency	Rod Amplitude	File No.
F2	1.4 inches	12	F2	1.4 inches	23
F3	1.4	13	F4	1.4	25
F5	1.4	15	F5	1.4	26
F6	1.4	16	F6	1.4	27
S1	1.4	17	S1	1.4	28
S2	1.4	18	S2	1.4	29
S3	1.4	19	S3	1.4	30
S4	1.4	20	S4	1.4	31
S5	1.4	21	S5	1.4	32
S6	1.4	22	S6	1.4	33

Table 21H
OSCILLATOR DATA

10 MW, 50% Flow, Series B

Set 1 & 2
(Tape #65)

Frequency	Rod Amplitude	File No.
F2	1.4 inches	3
F3	1.4	4
F4	1.4	5
F5	1.4	6
F6	1.4	7
S1	1.4	8
S2	1.4	9
S3	1.4	10
S4	1.4	11
S5	1.4	12
S6	1.4	13
F2	1.4	14
F4	1.4	15
F6	1.4	16
S2	1.4	17
S4	1.4	18
S6	1.4	19

Table 21I
OSCILLATOR DATA
10 MW, 100% Flow, Series A

Set 1 (Tape #32)			Set 2 (Tape #32)		
<u>Frequency</u>	<u>Rod Amplitude</u>	<u>File No.</u>	<u>Frequency</u>	<u>Rod Amplitude</u>	<u>File No.</u>
F2	1.4 inches	12	F2	1.4 inches	23
F3	1.4	13	F3	1.4	24
F4	1.4	14	F4	1.4	25
F5	1.4	15	F5	1.4	26
F6	1.4	16	F6	1.4	27
S1	1.4	17	S1	1.4	28
S2	1.4	18	S2	1.4	29
S3	1.4	19	S3	1.4	30
S4	1.4	20	S4	1.4	31
S5	1.4	21	S5	1.4	32
S6	1.4	22	S6	1.4	33

Table 21J
OSCILLATOR DATA

10 MW, 100% Flow, Series B

Set 1 (Tape #65)			Set 2 (Tape #65)		
<u>Frequency</u>	<u>Rod Amplitude</u>	<u>File No.</u>	<u>Frequency</u>	<u>Rod Amplitude</u>	<u>File No.</u>
F2	1.4 inches	24	S5	1.4 inches	35
F3	1.4	25	S3	1.4	36
F4	1.4	26	S1	1.4	37
F5	1.4	27	F5	1.4	38
F6	1.4	28	F3	1.4	39
S2	1.4	30			
S3	1.4	31			
S4	1.4	32			
S5	1.4	33			

Table 22

**POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
2 MW 40% Flow (Set #1, Series A)
Power Channel 28**

<u>Frequency (rad/sec)</u>	<u>Power-Reactivity Transfer Function</u>		<u>Feedback Kernel</u>	
	<u>Amplitude (percent power) (cents reactivity)</u>	<u>Phase (degrees)</u>	<u>Amplitude (\$/MW)</u>	<u>Phase (degrees)</u>
1.555	1.118	-12.79	0.003632	187.0
0.9518	1.193	-16.02	0.006771	148.5
0.570	1.317	-20.09	0.01119	122.0
0.5875	1.300	-20.34	0.008592	143.5
0.337	1.517	-24.60	0.019211	119.3
0.1933	1.900	-28.20	0.28295	111.4
0.1603	2.058	-28.54	0.3264	109.3
0.1292	2.294	-27.87	0.04033	109.4
0.07742	2.845	-22.17	0.06307	116.5
0.0480	3.181	-13.04	0.09084	126.3
0.0481	3.157	-13.03	0.09129	127.0
0.0300	3.243	- 4.21	0.12049	134.0
0.01669	3.190	0.08	0.1428	150.0
0.01038	3.282	4.42	0.1495	153.3

Table 23
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
2 MW, 40% Flow (Set #2 and #3, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude ($\frac{\text{percent power}}{\text{cents reactivity}}$)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5641	1.085	-12.56	0.01707	186.63
0.9760	1.164	-15.99	0.01387	180.70
0.6073	1.266	-19.80	0.01495	165.51
0.6057	1.268	-19.79	0.01499	164.20
0.3479	1.481	-24.99	0.01609	137.20
0.2194	1.810	-27.81	0.02535	105.84
0.2178	1.786	-27.84	0.02497	115.82
0.1301	2.310	-27.53	0.04186	106.55
0.07870	2.843	-21.65	0.06460	115.39
0.07870	2.945	-22.67	0.06147	109.87
0.04937	3.245	-12.41	0.09077	122.78
0.04934	3.203	-12.50	0.09128	124.04
0.03096	3.226	- 3.49	0.1141	138.23
0.01787	3.067	0.90	0.1481	148.96
0.01796	3.310	0.30	0.1370	146.89
0.01134	3.196	3.45	0.1518	155.18
0.01755	3.235	0.06	0.1422	149.06
0.03030	3.263	- 4.39	0.1210	136.80
0.04880	3.228	-12.84	0.09114	125.60
0.07787	2.867	-22.24	0.06296	117.60
0.1292	2.298	-27.79	0.04029	114.06
0.2165	1.790	-28.10	0.02427	127.30
0.2170	1.797	-27.81	0.02541	123.73

Table 24
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
2 MW, 40% (Set #1, Series A)
Power Channel 27

Frequency rad/sec	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.555	1.106	-12.79	0.008826	169.56
0.9518	1.184	-15.92	0.009816	160.45
0.570	1.305	-20.23	0.01157	139.91
0.5875	1.308	-20.00	0.007449	136.80
0.337	1.512	-25.28	0.01884	117.22
0.1933	1.919	-28.50	0.02749	103.38
0.1603	2.038	-28.26	0.03368	115.52
0.1292	2.276	-27.82	0.04031	111.81
0.07742	2.837	-22.11	0.06330	117.13
0.0480	3.173	-13.03	0.09101	126.53
0.0481	3.181	-12.92	0.09110	126.15
0.0300	3.276	- 4.50	0.1189	136.07
0.01669	3.239	- 0.01	0.1406	149.77
0.01038	3.332	+ 4.85	0.1478	154.60

Table 25
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
2 MW, 40% (Set #2 and #3, Series A)
Power Channel 27

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power) (cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5641	1.079	-12.66	0.01980	189.77
0.9760	1.157	-15.66	0.01715	175.03
0.6073	1.277	-19.87	0.01208	158.43
0.6057	1.260	-19.98	0.01627	172.70
0.3479	1.472	-24.83	0.01777	141.96
0.2194	1.785	-27.74	0.02517	114.40
0.2178	1.765	-27.73	0.02560	123.38
0.1301	2.270	-27.60	0.04105	111.70
0.07870	2.869	-21.96	0.06358	114.01
0.07870	2.876	-21.66	0.06445	113.55
0.04937	3.218	-12.57	0.09082	123.65
0.04934	3.167	-12.82	0.09113	125.29
0.03096	3.166	- 3.84	0.1230	136.32
0.01787	3.008	1.53	0.1518	148.97
0.01796	3.260	0.10	0.1387	147.59
0.01134	3.162	3.53	0.1534	155.33
0.01755	3.222	0.50	0.1434	148.77
0.03030	3.239	- 4.78	0.1210	137.53
0.04880	3.197	-13.18	0.09020	124.95
0.07787	2.890	-22.41	0.06234	116.40
0.1292	2.320	-27.83	0.04033	111.14
0.2165	1.810	-28.00	0.02406	119.68
0.2170	1.815	-27.83	0.02517	117.29

Table 26
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
2 MW, 40% Flow (Set #1, Series B)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power) (cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5718	1.121	-12.97	0.002594	-126.00
0.9845	1.205	-15.92	0.003950	81.48
0.6150	1.316	-19.78	0.008442	85.62
0.6177	1.313	-20.07	0.006337	79.70
0.3585	1.526	-25.04	0.01453	85.52
0.2285	1.790	-28.05	0.02286	100.24
0.2301	1.798	-28.07	0.02316	95.35
0.1299	2.311	-28.24	0.03928	105.86
0.07791	2.904	-22.94	0.06057	112.88
0.04847	3.322	-13.95	0.08600	122.37
0.04852	3.316	-14.38	0.08498	122.73
0.02996	3.395	- 5.89	0.1127	135.28
0.01692	3.365	- 0.72	0.1345	148.84

Table 27
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
2 MW, 40% Flow (Set #2 and #3, Series B)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (<u>percent power</u>) (<u>cents reactivity</u>)	Phase (degrees)	Amplitude (<u>\$/MW</u>)	Phase (degrees)
0.9640	1.208	-16.38	0.002122	64.53
0.5925	1.328	-20.13	0.008951	82.86
0.5943	1.323	-20.13	0.008254	90.28
0.3335	1.559	-25.45	0.01629	91.90
0.2022	1.901	-28.51	0.02660	97.07
0.2028	1.894	-28.41	0.02666	98.73
0.1305	2.314	-28.50	0.03840	104.52
0.07900	2.906	-23.12	0.06004	111.62
0.04955	3.319	-14.47	0.08441	121.62
0.04964	3.302	-14.69	0.08409	122.16
0.03116	3.400	- 6.41	0.1106	134.18
0.01815	3.379	- 1.41	0.1318	147.47
0.01155	3.358	3.23	0.1444	153.94
1.5756	1.123	-12.73	0.001296	189.51
0.9876	1.207	-15.86	0.004678	71.96
0.6167	1.311	-19.92	0.006948	90.43
0.6181	1.300	-20.02	0.005774	115.83
0.3590	1.518	-24.92	0.01435	92.20
0.2277	1.799	-28.06	0.02332	98.03
0.2288	1.789	-27.84	0.02378	100.92

Table 28
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
5 MW, 30% Flow (Set #1, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power) (cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.579	1.151	-12.44	0.003970	25.04
0.9908	1.229	-15.16	0.005372	55.92
0.6201	1.343	-18.51	0.007800	71.91
0.3592	1.551	-22.12	0.01269	90.84
0.2280	1.859	-23.31	0.01906	95.85
0.1299	2.403	-16.96	0.03269	103.54
0.07819	2.773	- 0.81	0.05144	110.47
0.04881	2.511	15.00	0.07401	119.74
0.03058	1.943	21.04	0.1019	132.60
0.01764	1.604	18.27	0.1247	146.72
0.01116	1.530	14.81	0.1312	155.28

Table 29
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
5 MW, 30% Flow (Set #2, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5751	1.147	-12.21	0.003683	38.52
0.9871	1.230	-15.10	0.005580	57.73
0.6138	1.345	-18.38	0.008247	74.89
0.3525	1.570	-22.34	0.01293	87.74
0.2192	1.894	-22.90	0.02035	96.57
0.1307	2.421	-17.05	0.03262	102.03
0.07890	2.774	- 1.09	0.05112	110.24
0.04925	2.522	15.09	0.07382	119.27
0.04939	2.469	16.18	0.07609	119.31
0.03063	1.973	20.67	0.1002	132.50
0.01758	1.603	14.69	0.1228	150.26
0.01116	1.536	14.85	0.1307	155.20

Table 30
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
5 MW 30%, Flow (Set #1 and #2, Series B)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power) (cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
0.9613	1.140	-12.47	0.002110	51.80
0.9626	1.218	-15.32	0.004196	79.67
0.5941	1.340	-18.46	0.008083	87.87
0.3378	1.572	-22.37	0.01325	96.03
0.2098	1.902	-22.93	0.02045	101.20
0.1283	2.404	-16.67	0.03307	104.47
0.07729	2.734	-0.99	0.05158	112.09
0.04826	2.452	15.34	0.07578	120.78
0.03028	1.950	20.50	0.1012	133.15
0.01755	1.609	15.38	0.1227	149.55
0.01089	1.604	16.30	0.1259	153.56
0.01739	1.612	16.89	0.1234	148.15
0.1291	2.388	-16.94	0.03267	104.95
0.9739	1.219	-15.05	0.004789	77.26

Table 31
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
5 MW, 50% Flow (Set #1, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude ($\frac{\text{percent power}}{\text{cents reactivity}}$)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5570	1.144	-13.13	0.002795	-6.81
0.9765	1.211	-15.80	0.002367	73.07
0.2105	1.893	-23.57	0.01920	101.41
0.1288	2.324	-16.96	0.03265	109.13
0.07795	2.627	- 1.95	0.05155	115.36
0.04858	2.375	12.65	0.07498	124.16
0.03032	1.951	17.53	0.09877	135.85
0.02996	1.939	16.89	0.09895	136.80
0.01082	1.562	17.52	0.1297	152.70

Table 32
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
5 MW, 50% Flow (Set #2, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude ($\frac{\text{percent power}}{\text{cents reactivity}}$)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5562	1.134	-13.14	0.001385	-31.31
0.9691	1.208	-15.94	0.001914	83.51
0.5996	1.319	-19.49	0.004645	98.44
0.3526	1.538	-22.90	0.01085	97.33
0.2179	1.851	-23.52	0.01867	103.69
0.1291	2.316	-17.12	0.03241	109.53
0.07770	2.630	- 2.19	0.05123	115.52
0.04866	2.395	12.52	0.07432	123.85
0.02930	1.924	17.70	0.1005	136.61
0.01656	1.672	16.67	0.1193	148.39
0.009925	1.542	17.69	0.1316	153.37

Table 33

POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT

5 MW, 50% Flow (Set #1 and #2, Series B)

Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power) (cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5198	1.133	-12.52	0.001517	92.12
0.9500	1.214	-15.33	0.004144	92.43
0.5879	1.340	-18.82	0.007331	88.53
0.3343	1.572	-22.42	0.01325	98.15
0.2065	1.902	-22.55	0.02121	103.58
0.1269	2.370	-16.69	0.03301	107.30
0.04776	2.436	12.04	0.07296	123.93
0.02969	2.024	17.42	0.09559	135.31
0.01698	1.760	16.44	0.1132	147.53
0.01052	1.654	18.45	0.1231	151.49
1.5376	1.133	-12.36	0.001826	82.63
0.5947	1.332	-18.57	0.007535	92.52
0.2078	1.910	-22.69	0.02101	101.60
0.07682	2.677	- 2.07	0.05087	114.49
0.02978	2.005	17.21	0.09619	135.70
0.01054	1.667	18.08	0.1220	151.75

Table 34
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
5 MW, 100% Flow (Set #1, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (<u>percent power</u>) (<u>cents reactivity</u>)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5510	1.110	-12.19	0.003541	156.55
0.9683	1.190	-15.27	0.004228	132.62
0.6050	1.300	-18.53	0.006967	117.51
0.3472	1.518	-22.13	0.01263	110.39
0.2182	1.816	-23.13	0.01917	109.92
0.1285	2.286	-18.16	0.03091	112.09
0.07754	2.643	- 5.02	0.04762	116.87
0.04852	2.505	8.78	0.06797	124.87
0.03026	2.126	16.14	0.09018	134.76
0.01750	1.805	17.57	0.1110	145.65
0.01108	1.664	18.12	0.1222	151.27

Table 35
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
5 MW, 100% Flow (Set #2, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (<u>percent power</u>) (<u>cents reactivity</u>)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
0.9768	1.187	-15.98	0.002521	160.73
0.6086	1.300	-19.33	0.004633	119.15
0.3509	1.525	-22.84	0.01081	104.77
0.2204	1.821	-23.60	0.01820	107.37
0.1286	2.290	-18.22	0.03081	111.76
0.07785	2.633	- 5.31	0.04738	117.21
0.04887	2.513	8.56	0.06752	124.70
0.03067	2.138	16.22	0.08967	134.28
0.01773	1.806	17.24	0.1106	145.78
0.01055	1.647	18.01	0.1235	151.93

Table 36

POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT

5 MW, 100% Flow (Set #1 and #2, Series B)

Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power) (cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5408	1.141	-12.42	0.002401	50.21
0.9664	1.218	-15.08	0.006727	81.65
0.6007	1.332	-18.45	0.007647	90.40
0.3449	1.558	-22.05	0.01339	97.27
0.2171	1.863	-22.73	0.02029	102.86
0.1285	2.326	-17.21	0.03228	109.24
0.07780	2.666	- 4.16	0.04844	115.53
0.03073	2.130	16.95	0.09056	133.70
0.01789	1.814	17.59	0.1103	145.26
0.01144	1.675	18.76	0.1215	150.29
1.5635	1.143	-12.34	0.002807	42.09
0.6145	1.329	-18.38	0.007368	86.64
0.2258	1.830	-22.80	0.01962	102.80
0.07825	2.669	- 4.44	0.04805	115.35
0.03090	2.147	16.85	0.08978	133.46
0.01161	1.655	19.18	0.1231	149.87

Table 37
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
10 MW, 50% Flow (Set #1, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power) (cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.555	1.127	-11.66	0.001822	103.30
0.976	1.206	-14.19	0.003350	100.20
0.355	1.551	-18.69	0.01013	99.72
0.227	1.864	-16.87	0.01548	101.41
0.128	2.294	- 3.76	0.02693	106.73
0.077	2.205	15.71	0.04045	114.47
0.0484	1.724	26.10	0.05715	124.16
0.0303	1.336	26.60	0.07493	135.56
0.0175	1.148	22.96	0.08780	146.40
0.0110	1.067	23.87	0.09543	149.42

Table 38
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
10 MW, 50% Flow (Set #2, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power) (cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5707	1.132	-11.53	0.001951	89.72
0.6146	1.330	-16.72	0.005786	93.55
0.3559	1.561	-18.82	0.009992	96.80
0.2256	1.872	-16.55	0.01582	101.32
0.1296	2.301	- 4.08	0.02600	106.59
0.0785	2.217	15.18	0.03997	114.24
0.0493	1.749	26.11	0.05639	123.47
0.0311	1.368	26.43	0.07309	135.01
0.0182	1.168	23.18	0.08632	145.69
0.0118	1.080	22.73	0.09407	150.07

Table 39
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
10 MW, 50% Flow (Set #1 and #2, Series B)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power) (cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5221	1.119	-11.60	0.002325	124.58
0.9491	1.201	-14.50	0.003270	113.01
0.5855	1.324	-17.13	0.005737	105.52
0.3333	1.571	-18.98	0.01041	103.21
0.2064	1.918	-16.11	0.01649	104.16
0.1271	2.266	-4.52	0.02583	109.12
0.0767	2.205	14.38	0.03966	115.67
0.0480	1.767	24.81	0.05533	124.71
0.0300	1.375	25.60	0.07256	136.10
0.0173	1.188	22.43	0.08483	146.71
0.0104	1.079	23.11	0.09426	150.18
1.5438	1.125	-11.53	0.002122	109.04
0.5974	1.324	-16.96	0.005714	101.87
0.2095	1.910	-16.18	0.01639	103.68
0.0775	2.222	13.79	0.03907	115.46
0.0300	1.388	25.73	0.07196	135.82
0.0106	1.089	23.84	0.09356	149.57

Table 40
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
10 MW, 100% Flow (Set #1, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5223	1.119	-11.65	0.002271	126.37
0.9511	1.203	-14.36	0.003420	110.11
0.5891	1.331	-17.04	0.005816	100.10
0.3354	1.569	-21.55	0.007520	100.20
0.2099	1.883	-16.01	0.01652	106.22
0.1273	2.235	- 5.37	0.02535	110.76
0.0767	2.224	11.81	0.03785	117.12
0.0479	1.826	23.78	0.05325	124.65
0.0299	1.444	26.86	0.06980	134.06
0.0173	1.186	25.66	0.08587	143.58
0.0107	1.039	25.77	0.09837	147.91

Table 41
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
10 MW, 100% Flow (Set #2, Series A)
Power Channel 28

Frequency (rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel	
	Amplitude (percent power cents reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
1.5525	1.127	-11.73	0.001730	104.77
0.9690	1.207	-14.12	0.003537	100.84
0.6003	1.328	-16.59	0.006165	99.53
0.3426	1.560	-18.65	0.01050	102.32
0.2125	1.889	-16.31	0.01622	104.56
0.1291	2.237	- 5.72	0.02507	110.15
0.0780	2.243	11.47	0.03738	116.47
0.0487	1.853	23.33	0.05229	124.34
0.0305	1.456	26.52	0.06904	134.00
0.0177	1.189	25.20	0.08545	143.78
0.0113	1.064	25.76	0.09603	147.47

Table 42
POWER-REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
10 MW, 100% Flow (Set #1 and #2, Series B)
Power Channel 28

<u>Frequency</u> <u>(rad/sec)</u>	<u>Power-Reactivity Transfer Function</u>		<u>Feedback Kernel</u>	
	<u>Amplitude</u> <u>(percent power)</u> <u>(cents reactivity)</u>	<u>Phase</u> <u>(degrees)</u>	<u>Amplitude</u> <u>(\$/MW)</u>	<u>Phase</u> <u>(degrees)</u>
1.5441	1.131	-11.73	0.001797	95.45
0.9562	1.218	-14.22	0.003621	91.89
0.5856	1.341	-17.02	0.005989	96.16
0.3281	1.592	-18.72	0.01091	101.04
0.1979	1.929	-15.60	0.01708	106.57
0.0771	2.262	10.87	0.03680	116.79
0.0489	1.873	23.18	0.05178	124.39
0.0296	1.466	26.53	0.06869	134.17
0.0167	1.200	24.76	0.08473	144.63
0.0166	1.185	24.43	0.08573	145.10
0.0480	1.873	23.07	0.05172	124.46
0.1287	2.237	- 6.22	0.02469	110.50
0.3299	1.595	-18.85	0.01074	99.50
0.9607	1.208	-14.09	0.003677	102.13

Figure 11. Doppler Power Reactivity Measurements

Figure 12. Amplitude of Power Reactivity Transfer Function at 2 MW (Set 1, Series A)

Figure 13. Phase of Power Reactivity Transfer Function at 2 MW (Set 1, Series A)

Figure 14. Amplitude of Power Reactivity Transfer Function at 5 MW (Set 1, Series A)

Figure 15. Phase of Power Reactivity Transfer Function at 5 MW (Set 1, Series A)

Figure 16. Amplitude of Power Reactivity Transfer Function at 10 MW (Set 1, Series A)

Figure 17. Phase of Power Reactivity Transfer Function at 10 MW (Set 1, Series A)

Figure 18. Amplitude for Feedback Kernel at 2 MW (Set 1, Series A)

Figure 19. Phase of Feedback Kernel at 2 MW (Set 1, Series A)

Figure 20: Amplitude of Feedback Kernel at 5 MW (Set 1, Series A)

Figure 21. Phase of Feedback Kernel at 5 MW (Set 1, Series A)

Figure 22. Amplitude for Feedback Kernel at 10 MW (Set 1, Series A)

Figure 23. Phase of Feedback Kernel at 10 MW (Set 1, Series A)

Table 43
PRELIMINARY OSCILLATOR TEST DATA

Power (MW)	α Power-Reactivity Coefficient ($\\$/\text{MW}$)	τ Fuel Time Constant (Sec)	η Doppler Energy Coefficient ($\\$/\text{MW-Sec}$)
2	-16.4	29	-0.57
5	-11.3	28	-0.41
10	- 8.7	28	-0.31

$$\eta = \frac{1}{.943} \left(\frac{\alpha + 0.5}{\tau} \right)$$

The preliminary estimated uncertainties in the parameters are $\pm 15\%$.

Table 44
COMPOSITE POWER REACTIVITY COEFFICIENTS

Data Set	Coefficient At Zero Power ($\\$/\text{MW}$)	Coefficient At 2 MW ($\\$/\text{MW}$)	Coefficient At 5 MW ($\\$/\text{MW}$)	Coefficient At 10 MW ($\\$/\text{MW}$)
Static Tests 6/30-7/24	-18.4 $\pm$ 1.5	-16.0 $\pm$ 1.5	-12.4 $\pm$ 1.5	-
Static Tests 8/6-8/17	-16.5 $\pm$ 2.0	-15.2 $\pm$ 1.5	-13.2 $\pm$ 1.5	-9.8 $\pm$ 1.5
Static Tests 8/23-8/29	-16.4	-14.9	-12.7	-8.9
Oscillator Tests	-	-16.4	-11.3	-8.7

Preliminary analysis indicates that the uncertainties in the coefficient determined from the oscillator tests are of the order of $\pm 15\%$.

CHAMBER CURRENT (dc)

Figure 24. Direct Current-Voltage Characteristics of the He Chamber

Figure 25. Power Spectral Density Measured in the Serpentine Ring at Different D.C. Current Levels

6. TASK 11—TRANSIENT TESTS

6.1 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 sub-prompt-critical transient reactivity changes.

6.2 PAST PERFORMANCE

An irradiated-fuel shipping cask, rented with the authorization of the SAN office of the AEC, was corrected for deficiencies and shipped to the SEFOR site. License applications to use the cask for shipment of irradiated SEFOR fuel rods were made to the AEC and the U.S. Department of Transportation (DOT). Equipment required for the fuel transfer operation was fabricated and sent to the SEFOR site. Shielding around the vertical transfer valve

in the SEFOR refueling cell was designed and fabrication has begun.

6.3 CURRENT RESULTS

The AEC requested additional information concerning the structural analysis of the fuel-rod shipping-cask license application. The information was supplied and the AEC issued a license for the cask. The DOT has issued a special permit allowing the loaded cask to be transported.

Calculations were made for the radiation levels in the refueling cell and around the cask during the time fuel is being loaded into the cask. The calculations have resulted in additional shielding being added to the transfer valve shielding design. The additional shielding will reduce the effects of radiation scattering off the wall behind. Fabrication and installation of the shielding, including modifications, has been completed.

7. SEFOR SAFETY AUDIT

Test results up to 10 MWt showed that the Doppler coefficient, net non-fuel coefficient, and cover gas monitor sensitivity meet the requirements of the Technical Specifications.

The temperature record following a scram from 15 MWt showed a steady decrease in reactor outlet temperature, indicating that the programmed reduction in coolant flow rate is well matched with the thermal diffusivity of the fuel.

The natural circulation tests were observed. Initial results show that the tests were satisfactory.

Temperature and cooling flow data from the reflector temperature coefficient of reactivity tests was reviewed. The data show that the reflector temperatures are below the maximum predicted values. Therefore, the heat generation rates in the reflector region are correspondingly lower than the maximum predicted values. This provides additional safety margin for the reflector guide structure.

The shift in power coefficient which occurred during initial operation to 10 MWt is within the change predicted analytically through the mechanism of fuel pellet cracking and a resultant decrease in the fuel-clad gap. Recent operation including power cycles of zero to 15 MWt to zero to 17.5 MWt to zero indicate no measurable change in the power coefficient, indicating that the changes in heat transfer coefficient between the clad and pellet due to pellet cracking and shifting radially are no longer occurring.

The results of the surveillance examinations on guinea pig rods A16 and A17 after operation at 17.5 MWt (equivalent to a standard hot rod at 20 MWt) were reviewed. The gamma scans indicate no significant change in the axial length of the fuel column. The Profilometer traces indicate no net diametral change or ridging at pellet interfaces, or increase in ovality. The spiral trace taken on rods A16 and A17 revealed that the rods are bowed a maximum of 0.018 and 0.045 inches, respectively. The length of the fuel rods did not change.

The sixth meeting of the SEFOR Safety Review Committee was held at the SEFOR site in Fayetteville on October 21 and 22 to review SEFOR operations. No unusual occurrences were noted.

Diagnostic tests were run at coolant inlet temperatures of 400°F to 745°F and power levels of 5 to 16 MWt to determine the wide-range monitors (WRM) variation with coolant temperature and reflector position. Changes in coolant temperature and reflector position both affect WRM readings at constant power.

A meeting has been scheduled with the DRL of the AEC on November 3 in Washington, to answer questions about the Technical Specifications changes required for 20 MWt operation proposed in PC-4 concerning anomaly limits, operation with failed fuel, and limits on cover gas activity.

8. SEFOR SUPPORT

The plant engineering effort is a supporting function to the SEFOR facility. During August and early September, the reactor power was raised in increments, oscillator tests were accomplished and operation and performance checks of the coolant systems completed. On September 5, 1970, power was raised to 17.5 MWt and satisfactory plant operation was achieved.

During power operation, heat balances were obtained for the major transfer equipment. A comparison of heat transfer coefficients was made for the intermediate heat exchangers with the following results:

	<u>BTU</u> <u>Hr-Ft²-°F</u> <u>Predicted</u>	<u>BTU</u> <u>Hr-Ft²-°F</u> <u>Measured</u>
Main IHX		
4500 gpm	1180	1340
9.8 MW		
Aux. IHX		
245 gpm	1140	525
1.0 MW		

Performance on the main IHX has been outstanding. The heat transfer has exceeded the design calculated prediction for all phases of operation. The auxiliary heat exchanger unit has not performed up to expectations, however, The measured performance level is considerably below the design rating, although no significant changes have been noted during power operation. Investigation of the reason for the apparent poor heat transfer coefficient is in progress. The capacity of the auxiliary unit has significant margin built in, and the unit has achieved a thermal output sufficient to allow operation without difficulty for the current experimental program.

Predicted performance of the heat exchangers was based upon the following correlations for film heat transfer:

Tube Side

$$N_u = 0.625 (P_r R_e)^{0.4} \quad (\text{Lubarsky-Kaufman}^*)$$

P_r = Prandtl Number

R_e = Reynolds Number

N_u = Nusselt Number

Shell Side

$$N_u = 0.212 (de Re Pr)^{0.6}^{**}$$

de = Equivalent Diameter

*NACA TN 1270

**Liquid Metals Handbook—Sodium (NaK) Supplement, 1955 Ed.

The design and fabrication of the replacement argon purification system was completed. All components were fabricated, received the code inspection and were delivered to the site. Figure 26 illustrates one of the bubbler tanks near the completion of fabrication. The design of this system is covered in the SEFOR Quarterly Progress Report No. 24, GEAP 10010-24. The design conditions for the large tanks are:

	<u>Bubbler Tank</u>	<u>Back Flow Tank</u>
Capacity		
Weight (NaK)	1500 lb	-
Flow Rate (ACFM)	100	100
Design		
Temperature (°F)	250	250
Pressure (psig)	15	15
Operating		
Temperature (°F)	125	125
Pressure (psig)	6	6

The bubbler tanks utilize a horizontal tank containing NaK to the mid-plane for removal of trace oxygen impurities from the refueling cell gas system. Flow at 100 ACFM enters through a 4-inch downcomer at one end of the tank, and is distributed through a perforated plate. Mesh located above the plate breaks up bubbles and provides a longer transit time through the NaK. The mesh also provides nucleation surfaces for the formation and retention of oxides.

Gas leaving the NaK surface at the tank mid-plane passes through two horizontal and two vertical mesh sections before leaving the tank. The mesh sections provide knock-out of any gross carryover. Gas then enters the separator located downstream of the bubblers where small particulate entrained in the gas stream are removed. The separator is a centrifugal design which provides for separation of either liquid droplets or solid particles, down to the range of 10 microns or less. Gas enters the side of the centrifugal nozzle and heavier particles are thrown to the outer periphery where they fall into a collection chamber located at the bottom of the separator. Gas leaves the nozzle upward through the center portion. The design conditions for the separator are as follows:

	<u>Centrifugal Separator</u>
Design	
Pressure (psig)	15
Temperature (°F)	250
Flow Rate (ACFM)	30-120
Efficiency	99% of all particulate 10 microns and larger 90% of all particulate between 5-9 microns

In order to ensure that good flow distribution existed in the bubblers as the gas passed through the mesh section, a water flow test was performed. Various combinations of perforated plate, inlet distribution sparger, and mesh were tested to determine the combination which gave the best distribution with a minimum potential for plugging. Results

of the testing are illustrated in Figure 27. As a result of the testing it was concluded that the combination of the perforated plate, mesh, and downcomer extending to the mid-point of the tank gave the best distribution and minimized the possibilities for plugging.

Difficulties with the oscillator were experienced when the angular position transmitter shaft was found to be broken. It appeared that the failure occurred due to fatigue. A flexible bellows assembly coupling was designed for the position transmitter. The transmitter was replaced using a flexible coupling.

The oscillator positioner failed to function properly, and it was discovered that the teeth on the gear reduction unit were badly worn. The gear reduction system was redesigned to provide a unit with a larger torque rating. Failure of the original unit may have been due to lack of lubrication.

Boral sheet was ordered for installation in the primary equipment zone. This will be utilized outside of the pipe tunnel shield as a means of reducing the neutron flux in the primary equipment area, and thereby reducing the activity of the secondary sodium.

Fabrication of the backup reactor vessel through head tip probe (Type I) was completed to the point of final inspection and code testing. Final assembly of wiring and electrical connections remains to be done.

Modifications were designed and completed on the existing replacement J probes to improve performance. The modifications consist of providing additional clearance between the active portions of the probe and the surrounding structure. It is expected the additional clearance will improve the reliability of probe and reduce the potential for sodium hangup.

A design study was initiated covering retrieval methods for the refueling cell crane system. Methods, equipment and procedures designed to cope with crane failure modes are under study.

Procurement was initiated for an axial fan type blower which will replace the existing centrifugal fan used for cooling the argon refueling cell. The change is being made to the axial fan unit in order to avoid difficulties that have occurred with the maintainability of the existing blower shaft seal.

TANK DURING FABRICATION

COMPLETED TANK

Figure 26. Bubbler Tank

100 CFM FROM THE SPARGER (NO PERFORATED PLATE). RESULTED IN CHANNELING ALONG TANK CENTERLINE.

100 CFM FROM 4" PIPE AT TANK CENTER, BUBBLING THROUGH PERFORATED PLATE AND STEEL WOOL. THIS IS THE RECOMMENDED DESIGN.

100 CFM FROM SPARGER BUBBLING THROUGH PERFORATED PLATE AND STEEL WOOL. THIS WAS THE BEST FLOW PATTERN, ALTHOUGH NOT SIGNIFICANTLY BETTER THAN ABOVE.

Figure 27. Meter Flow Test

9. SEFOR OPERATIONS

9.1 EXPERIMENTAL PROGRAM

Group II tests through reactor power of 17.5 megawatts were completed on September 6, 1970. The power level of 17.5 megawatts, the highest power achieved to date, was first established on September 5, 1970.

Following 17.5 megawatt fuel surveillance, AEC approval for continuation of Group II tests to 19 and 20 MW had not been received, and Group II tests above 17.5 MW were postponed. Group III static and oscillator tests below 15 megawatts were begun, and all Group III tests, below 16 MW, except balanced oscillator tests, were performed. Type 1 manual balance oscillator tests were also done at 10 MW, 760°F and 2000 pgm. In the type 1 test, the reactivity and the primary and secondary flow rates are oscillated in a sinusoidal manner. The relative phase angles and amplitudes of the reactivity and flow are adjusted such that the inlet and outlet coolant temperatures remain constant. Tests completed during the quarter are tabulated in Tables 45 through 47.

Initial on-site analysis of the Fourier Coefficient indicates that coolant temperature balances were obtained for frequencies higher than 0.0045 Hz to less than 0.7°F. At frequencies of 0.0027 and 0.00165, 1°F was the lowest coolant temperature balance that could be obtained because of the proximity of the low flow trip to the minimum flow rate during oscillation.

In performing the static tests at decreasing core average temperatures (600, 500 and 400°F), it was found that the power as indicated by the Wide Range Monitors (WRM's) and the heat balance tended to diverge. At 400°F core average temperature, the disparity between power indicated on the WRM's and the heat balance was 11%. The WRM's were originally calibrated at various power levels with core average temperature of 760°F.

Static tests at several power levels and 760°F core average temperature were repeated which showed that there was still excellent agreement between the WRM's and the heat balance at the conditions existing during calibration. The disparity was thought to be caused by a change in reflector configuration and sodium density between the core and WRM detectors; both of which apparently could affect the flux level of the detectors. The core reactivity was reduced by 1.54\$ by replacing two guinea pig rods with B₄C rods, effecting a drastically different reflector configuration at operating power. A two week program of static tests at 5 megawatts and varying coolant temperature and coolant flow rates was performed. It was concluded from these tests that the reflector position change introduced a change in the heat balance/WRM ratio of about 1.3% per 0.5\$ reflector movement. The change in sodium density with temperature results in a change of the heat balance/WRM ratio of about 2 percent per 100°F. The total change in the ratio for a reactor temperature rise of 100°F was,

therefore, about 3.3 percent. Within the apparent limits of reproducibility on the heat balance/WRM ratio (about 3%), the measurements agreed with theoretical calculation of the sodium density and control rod configuration effects. These measurements also supported the earlier indication that a systematic error may exist in the flowmeter calibration in the low flow range (≤ 1000 gpm) of as much as 100 gpm.

In addition to the special static tests, integral worths of reflectors 5 and 10 were measured. The average worth of reflector 5 was calculated to be 1.341\$ and reflector 10, 1.260\$.

Fuel Surveillance was performed at 5, 10, 15, and 17.5 MW. One rod was examined after initial 5 MW operation, and two rods after each of the other three power levels.

The fuel surveillance consists of a comparison with the pre-irradiated values for fuel rod length, cladding diameter, and rod bow. Transmission and emission gamma scans provide information on the pellet condition and pellet stack length.

Results of all examination to date are in Table 48. In summary, there have been no significant changes observed in any feature of the cladding or pellets.

As indicated in the Table, rod A17 was found to be slightly bent near the top. It is believed the bending occurred during in-cell handling and not in the reactor. The rod was originally scheduled for destruction examination at 17.5 megawatts, and is now in the irradiated fuel storage tank awaiting shipment.

The transmission gamma scans on the surveillance rods produced good fidelity scans which compared almost exactly with pre-irradiation scans. Gamma emission scans were obtained on the rods. Two methods were investigated to understand the results and to obtain the most reasonable emission results. Good emission scans were obtained both by processing the NaI output through the Victoreen 400 channel analyzer (using the single channel capacity) and into a Tracerlab countrate meter, and taking the signal directly into the Tracerlab countrate meter. The multi-channel analyzer was used to analyze the emission spectrum and to set the discriminator levels. The gamma energy range above approximately 700 KeV was used for this work, and the Zr⁹⁵-Nb⁹⁵ and Ba¹⁴⁰-La¹⁴⁰ energy peaks were observed in this spectrum. The pellet interfaces and the center spacer could be discerned from the scans as well as the total fuel column length. The length determinations from the emission scans were in good agreement with the transmission scans. An interesting result was noted at the pellet interfaces in the emission scans. The pellet interfaces at the ends of the rods are defined by distinct and sharp *depressions* in the gamma level. The pellet interfaces near the center of the rod are defined by distinct and sharp *spikes* in the gamma level. There is a short transition region at

Table 45
SEFOR EXPERIMENTAL PERFORMANCE

During August 1970

Experimental Procedure	Power MW(t)	Nominal Main Pri Flow			Nominal Core Ave. Temp °F
		30%	50%	100%	
Radiation Survey	5	-	-	-	-
TP II-1	10	-	-	-	-
Nuclear Instrumentation and Heat Balance Comp.	5		X		600
TP II-3	10			X	760
Static Test	5	X			600
TP II-4	5	X			500
	5	X			400
	5	X			760
	5		X		760
	5			X	760
	2		X		760
	5		X		760
	10		X		760
	10			X	760
	10	2000 gpm			760
	10	2000 gpm			560
Conventional Oscillator	2		X		760
TP II-5	5	X			760
	10		X		760
	10			X	760
	10*	2000 gpm			560
	5	X			760
	5		X		760
	5			X	760
Plant Heat Balance	5		X		760
TP II-6					
Natural Circulation	1**	-	-	-	-
TP II-7					
Reflector Temperature	10	-	-	-	760
Coeff. TP II-8					
Flow Oscillation Tests	5	X			760
TP II-9	10	2000 gpm			560
	10	2000 gpm			560
Monitoring of Primary and Bottom Shld. Temps.	5	-	-	-	760
TP II-10	10	-	-	-	560

*Abbreviated Oscillator Test for Balanced Oscillator Pre-Test Data.

**Test performed at a nominal 1 MW(t) power level.

Table 46

SEFOR EXPERIMENTS PERFORMED DURING SEPTEMBER, 1970

<u>Experimental Procedure</u>	<u>Power MW (t)</u>	<u>Nominal Core ΔT (°F)</u>	<u>Nominal Core Ave Temp (°F)</u>	<u>Main Pri Flow (gpm)</u>
Flow Oscillation Test	15	100	760	3750
II-9		120	760	3000
Nuclear Inst & Heat Balance Comparison	15	—	—	—
II-3				
Plant Heat Balance	5		500	700
II-6	10		745	1500
	10		760	2000
	10		515	4500
	15		760	3000
	16		595	3500
Primary Shield Temp. Measurements	15	120	760	3000
II-10				
Conventional Oscillator	10		760	2500
Tests	10		760	5000
II-5	15		760	3750
	15		760	3000
	17.5		760	4750
Static Test	10		760	2500
II-4	10		760	5000
and	15		760	3000
Group III	15		760	3750
	17.5		760	4750
	5	120	500	
	5	30	500	
	5	120	545	
	5	30	590	
	5	120	590	
	5	120	635	
	5	30	635	
	5	120	680	
	5	30	680	
	5	120	725	
	5	30	725	
	5	120	770	
	5	30	770	
	5	90	815	
	5	50	815	
	10	60	790	
	10	150	780	
	10	60	745	

Table 46 (Continued)

<u>Experimental Procedure</u>	<u>Power MW (t)</u>	<u>Nominal Core ΔT (°F)</u>	<u>Nominal Core Ave Temp (°F)</u>	<u>Main Pri Flow (gpm)</u>
Static Test	10	150	745	
II-4	10	60	700	
and Group III	10	150	700	
(Continued)	10	60	655	
	10	150	655	
	10	150	610	
	10	60	610	
	10	150	565	
	10	60	565	
	10	60	520	
	10	150	520	
	10	120	500	
	10	30	500	
	2	60	600	
	4	60	600	
	6	60	600	
	8	60	600	
	10	60	600	
	10	120	600	
	12	120	600	
	14	120	600	
	16	120	600	
	4	60	760	
	5	60	760	
	6	60	760	
	8	60	760	
	10	60	760	
	10	120	760	
	12	120	760	
	14	120	760	
	15	120	760	
Zero Power Criticality Check	0			
Reflector Temperature Coefficient	15			
II-8				

Table 47

SEFOR EXPERIMENTS PERFORMED DURING OCTOBER, 1970

5 MW Static Tests (Performed in the order listed)

Main Pri Flow (gpm)	Reactor Inlet Temp (°F)	Main Pri Flow (gpm)	Reactor Inlet Temp (°F)
4400	400	2500	700
3400	400	1500	700
2500	400	4400	745
1500	400	4400	745 (Repeat)
4400	450	1500	700
4400	500	2500	700
3400	500	3400	700
2500	500	1500	600
1500	500	2500	600
4400	550	3400	600
4400	600	1500	500
3400	600	2500	500
2500	600	3400	500
1500	600	1500	400
4400	650	2500	400
4400	700	3400	400
3400	700		

Manual Balanced Oscillator Tests

Power (MW)	Coolant Flow (gpm)	Average Coolant Temp (°F)
10	2000	760

Reflector Worth Test

Integral worths of reflectors 5 and 10 were measured.

Table 48
FUEL SURVEILLANCE RESULTS

<u>ROD A16</u>	<u>5 MW</u>	<u>15 MW</u>	<u>17.5 MW</u>
Observable Defects	None	None	None
Change in Pellet Column Length (from pre-irrad.)	<1/16"	<1/16"	<1/16"
Cladding Diametral Change (from pre-irradiation)	<0.001"	<0.001"	<0.001"
Bowing	Slight increase	No change from 5 MW	No change from 5 MW
<u>ROD A17</u>	<u>10 MW</u>	<u>15 MW</u>	<u>17.5 MW</u>
Observable Defects	None	None	None
Change in Pellet Column Length	<1/16"	<1/16"	<1/16"
Cladding Diametral Change	<0.001"	<0.001"	<0.001"
Bowing	Slight bend near top	No change from 10 MW	No change from 10 MW
	<u>ROD A01</u>	<u>10 MW</u>	
	Observable Defects	None	
	Change in Pellet Column Length	<1/16"	
	Cladding Diametral Change	<0.001"	
	Bowing	0.018 T.I.R.	

both the top and bottom of the rod, near the ends, where the pellet interfaces become indistinct. Due to the fairly clearly defined transition region, it is believed to be a temperature dependent phenomenon. Because the gamma transmission scans gave such clear definition of the fuel rod internals, an exfoliation of the fuel pellets has been discounted.

Spectrum measurements made with the multi-channel analyzer failed to show a difference between measurements at the pellet interface and at the midplane of pellets. Personnel at the Vallecitos Nuclear Center, Radioactive Materials Laboratory, have observed the same phenomenon on freshly irradiated fuel and believe the peaks are due to migration of volatile fission products to the cladding at the interfaces. The peaks are reported to disappear after a few days decay time. Rod A17 is stored in the IFST, and will be re-scanned before shipment to VNC.

Karlsruhe Laboratory personnel arrived early in September and began checkout of the automatic balanced oscillator equipment. The equipment is now fully checked and installed in cabinets in the control room. It will be connected to the reactor instrumentation and controls during the outage, and the final checks will be made before reactor operation.

Data from the Natural Circulation Test were reduced and the SEFOR contribution to the report was sent to

Sunnyvale. The report will be completed and issued at Sunnyvale.

Results of plant heat balances, TP II-6, show that the heat transfer coefficient for the Main IHX is as designed, but the coefficient for the Auxiliary IHX is about 1/2 of the design value. The discrepancy is being investigated. Because the Auxiliary IHX was oversized, the reactor will not be limited by the lower transfer coefficient.

Steps were initiated to provide for remote assembly-disassembly of the FRED poison slug. It appears that it is feasible to modify the fuel assembly-disassembly station for this purpose. Drawings of the revised poison rod and disassembly station are almost complete.

Following the 5 MW(t) fuel surveillance, a period calibration as a function of position in the core was made on the oscillator poison rod. The period calibration of the oscillator poison rod was made to resolve the nonconformity observed earlier between the static and the dynamic calibrations of the oscillator poison rod. The period calibration exhibited a close comparison with the previous static calibration.

Core excess reactivity content was increased during the 10 MW(t) fuel surveillance shutdown. Early plans were to intentionally load the full power core configuration so that it would be under-reactive. When more data had been generated and the uncertainties diminished, a reactivity

adjustment could be made to insure that Technical Specifications requirement on excess reactivity at 20 MW(t) was not exceeded. The reactivity adjustment performed amounted to an increase in reactivity of 36¢. This reactivity increase is estimated to provide approximately 41¢ excess at 20 MW(t). The adjustment was accomplished by removing a B₄C poison rod in W3.0 S2.0E (through-head port) and replacing it with a stainless steel rod. The core was poisoned for the special static tests discussed earlier, but following the special tests was returned to the 20 MW configuration.

Difficulty with the Data Acquisition System (DAS) and the shaft encoder was encountered after attaining the 5 MW(t) power, initially. A failed power supply filter in the DAS was repaired, and in checking the DAS it was discovered that the servo transmitter which drives the servo receiver (and the shaft encoder) was giving an erroneous indication. The oscillator mechanism was removed from the refueling cell and it was found that the connecting drive shaft of the servo transmitter had broken. The apparent cause was due to the rigid anchoring of each end of the shaft and then a continued relative motion between the two anchor points causing a fatigue failure in the shaft. A flexible coupling which allows unrestricted flexure while maintaining oscillator mechanism angular fidelity was designed, installed and tested.

The shaft encoder noise problem, encountered on earlier testing, was solved by using two unused inputs from the shaft encoder and a shaft encoder input logic modification. This modification triggers the DAS so that scanning is accomplished a few degrees of shaft encoder rotation later than the original design. By rotationally delaying the sampling interval, the input pulse leading edge noise, resulting from contact bounce or arcing, is avoided and erroneous sampling commands have been completely excluded.

Efforts continued on transient test preparations. The first test procedure covering the familiarization transient testing was approved and issued. The procedure for the sub-prompt transients was drafted. Estimates were made on the dose rates to be expected from the FRED slugs in anticipation of FRED slug in-cell handling. Check-out of the transient detectors continued. Due to the inability to establish whether the fission detectors were functional at SEFOR, the detectors were returned to the vendor for check-out and operational verification of the detectors, since the necessary gamma and neutron sources are not available at SEFOR.

A mass spectrometer was purchased for use during receiving inspections of the new fuel order. The leak check chamber is being modified to make it possible to perform fuel rod leak tests at 700°F.

Work is in progress to define anomalous behavior of core reactivity, reactor upper region temperature, main primary flow versus pump volts and auxiliary primary flow versus pump volts. These definitions are required as an addition to the Technical Specifications prior to reactor operation above 17.5 MW.

9.2 OPERATIONS AND MAINTENANCE

Reactor Operation Statistics

	August	Sept.	Oct.	Total for Quarter	Total to Date
Hours Critical (hrs)	353	290	263	906	1734
Power Generated (MWhrs)	2438	2459	1709	6606	7944
Startups	13	15	14	42	402
Scrams	6	7	3	16	98
Maximum Power Level (MW)	15	17.5	10	17.5	17.5
Operating Efficiency (%)	74	80	58	71	37

Longest continuous operation to date 78.9 hours, maximum power level was limited to 17.5 MW by Technical Specifications until proposed changes are approved for Tech. Specs.

This figure represents the percent of time that the reactor complex was available for performance of the test program or work necessary for the operation of the reactor, such as, normally scheduled License Test Procedures. The accountability is to the nearest full day.

A total of 131 equipment malfunctions were corrected. The malfunctions were distributed as follows:

Mechanical	43
Electrical	47
Instrumentation	41
	131

Thirty-nine (39) modifications were completed, 10 electrical and 29 mechanical.

Significant maintenance and modification activities included:

✓ Addition of capacitors to improve the power factor of the auxiliary primary pump power supply.

Addition of a flexible coupling for oscillator servo transmitter (for angular position indication).

Re-adjustment of time delay and rate for programmed flow reduction after scram.

Installation of nitrogen supply to valve operators on outdoor equipment.

Modification of energy probe housing.

✓ Replacement of oscillator positioner motor reductor.

Removal, cleaning, and re-installation of reactor vessel sodium level probe.

Maintenance of Refueling Cell fuel cleaning, handling and inspection of equipment.

Fabrication and installation of Refueling Cell to shipping cask shielding

Diagnostic tests and re-adjustment of auxiliary contacts on contactors in the safety system.

✓ Replacement of rectifier in the 26.5 VDC battery charger and SCR in 125 VDC inverter.

✓ Replacement of volt-pac cores in Auxiliary Primary Pump Power Supply and sodium pipe heater circuits.

Fabrication and installation of additional neutron and gamma shielding at localized positions.

Modification of replacement reactor sodium level probe assemblies to reduce potential for sodium bridging.

✓ Installation of switches in control circuits for manual operation (option) of freon unit de-super heater valves.

Modification of control circuitry on freon units for compressor unloading.

Fabrication and installation of freon unit test panel and instrumentation.

✓ Fabrication of a test fixture for FRED rupture discs.

Installation of snubbers in freon units compressor suction and discharge pressure gauge lines.

Repair of freon unit compressor shaft oil seal leaks and freon unit line leaks.

Five operations personnel were issued AEC Senior Operator Licenses and one, and Operator License from examinations administered in July. The SEFOR operations personnel now includes 15 with senior operator licenses, one with an operator license, and two without a license.

Radiation surveys of the Facility were made at 10 MW, 15 MW, and 17.5 MW. A few locations were identified which require additional neutron and gamma shielding. The main Secondary sodium is being activated slightly by neutrons in the vicinity of the intermediate heat exchanger. Boral sheets have been procured and will be installed during the next outage to eliminate this activation. Outside of the reactor building, some gamma leakage occurred at the floor-reactor building interface near the Equipment Access door. Concrete blocks were added to reduce the dose rate at this location. Locked barriers were added to restrict entry into areas near the Primary Sodium Equipment Door and Crane Bay Access until additional shielding is installed at these locations.

Reactor cover gas samples were obtained at each power level to establish the background activity. Analyses indicated no fission product activity above background which was predominately argon 41. The discrimination level of the nitrogen radiation monitors was increased to reduce the argon 41 background.

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