135 Fast Reactor
A & B Formal

GEAP-10010-28
AEC RESEARCH AND
DEVELOPMENT REPORT
MAY 1971

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**SOUTHWEST EXPERIMENTAL
FAST OXIDE REACTOR
DEVELOPMENT PROGRAM
TWENTY-EIGHTH QUARTERLY REPORT
FEBRUARY-APRIL 1971**

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

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SOUTHWEST EXPERIMENTAL FAST OXIDE
REACTOR DEVELOPMENT PROGRAM

Twenty-Eighth Quarterly Report
February-April 1971

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TABLE OF CONTENTS

Abstract	1
1. Introduction	2
2. Summary	6
2.1 Completed Tasks	6
2.2 Task 6 - Reactivity Controls Required for Experimentation	6
2.3 Task 9 - Fuel Fabrication	6
2.4 Task 10.4 - Approach to Power	6
2.5 Task 11 - Transient Tests	7
2.6 Safety Audit and Licensing	7
2.7 SEFOR Support	7
2.8 SEFOR Operations	7
3. Task 6 - Reactivity Controls Required for Experimentation	8
3.1 Task 6.1 - Fast Reactivity Excursion Device	8
4. Task 9 Fuel Fabrication	13
4.1 Purpose	13
4.2 Past Accomplishments	13
4.3 Current Results	13
5. Task 10 - Startup Testing	26
5.1 Purpose	26
5.2 Past Performance	26
5.3 Current Results	26
6. Task 11 - Transient Tests	50
6.1 Transient Tests Results	50
6.2 Dynamic Calibration	50
6.3 Static Calibration	50
7. SEFOR Safety Audit and Licensing	60
8. SEFOR Support	60
8.1 Reactor Vessel Head Bolts	60
8.2 Reactor Vessel Head Seal	63
8.3 Crane Retrieval	63
8.4 FRED Rod Assembly Modification	70
8.5 Argon Gas Meter	70
9. SEFOR Operation	70
9.1 Experimental Program	70
9.2 Operation and Maintenance	81
References	82
Contributors	83
Distribution	85

LIST OF ILLUSTRATIONS

Figure	Title	Page
3-1	B ₄ C Rod	9
3-2	FRED Rod—Phase I Tests	11
3-3	FRED Rod—Phase II & III Tests	12
4-1	Core II Instrumented Fuel Rods Installation	20
4-2	Plan View of IFR Locations in Reactor Through Head Ports	21
4-3	IFR Reactor Penetration Assembly	22
4-4	Instrumented Fuel Rod	23
4-5	Accuracy of Transient Fuel Temperature Rise	24
4-6	Fuel Thermocouple Installation	25
5-1	Doppler Power Reactivity Measurements	37
6-1	Flux Data From Wide Range Monitor No. 2 for 50-Cent Transient From 10 MW	53
6-2	Reactivity Data From Wide Range Monitor No. 2 for 50-Cent Transient From 10 MW	54
6-3	FRED Familiarization Poison Slug	55
6-4	FRED Rod Transit Time	56
6-5	49.5-Cent Slug-Reactivity Worth as a Function of Time	57
6-6	42.7-Cent Slug-Reactivity Worth as a Function of Time	58
6-7	40.5-Cent Slug-Reactivity Worth as a Function of Time	59
8-1	SEFOR Reactor Vessel	61
8-2	Reactor Head Bolt	62
8-3	Flange and Head Rotations	64
8-4	SEFOR Head Bolts—Strain Gages	65
8-5A	SEFOR Support—Bridge Retrieval System, Refueling Cell Plan View	66
8-5B	SEFOR Support—DETAIL—Bridge Emergency Retrieval System	67
8-6	SEFOR Support—Plan View Refueling Cell	68
8-7	SEFOR Support—10 Ton and 1/2 Ton Hoist Retrieval	69

LIST OF TABLES

Table	Title	Page
4-1	SEFOR Fuel Rods	14
4-2	SEFOR Fuel Rods Verification of the Chemical Analysis of Inert Parts	18
4-3	Fuel Thermocouple Requirements	19
4-4	Fuel Temperatures from Transient Tests	19
4-5	Fuel Thermocouple Design Features	26
5-1	Power Reactivity Data to 20 MW	28
5-2	Low Power (0.06 MW) Critical Points	28
5-3	Heat Balance Data	29
5-4	Doppler Power Reactivity Measurements	30
5-5	Power Reactivity Transfer Function and Feedback Kernel at 19 MW	31
5-6	Average Coolant Temperature to Power Transfer Function at 19 MW	32
5-7	Power Reactivity Transfer Function at 15 MW, 75% Flow, Series B	33
5-8	Average Coolant Temperature to Power Transfer Function at 15 MW, 75% Flow, Series B	34
5-9	Nominal Frequencies of Oscillation	34
5-10	Oscillator Amplitudes and Records	35
5-11	Oscillator Amplitudes and Records 15 MW, 75% Flow, Series B	36
5-12	Doppler Power Reactivity Coefficients	38
5-13	Average Fuel Time Constants	39
5-14	Doppler Energy Coefficients	41
5-15	Standard Deviation in Doppler Coefficient	43
5-16	IFA Fuel Temperatures to 20 MW	43
5-17	Reflector Temperature—Reactivity Measurements	45
5-18	First Balanced Oscillator Data 10 MW - 560°F - BOE I	46
5-19	First Balanced Oscillator Data ~10 MW - 760°F - BOE I	46
5-20	First Balanced Oscillator Data ~13 MW - 760°F - BOE I	47

LIST OF TABLES (Continued)

5-21	First Balanced Oscillator Data ~15 MW - 760°F - BOE I	47
5-22	First Balanced Oscillator Data ~15 MW - 760°F - BOE I	48
5-23	First Balanced Oscillator Data ~17 MW - 760°F - BOE I	49
5-24	First Balanced Oscillator Data ~19 MW - 760°F - BOE I	49
6-1	Familiarization Slug Reactivity Worth Measurements	51
6-2	Familiarization Slug Worth Versus Position	52
9-1	Revised Power Ascension Test Sequence	70
9-2	Post 20 MW Operation Fuel Surveillance Results	71
9-3	Transient Detector System Frequency Response Tests	75
9-4	Familiarization Transient Test Fuel Surveillances	78
9-5	Transient Test Slug Worth Measurements	80

ABSTRACT

This report summarizes the work performed on the Research and Development Program for the Southwest Experimental Fast Oxide Reactor during the period from February 1, 1971, through April 30, 1971. 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 MW, 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 characteristic 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, Eight Quarterly Report, February—April 1966," GEAP-5160 (May 1966).
9. "Southwest Experimental Fast Oxide Reactor Development Program, Ninth Quarterly Report, May—July 1966," GEAP-5208 (August 1966).

10. "Southwest Experimental Fast Oxide Reactor Development Program, Tenth Quarterly Report, August—October 1966," GEAP-5301 (November 1967).
11. "Southwest Experimental Fast Oxide Reactor Development Program, Eleventh Quarterly Report, November 1966—January 1967," GEAP-5442 (February 1967).
12. "Southwest Experimental Fast Oxide Reactor Development Program Twelfth Quarterly Report, February—April 1967," GEAP-5498 (May 1967).
13. "Southwest Experimental Fast Oxide Reactor Development Program, Thirteenth Quarterly Report, May—July 1967," GEAP-5533 (August 1967).
14. "Southwest Experimental Fast Oxide Reactor Development Program, Fourteenth Quarterly Report, August—October 1967," GEAP-5561 (November 1967).
15. "Southwest Experimental Fast Oxide Reactor Development Program, Fifteenth Quarterly Report, November 1967—January 1968," GEAP-5595 (February 1968).
16. "Southwest Experimental Fast Oxide Reactor Development Program, Sixteenth Quarterly Report, February—April 1968," GEAP-5629 (May 1968).
17. "Southwest Experimental Fast Oxide Reactor Development Program, Seventeenth Quarterly Report, May—July 1968," GEAP-5673 (August 1968).
18. "Southwest Experimental Fast Oxide Reactor Development Program, Eighteenth Quarterly Report, August—October 1968," GEAP-5705 (November 1968).
19. "Southwest Experimental Fast Oxide Reactor Development Program, Nineteenth Quarterly Report, November 1968—January 1969," GEAP-5754.
20. "Southwest Experimental Fast Oxide Reactor Development Program, Twentieth Quarterly Report, February—April 1969," GEAP-10010 (May 1969).
21. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-First Quarterly Report, May—July 1969," GEAP-10010-21 (August 1969).
22. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Second Quarterly Report, August—October 1969," GEAP-10010-22 (November 1969).
23. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Third Quarterly Report, GEAP-10010-23 (February 1970).
24. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Fourth Quarterly Report, GEAP-10010-24 (May 1970).
25. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Fifth Quarterly Report, GEAP-10010-25 (August 1970).

26. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Sixth Quarterly Report, GEAP-10010-26 (November 1970).
27. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Seventh Quarterly Report, GEAP-10010-27 (February 1971).

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. Pfisterer, 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 Stewart, S. L., "Analysis of the SEFOR Mockup Critical Experiments in ZPR-3," GEAP-5294 (March 1967).
14. Wilkinson, D., "Weighted Doppler Analysis Code—WEDOP," GEAP-5513 (October 1967).
15. Craig, E. R., Fulton, R. H., Damon, J. J., "Design and Testing of Solenoid and Instrument Lead Connector for Use in SEFOR Reactor Vessel," GEAP-5307 (March 1967).

16. Asamoto, R. R., Perry, K. J., and Zebroski, E. L., "The Effective and Diametral Thermal Expansion of UO_2 Under a Radial Temperature Gradient," GEAP-5284 (October 1966).
17. McVean, R. L., Weitzberg, A., Kremser, J. A., Leridon, A. M., and Long, J. K., "Critical Studies of the Southwest Experimental Fast Oxide Reactor (SEFOR) in ZPR-3," ANL-7348.
18. Polomik, E., "Fluid Flow and Vibration Tests, 1/2-Scale Model of SEFOR Reactor Vessel," GEAP-5440 (April 1967).
19. Field, J. H., Johnson, M. L., and Novak, P. E., "An Evaluation of the Effect of Design and Operating Variables on SEFOR Fuel and Fuel Cladding," GEAP-5309 (December 1967).
20. Noble, L. D., "Final Specifications for the SEFOR Experimental Program," GEAP-5576 (April 1968).
21. Russell, C. E., "Flow Distribution Measurements in a 3-Scale Model of the SEFOR Fuel Bundle," GEAP-5184 (May 1968).
22. Cochran, J., "SEFOR Core Clamping Model Tests, One-Sixth Segment," GEAP-5465 (May 1968).
23. Asamoto, R. R., and Movak, 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. Cockran, 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 (February 1970).
31. Meyer, R. A., Stewart, S. L., Craig, E. R., Reynolds, A. B., and Johnson, M. L., "Design and Analysis of SEFOR Core I," GEAP-13598 (April 1970).
32. McKeehan, E. R., "Design and Testing of the SEFOR Fast Reactivity Excursion Device (FRED)," GEAP-13649 (October 1970).
33. Noble, L. D., et al., "SEFOR Core I Test Results to 20 MW," GEAP-13702 (March 1971).

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

The Phase I (<50 cents), II (<98 cents) and III (<130 cents) absorber rods have been fabricated. The Phase I rod was used in the familiarization testing at SEFOR. The Phase II rod is at SEFOR ready for use in the sub-prompt tests. The Phase III rod will remain at GE-BRDO until the sub-prompt tests are completed. Poison rod surface emissivity measurements were made.

2.3 TASK 9 - FUEL FABRICATION

Nuclear Fuel Services (NFS) completed the fabrication of SEFOR fuel rods during the report period. A total of 104 new fuel rods were shipped to the SEFOR site along with 19 repaired fuel rods and two unrepairable fuel rods. The loose fuel pellets (20.5 kg) were shipped to the GE Vallecitos Nuclear Center.

Receiving inspection of the fuel rods and loose fuel pellets was started.

Sample fuel pellets from all fuel pellet batches were received by GE and the verification analysis was completed on each batch.

The sixth instrumented fuel assembly (IFA) was found to have two sensor circuit failures at final inspection. Rework of the defective part of the sixth IFA was planned.

The final design drawings for the Core II instrumented fuel rods are near completion. The design, fabrication, and quality assurance plan specifications are in preparation. The fuel thermocouple operational requirements were defined, and the design is in progress.

A thermal transient response test of different type thermocouple junctions was performed. The fuel thermocouple purchase specification is in preparation. Both tungsten/rhenium and platinum/molybdenum thermocouples are being considered. Material orders for thin- and thick-wall tubing and bar stock for the fuel cladding, extension rod, and end plugs, respectively, are in progress.

2.4 TASK 10.4 - APPROACH TO POWER

Measurements of the reactivity feedback in SEFOR have been performed at power levels up to 20 MW and at average coolant temperatures between 350°F and 760°F. During these measurements the mixed oxide ($\text{PuO}_2\text{-UO}_2$) fuel temperatures ranged between a minimum of 350°F and a calculated maximum of approximately 4800°F. Analysis of the static and time-dependent experimental data has provided a measured power-Doppler reactivity relationship with an estimated standard deviation of $\pm 8\%$. Comparison of the results with calculations demonstrated that the Doppler component of the feedback can be described by a fuel temperature coefficient $T(\text{dk}/\text{dT})$ of -0.0081 with an estimated standard deviation of ± 0.0014 . This coefficient compares favorably with the value of -0.0085 that was predicted¹ prior to the measurements.

The SEFOR reflector temperature coefficient of reactivity was measured at power levels of 5, 10, 15, and 20 MW. The maximum observed change in reactivity at any power level due to temperature changes in excess of 100°F was 1.1 cents. At all power levels the reactivity appeared to decrease slightly with increasing reflector temperature. The magnitudes of the reactivity changes were less than the estimated standard deviation (± 1.6 cents) in reactivity changes at power, however, and for all practical purposes the reflector temperature coefficient of reactivity may be treated as zero.

2.5 TASK 11 - TRANSIENT TESTS

Transients involving rapid insertion of positive reactivities up to 50 cents (the familiarization transients) have been completed. A preliminary analysis of the flux data recorded during the transient was performed to determine the time-dependent reactivity. This "dynamic" reactivity agreed well with the results which had been predicted from the previous measurements of FRED worth vs position, and position vs time.

2.6 SAFETY AUDIT AND LICENSING

An audit trip was made to SEFOR during February 16-19. Final copies of the SEFOR Technical Specifications pages affected by Changes No. 2 and No. 3 were received from the Division of Reactor Licensing (DRL). Four modification requests were reviewed and approved. Three malfunction reports were sent to the DRL. Reports summarizing the results of natural circulation testing and testing of the Fast Reactivity Excursion Device were issued. An investigation of head bolt tension and related reactor vessel cover gas leakage was accomplished and the results were discussed with the AEC.

2.7 SEFOR SUPPORT

A test and evaluation program concerning the reactor head bolts was completed. Diagnostic tests and analyses were completed to measure the reactor head bolt stresses and reactor vessel flange rotations. As a result of these tests, examinations, and analyses, corrective measures were taken by reducing the reactor outlet sodium heatup and cool-down rate to 10°/hr for temperature changes in excess of 125°F and reducing the bolt preload stresses.

Design work was initiated on a seal for the reactor head. The seal is designed as a backup to prevent reactor cover gas from leaking past the outer head O-ring seal.

Methods of retrieving refueling cell crane loads following malfunctions or failures have been under investigation, and the design of retrieval equipment has been completed which allows efficient recovery in the event of unforeseen equipment malfunctions.

2.8 SEFOR OPERATIONS

AEC approval was received for proposed changes PC-3 and PC-4 to the Technical Specifications. These changes permitted steady-state and transient testing at power levels up to full power (20 MWt) with guinea pig rods in the reactor core. Static and oscillator tests at power levels up to 20 MW were completed and associated fuel inspections were completed.

During the subsequent plant outage the oscillator was replaced with the Fast Reactivity Excursion Device (FRED) and modifications completed to permit transient testing.

All planned tests (8) in the Familiarization Transient Test series were completed including the associated fuel inspections. The FRED slug for the Sub-prompt Transient Test series was calibrated, reactivity worth adjusted, and recalibrated in preparation for the sub-prompt transient tests.

A reactor outage was initiated to investigate the loading on the reactor vessel outer-head studs from differential movement of vessel flange and head during temperature changes of primary coolant. Tests have been completed and data are being evaluated.

Plant modifications and maintenance activities were performed.

3. TASK 6 - REACTIVITY CONTROLS REQUIRED FOR EXPERIMENTATION

3.1 TASK 6.1 - FAST REACTIVITY EXCURSION DEVICE (FRED)

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 excursion during the experimental program.

3.1.2 Past Accomplishments

The FRED design, fabrication, and testing have been completed and a topical report on the design and testing has been issued.¹

3.1.3 FRED Rod

The experimental program requires neutron absorber rods with three different reactivity worths for the FRED tests. The FRED rods are designed to enable reactivity adjustment and remote assembly and disassembly from the extension shaft. For the latter purpose the fuel rod assembly-disassembly equipment in the refueling cell was modified to accommodate the FRED absorber rod as well as the fuel rods.

The absorber rod is attached to the extension adapter as shown in Figure 3-1 in a manner similar to the pin joint used on the SEFOR fuel rods.

The three rods are similar in length, weight, and worth adjustment method. The Phase I rod (Figure 3-2) differs from the Phase II and III rods (Figure 3-3) in that it contains vibratory-compacted, natural B_4C . The Phase II and III rods contain pelletized, enriched B_4C .

Should it prove necessary, design provisions provide for reactivity adjustment by adjusting the elevation of the B_4C column with respect to the core midplane. The B_4C column can be moved up or down within the core over a 6-inch span by shortening or lengthening the handle (Figures 3-2 and 3-3). A corresponding length is either added or removed from the bottom to keep the rod a fixed length and weight. The B_4C assembly remains sealed during the adjustment modification.

3.1.4 FRED Rod and Drywell Emittance Measurements

The thermal emittance properties of five oxidized type 304 stainless steel samples were determined at temperatures up to 1650°F and at wavelengths between 2 and 14 microns.

Three samples were fabricated from a piece of the same 1-inch schedule 40 pipe used to fabricate the FRED slugs. Both the slug and the three samples were oxidized at the same time and placed under the same conditions. Oxidation was performed in air at 1650°F for 12 hours. The total hemispherical emittance value at 1650°F was determined to be 0.82 and the corresponding total normal emittance was 0.86.

The other two samples were measured to obtain a better idea of the emittance properties of the drywell surface which surrounds the FRED rod. Both samples were oxidized in an atmosphere of wet hydrogen at 900°F for 8 hours which was the same surface preparation used for the drywell used at SEFOR. The total hemispherical emittance value was determined and from that information and spectral reflectance data an effective absorptance value of 0.57 was obtained.

Evaluation of these results is in progress; initial indications are that there will be no significant limitation on the transient test program as a result of predicted FRED temperatures.

NOTES

1. FABRICATION PROCEDURE:

- INSTALL STEEL WOOL, P10, AROUND PART 4 AS SHOWN AND INSERT INTO TUBE, P7, FROM BOTTOM END.
 - WELD END PLUG, P5, TO TUBE, P7, PURGING WELD ID WITH HELIUM.
 - FILL WITH 175 GRAMS OF NATURAL BORON CARBIDE, P11, COMPACTED TO 71% THEORETICAL DENSITY (T.D. = 2.51 G/CC).
 - INSTALL P10, P3, P9, P3 AND P8, (TACK WELD AS SHOWN).
 - PER GE SPECIFICATION 22A1971, SECTION 4.4.2 EVACUATE TO 500 MICRONS OR LESS, BACKFILL WITH BUREAU OF MINES GRADE A HELIUM TO ONE ATMOSPHERE. POSITION TOP END PLUG, P2, AND WELD.
 - INSTALL FILLER ROD, P12, INTO TOP END PLUG, P2 & PLUG WELD AS SHOWN.
- LEAK CHECK SEALED SUBASSEMBLY PER GE SPECIFICATION 22A1971, SECTION 4.4.4.
 - WELD HANDLE, P1, TO TOP END PLUG, P2, AND NOSE, P6, TO BOTTOM END PLUG, P5, AS SHOWN PER SPECIFICATION 22A1971, SECTION 4.4.2.
 - WELDS MUST NOT EXCEED DIA. OF TUBE, P7.
 - ASSEMBLY TO BE $\bar{C}20$ TOTAL.

*OR ENG APPD EQUIV.

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES--
TOLERANCES ON:
2 PLACE DECIMALS $\pm$
3 PLACE DECIMALS $\pm$
ANGLES $\pm$
FRACTIONS $\pm$

ALL
SURF.

✓

FCF-277E255

1	12	FILLER ROD	.063 DIA X 3/16 LG	TYPE 308 SST
AR	11	NATURAL BORON CARBIDE	GE. SPEC. 22A1969 REV.0	
AR	10	STEEL WOOL	EXTRA FINE GRADE	TYPE 304 SST
	9	SPRING	INC 090D-B INCONEL X #1 TEMPER	SEE SPRING*
	8	TUBE	3/4 O.D. X .049 WALL X 4 1/2 LG., ASTM A-269 304 SST	
	7	TUBE	178 B3142 PI	
	6	NOSE	226 A5133 PI	
1	5	BOTTOM END PLUG	226 A5132 PI	
1	4	FILLER TUBE	178 B3140 GI	
2	3	DISC	226 A5135 P2	
1	2	TOP END PLUG	226 A5131 PI	
1	1	HANDLE	178 B3138 PI	
GI	PART NO.	NAME	DESCRIPTION, MAT'L, DWG. NO.	

GEAP-10010-28

Figure 3-2. FRED Rod-Phase I Tests

FCF-277E 255

NOTES

1. FABRICATION PROCEDURE:

- INSTALL STEEL WOOL, P10, AND SPACER, P6, INTO TUBE, P3, FROM BOTTOM END.
 - WELD END PLUG, P4, TO TUBE, P3, PURGING WELD ID WITH HELIUM.
 - G1 - LOAD FIFTEEN B4C PELLETS, P11, INTO TUBE, P3.
G2 - LOAD FIFTEEN B4C PELLETS, P12, INTO TUBE, P3.
 - INSTALL STEEL WOOL, P10, DISC, P7, SPRING, P9, DISC, P7, AND TUBE, P8, (TACK WELD AS SHOWN).
 - PER GE SPECIFICATION 22A1971, SECTION 4.4.2 EVACUATE TO 500 MICRONS OR LESS, BACKFILL WITH BUREAU OF MINES GRADE A HELIUM TO ONE ATMOSPHERE. POSITION TOP END PLUG, P2, AND WELD.
 - INSTALL FILLER ROD, P13, INTO TOP END PLUG, P2, AND PLUG WELD AS SHOWN.
- LEAK CHECK SEALED SUBASSEMBLY PER GE SPECIFICATION 22A1971, SECTION 4.4.4.
 - WELD HANDLE, P1, TO TOP END PLUG, P2, AND EXTENSION, P5, TO BOTTOM END PLUG, P4, AS SHOWN PER SPECIFICATION 22A1971, SECTION 4.4.2.
 - WELDS MUST NOT EXCEED DIA. OF TUBE, P3.
 - ASSEMBLY TO BE $\pm .020$ TOTAL.

* OR ENGR. APPD EQUIV.

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES-
TOLERANCES ON
2 PLACE DECIMALS $\pm$
3 PLACE DECIMALS $\pm$
ANGLES $\pm$
FRACTIONS $\pm$

ALL
SURF.
✓

1	1	13	FILLER ROD	.063 DIA x 3/16 LG TYPE 308SST
15		12	PELLET	226A5053 P2
15	11	11	PELLET	226A5053 P1
AR	AR	10	STEEL WOOL	EXTRA FINE GRADE, TYPE 304SST
		9	SPRING	226A5141 P1
		8	TUBE	1" O.D. x .049 WALL x 4 1/2 LG. ASTM A-269 304SST
2	2	7	DISC	226A5135 P1
1		6	SPACER	226A5138 P2
1		5	EXTENSION	178B3141 P3
		4	BOTTOM END PLUG	226A5137 P1
1		3	TUBE	178B3143 P1
1		2	TOP END PLUG	226A5131 P1
		1	HANDLE	178B3138 P2
G2	G1	PART NO.	NAME	DESCRIPTION, DWG. NO., MAT'L

GEAP-10010-28

Figure 3-3. FRED Rod-Phase II & III Tests

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 core component and fuel rod order has been received at SEFOR in previous quarters. The stainless steel tightener rods for Core II were received. Five instrumented fuel assemblies were shipped to the SEFOR site. Fabrication was started on 102 additional fuel rods. Conceptual design of SEFOR Core II instrumented fuel rods was completed. The design includes three rods located at three separate through-head ports. Each rod contains one fuel thermocouple. Scoping drawings were made of fuel thermocouple design options, the instrumented fuel rod with vessel penetration of thermocouple leadout, and rod handling from shipment to post-irradiation storage.

4.3 CURRENT RESULTS

4.3.1 Fuel Rod Procurement

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

4.3.1.1 Repair of Fuel Rods

Nineteen repaired and two unrepairable fuel rods were shipped to the SEFOR site. Receiving inspection of the fuel rods was started.

4.3.1.2 Fabrication of New Fuel Rods

The process of fuel pellets was completed during the report period. An additional batch was processed so there was a total of six fuel pellet batches for this production run (701A, 802, 803, 804, 805, and 806) instead of five as previously reported.

Fuel pellets from part of Batch 803 and all of 804 and 805 were resintered successfully to reduce the oxygen-to-metal (O/M) ratio to the specified range of 1.97 to 1.99.

A temperature and time were developed for resintering that did not result in pellet shrinkage but were sufficient to change the O/M ratio. This was necessary since the pellets were ground to finish size. When the batches were resintered the sampling frequency was doubled as provided in the NFS Quality Control Procedure (NFS-M-10). Fuel pellet batch 806 was sintered after the O/M problem was resolved and the O/M ratios were within specification.

The problem with O/M ratio was traced to a deficiency in the sintering furnace atmosphere. It was determined that the volume of nitrogen sweep gas at the exit end of the sintering furnace was inadvertently increased sometime during the sintering of Batch 803. The nitrogen back diffused into the furnace atmosphere (normally 6-8% H₂ in argon) decreasing the hydrogen concentration in the furnace atmosphere.

The electron beam welding of center spacers into the tubes was completed.

Loading of fuel pellets into tubes, welding end plugs to the rods, and final processing and inspection were completed. A total of 104 rods was produced as shown in Table 4-1.

Inspection of the fuel rods included a gamma scan of each fuel rod to check the plutonium content of each rod. All fuel rods passed the gamma scan acceptance criteria successfully.

The fuel rods were shipped to the SEFOR site and the loose fuel pellets (20.5 kg) required by the GE-NFS contract were shipped to GE Vallecitos Nuclear Center in March 1971. Receiving inspection on the fuel rods and loose pellets was started.

4.3.1.3 Quality Assurance

The final audit of the NFS activities and records was completed by the GE QC Resident Engineer. Certifications were received from NFS for the individual fuel batches and fuel rods.

Fuel pellet batch samples were received at GE Vallecitos Nuclear Center for Batches 804, 805, and 806. The verification analysis for all fuel pellet batches was completed.

As part of the receiving inspection of the new and repaired fuel rods, the end plug welds were radiographed. This inspection was done at the fuel manufacturer's plant prior to the shipment of the fuel rods to the SEFOR site. The evaluation of the end plug weld radiographs was started.

**Table 4-1
SEFOR FUEL RODS**

Fuel Rod Number	Inert Parts (1)		Fuel Pellets				Weld Lot Number	Total Weight (grams)
	Tube (316SS) Heat Number	Bottom End Plug (304SS) Heat Number	Batch Number	Fuel Weight (grams)	Pu Weight (grams)	Pu 239 & 241 Weight (grams)		
001	26081	812750	802	3368.37	608.33	555.78	6;11	5046.0
002	26081	E110613	802	3359.57	606.74	554.33	6	5040.0
003	26081	812750	802	3357.55	606.37	554.00	6;11	5037.0
004	26081	E110613	802	3357.07	606.29	553.92	6	5035.0
005	26081	E110613	802	3362.35	607.24	554.79	5	5041.0
006	26081	812750	802	3355.27	605.96	553.62	6;11	5036.0
007	26081	E110613	802	3358.93	606.62	554.22	6	5033.0
008	26081	812750	802	3351.15	605.22	552.94	5;12	5030.0
009	26081	812750	802	3366.17	607.93	555.42	6;11	5045.0
010	26081	812750	802	3366.25	607.93	555.43	5;11	5047.0
011	26081	812750	802	3354.43	605.81	553.48	6;11	5031.0
012	26081	812750	802	3363.75	607.49	555.02	5;11	5044.0
013	26081	812750	802	3353.83	605.70	553.38	6;11	5033.0
014	26081	E110613	802	3353.25	605.60	553.29	5	5033.0
015	03697	812750	802	3355.30	605.97	553.62	7;11	5031.0
016	03697	812750	701A	3365.07	607.73	555.24	6;12	5038.0
017	26081	E110613	802	3341.35	603.45	551.32	5	5024.0
018	26081	E110613	Top 701A Bottom 802	3360.27	606.86	554.44	7	5043.0
019	26081	E110613	802	3353.17	605.58	553.27	7	5039.0
020	26081	E110613	802	3344.47	604.01	551.84	7	5028.0
021	26081	E110613	802	3342.27	603.61	551.47	7	5019.0
022	26081	E110613	802	3332.47	601.84	549.86	7	5010.0
023	26081	E110613	802	3342.35	603.63	551.49	5	5028.0
024	26081	E110613	802	3339.17	603.05	550.96	7	5019.0
025	26081	812750	802	3339.27	603.07	550.98	7	5021.0
026	03697	E110613	802	3344.07	603.94	551.77	7	5025.0
027	03697	812750	802	3347.87	604.63	552.40	7;12	5023.0
028	03697	E110613	802	3335.67	602.42	550.38	8	5009.0
029	03697	E110613	802	3344.67	604.05	551.87	8	5021.0
030	03697	E110613	Top 803 Bottom 802	3338.67	602.96	550.88	8	5015.0
031	03697	812750	803	3355.67	606.03	553.69	8.13	5030.0
032	03697	E110613	803	3342.77	603.70	551.56	8	5018.0
033	03697	E110613	803	3349.47	604.91	552.66	8	5029.0

Table 4-1 (Continued)

Fuel Rod Number	Inert Parts ⁽¹⁾		Fuel Pellets				Weld Lot Number	Total Weight (grams)
	Tube (316SS) Heat Number	Bottom End Plug (304SS) Heat Number	Batch Number	Fuel Weight (grams)	Pu Weight (grams)	Pu 239 & 241 Weight (grams)		
034	03697	E110613	803	3376.27	609.75	557.08	8	5050.0
035	03697	E110613	803	3371.27	608.85	556.26	8	5049.0
037	03697	812750	803	3373.27	609.21	556.59	8;12	5051.0
038	03697	812750	803	3372.27	609.03	556.42	8;12	5048.0
040	03697	E110613	803	3369.97	608.62	556.05	9	5046.0
041	03697	E110613	803	3377.67	610.01	557.32	9	5054.0
042	03697	812750	803	3380.57	610.53	557.79	9;12	5057.0
044	03697	812750	804	3345.90	604.27	552.07	21	5025.0
045	03697	E110613	803	3374.77	609.48	556.84	9	5055.0
046	03697	E110613	803	3384.97	611.33	558.52	9	5063.0
047	03697	E110613	803	3386.87	611.67	558.83	9	5067.0
048	03697	812750	803	3382.07	610.80	558.04	9;13	5061.0
050	03697	E110613	803	3382.97	610.96	558.19	9	5060.0
051	03697	E110613	803	3391.57	612.52	559.61	9	5067.0
052	03697	812750	803	3387.47	611.78	558.93	10	5067.0
053	03697	812750	803	3383.47	611.05	558.27	10	5061.0
054	03697	812750	803	3374.87	609.50	556.85	10	5055.0
055	03697	812750	803	3379.37	610.31	557.60	10	5054.0
056	03697	812750	Top 804 Bottom 803	3371.57	608.91	556.31	10;15	5050.0
061	03697	812750	805	3376.47	609.79	557.12	19;20	5056.0
067	03697	812750	804	3360.07	606.83	554.41	19	5038.0
068	03697	812750	805	3368.17	608.29	555.73	19	5044.0
069	03697	812750	803	3358.90	606.62	554.22	14	5040.0
072	03697	812750	804	3360.17	606.85	554.43	14	5036.0
073	03697	812750	804	3357.57	606.38	554.00	14	5041.0
074	03697	812750	804	3363.47	607.44	554.97	14	5049.0
075	03697	812750	804	3363.37	607.97	555.45	14	5053.0
076	03697	812750	804	3364.07	607.55	555.07	14	5045.0
077	03697	812750	803	3371.07	608.82	556.23	14	5051.0
078	03697	812750	803	3375.07	609.54	556.89	14	5055.1
079	26081	812750	803	3371.27	608.85	556.26	14	5048.0
080	26081	812750	803	3361.17	607.03	554.59	15	5043.0
081	26081	812750	804	3367.77	608.22	555.68	15	5047.0
082	26081	812750	804	3367.97	608.26	555.72	15	5050.0
083	26081	812750	804	3372.07	609.00	556.39	15	5055.0
084	26081	812750	804	3370.27	608.67	556.09	15	5055.0
085	26081	812750	804	3374.67	609.47	556.82	15	5055.0

GEAP-10010-28

Table 4-1 (Continued)

Fuel Rod Number	Inert Parts (1)		Fuel Pellets					Total Weight (grams)
	Tube (316SS) Heat Number	Bottom End Plug (304SS) Heat Number	Batch Number	Fuel Weight (grams)	Pu Weight (grams)	Pu 239 & 241 Weight (grams)	Weld Lot Number	
086	26081	812750	804	3362.57	607.28	554.82	15	5041.0
087	26081	812750	805	3363.57	607.46	554.99	15	5043.0
088	26081	812750	805	3377.57	609.99	557.30	15	5056.0
089	26081	812750	805	3364.57	607.64	555.15	16	5047.0
093	26081	812750	805	3363.57	607.46	554.99	16	5043.0
095	26081	812750	805	3378.57	610.17	557.46	16	5059.0
096	26081	812750	805	3366.17	607.93	555.42	16;19	5045.0
097	26081	812750	805	3368.57	608.36	555.81	16	5051.0
098	26081	812750	805	3369.67	608.56	556.00	16	5054.0
099	26081	812750	805	3366.17	607.93	555.42	16	5049.0
100	26081	812750	805	3365.47	607.80	555.30	16	5049.0
930	26081	812750	805	5373.07	609.18	556.56	16	5044.0
931	26081	812750	805	3369.97	608.62	556.05	16	5057.0
933	26081	812750	805	3378.37	610.13	557.43	17	5061.0
934	26081	812750	805	3373.27	609.21	556.59	17	5055.0
935	26081	812750	805	3371.57	608.91	556.31	17	5055.0
936	26081	812750	805	3368.37	608.33	555.78	17	5054.0
937	26081	812750	805	3383.50	611.06	558.28	17	5066.0
938	26081	812750	805	3368.17	608.29	555.75	17	5054.0
939	26081	812750	805	3368.57	608.36	555.81	17	5049.0
940	26081	812750	805	3371.84	608.95	556.35	17	5051.0
941	26081	812750	805	3382.37	610.86	558.09	17	5059.0
942	26081	812750	805	3370.37	608.69	556.11	17	5051.0
943	26081	812750	805	3371.17	608.94	556.34	18	5053.0
944	26081	812750	805	3369.97	608.62	556.05	18	5053.0
945	26081	812750	805	3374.27	609.39	556.75	18	5056.0
946	26081	812750	804	3359.07	606.65	554.25	18	5038.0
947	26081	812750	804	3361.77	607.14	554.69	18;19	5046.0
948	26081	812750	804	3359.67	606.76	554.35	18	5042.0
949	26081	812750	804	3361.97	607.17	554.73	18	5038.0
950	26081	812750	804	3361.17	607.03	554.59	18	5038.0
951	26081	812750	804	3360.67	606.94	554.51	18	5041.0
953	26081	812750	804	3361.87	607.15	554.71	19	5045.0
954	26081	812750	804	3357.67	606.40	554.02	19	5037.0

(1) All center spacers from HT. 27275 (304SS)
 All nickel reflectors from HT. N4214A (Ni)
 All bottom springs from HT. 9467X (Inconel 750)

All discs attached to bottom springs from HT. 135780F (304SS)
 All top springs from HT. 8305J2 (Inconel 750)
 All top end plugs from HT. 27275 (304SS)

During the fabrication campaign for the fuel rods, the GE Resident Engineer selected, at random, a sample of each finished inert part that makes up the fuel rod. If more than one heat of material was used to fabricate a part, a sample from each heat was obtained. The samples were analyzed for chemical composition to verify the composition of materials used for the fuel rods. The results of these analyses are given in Table 4-2.

4.3.2 First Core Instrumented Fuel Assemblies (IFA)

Fabrication and inspection of the sixth IFA was completed. The final acceptance of the unit has been delayed due to a sensor circuit failure which was found during final inspection.

The work accomplished for this report period included the seal block braze, forming of the ribbon lead section, the bulkhead braze, and the joining of the two sections of the IFA.

A short circuit in a fuel thermocouple and an open circuit (intermittent) in one of the sodium outlet thermocouples were found during the final QA inspection of the sixth IFA. A time domain reflectometer (circuit analyzing instrument) was used to locate the faults and provide guidance in determining the proper corrective action. Rework of the defective part of the sixth IFA was planned.

4.3.3 Core II Instrumented Fuel Rods (IFR)

4.3.3.1 General Description

One SEFOR Core II Instrumented Fuel Rod (IFR) is located in each of three SEFOR through-head port penetrations, as shown in Figures 4-1 and 4-2. Figure 4-3 shows a detail of the IFR penetration assembly. The spring-loaded assembly of the IFR extension rod and seal plate assembly allows for any differential thermal expansion between the IFR and the reactor vessel. The fuel thermocouple leadout penetrates the through-head port through the seal plate.

The instrumented fuel rod consists of five principal subassemblies: seal plate assembly, extension rod, T/C fuel rod, fuel thermocouple, and mini-fuel rod. The IFR is shown in Figure 4-4.

4.3.3.2 Functional Requirements

The operational requirements for the fuel thermocouple are shown in Table 4-3. These requirements are based on a representative Core II super-prompt critical transient test program shown in Table 4-4. This table assumes a 3-hour hold at temperature for calibration of the neutron detectors at 5, 10, 15, and 20 MW reactor power. The times at temperature also include a 4-hour hold at 1, 5, and 10 MW before running each power transient. Four transients (two with 1.10 dollars and two with 1.18 dollars reactivity worth) are assumed for the three power levels for a total of twelve. A prime requirement for the fuel thermocouple is the 1/2-second speed of response. The expected uncertainty in the measurement of the fuel temperature during a transient run as a function of the thermocouple time constant is shown in Figure 4-5. As noted from this figure, the uncertainty in fuel temperature measurement increases rapidly for the slower responding thermocouples. From oscillator test data, the grounded 1/16 diameter W/Re thermocouples in the Core I IFA's have a time constant of approximately 2 seconds (the Core I IFA's also contain bare-wire thermocouples for fast transient response). From these operational requirements, a special bare wire W/Re thermocouple design was chosen. An alternate thermocouple using Pt/0.1% Mo vs Pt/5% Mo swaged in Pt/Rh sheath to 30 to 40 mils diameter is being considered. Design features of the W/Re thermocouple are shown in Table 4-5. The bare wire junction is needed to meet the speed of response requirement, W/Re thermoelements have the temperature and time operational capability, and use of the stranded wire provides the ductility to W/Re needed for the transient thermal shock requirement. A typical installation of the thermocouple in the fuel rod is shown in Figure 4-6. The transition has a special flange that allows for seal welding the thermocouple to the fuel cladding spacer. The transition includes two ceramic-to-metal seals for containment of the fission gases. The swaged leadout wire uses extension wires that are compensated for W/Re at the high operating temperatures of the transition area.

4.3.3.3 Design Status

The final design drawings are near completion. These drawings will be reviewed in conjunction with respective fabrication specification, i.e., Mini Fuel Rod assembly and detail part drawings with the specification. The preliminary design, extension rod, and quality assurance specifications were completed and are being reviewed. The fuel rods, weld and braze, fuel thermocouple, and seal plate assembly specifications are in preparation.

The fuel thermocouple design is in progress. Several design features that have proven successful for other experimentors are part of this thermocouple. Fabrication techniques and ultra-clean processing methods that are presently being used for similar W/Re thermocouples will become part of the requirements for this thermocouple.

Table 4-2

SEFOR FUEL RODS

VERIFICATION OF THE CHEMICAL ANALYSIS OF INERT PARTS

	Weight Percent											
	C	Mn	P	S	Si	Cr	Ni	Mo				
Fuel Rod Tubing												
Specified: 316SS; ASTM A269	0.08	2.0	0.045	0.030	1.0	16.0	10.0	2.0				
	Max	Max	Max	Max	Max	18.0	14.0	3.0				
Heat Number 26081												
Certified Analysis from NFS	0.046	1.14	0.024	0.027	0.42	17.34	13.79	2.20				
Verification Analysis	0.053	1.20	0.023	0.023	0.45	17.51	14.0	2.27				
Heat Number 03697												
Certified Analysis from NFS	0.013	1.79	0.010	0.026	0.48	16.87	13.57	2.18				
Verification Analysis	0.014	1.82	0.14	0.021	0.50	16.94	14.16	2.29				
Top End Plugs and Spacers	C	Mn	P	S	Si	Cr	Ni					
Specified: 304SS; ASTM A276	0.08	2.00	0.045	0.030	1.0	18.0	8.0					
	Max	Max	Max	Max	Max	20.0	12.0					
Heat Number 27275												
Certified Analysis from NFS	0.023	1.41	0.013	0.012	0.43	18.37	10.23					
Verification Analysis	0.017	1.39	0.012	0.014	0.48	18.23	10.08					
Bottom End Plugs	C	Mn	P	S	Si	Cr	Ni					
Specified: 304SS; ASTM A276	0.08	2.00	0.045	0.030	1.0	18.0	8.0					
	Max	Max	Max	Max	Max	20.0	12.0					
Heat Number E110613												
Certified Analysis from NFS	0.022	1.66	0.035	0.024	0.74	18.26	9.68					
Verification Analysis	0.030	1.65	0.030	0.017	0.79	18.37	9.55					
Heat Number 812750												
Certified Analysis from NFS	0.023	1.15	0.021	0.020	0.33	18.40	9.61					
Verification Analysis	0.021	1.20	0.018	0.022	0.35	18.12	9.25					
Reflectors	C	Mn	Fe	S	Si	Cu	Ni					
	0.15	0.35	0.40	0.010	0.35	0.25	99.0					
Specified: Nickel; ASTM B160	Max	Max	Max	Max	Max	Max	Min					
Heat Number N4 214A												
Certified Analysis from NFS	0.08	0.24	0.11	0.005	0.04	0.01	99.49					
Verification Analysis	0.079	0.24	0.13	0.004	<0.02	0.01	99.7					
Springs	C	Mn	S	Si	Cr	Ni	Cu	Fe	Co	Al	Ti	Cb/Ta
Specified: Inconel 750; ASTM A461	0.08	1.0	0.01	0.5	14.0	70.0	0.5	5.0		0.4	2.25	0.70
	Max	Max	Max	Max	17.0	Min	Max	7.0		1.0	2.75	1.20
Heat Number 9467X												
Certified Analysis from NFS	0.05	0.10	0.007	0.22	15.53	72.85	0.03	7.14	0.14	0.80	2.37	0.88
Verification Analysis	0.07	0.54	0.005	0.40	14.80	73.3	0.04	6.87	0.08	0.66	2.25	0.82
Heat Number 8305J2												
Certified Analysis from NFS	0.05	0.09	0.007	0.20	15.65	72.85	0.04	6.80	0.04	0.73	2.65	0.91
Verification Analysis	0.064	0.10	0.005	0.23	15.58	73.2	0.05	6.68	0.05	0.71	2.25	0.85

Table 4-3

FUEL THERMOCOUPLE REQUIREMENTS

Temperature Range	Up to 3200°
Thermal Cycles	24 Cycles at 3000° F
Time at Temperature	250 Hours
Speed of Response	≤0.5 Second
Thermal Shock	800° F/Second
Neutron Fluence	10 ²⁰ nvt
Flux Energy, Fast	>10 keV
Accuracy	±2%
Sensitivity (Percent of Change)	±2%

Table 4-4

FUEL TEMPERATURES FROM TRANSIENT TESTS

Temperature* (°F)	Reactor Power (MW)	Transient (Dollars)	Cycles (Number)	Time at Temperature
870	1	--	--	16 hours
870 to 900	1	1.10	2	< 20 seconds
870 to 910	1	1.18	2	< 20 seconds
1330	5	--	--	52 hours
1330 to 1480	5	1.10	2	< 20 seconds
1330 to 1530	5	1.10	2	< 20 seconds
1840	10	--	--	52 hours
1840 to 2140	10	1.10	2	< 20 seconds
1840 to 2240	10	1.18	2	< 20 seconds
2405	15	--	--	36 hours
2975	20	--	12	36 hours

*Temperatures based on central fuel temperature for a pellet with 0.070-inch core, at a radial position of 7.8 inches from core centerline, and at an axial position of 5 inches below top core fuel.

Thermal response tests for comparing the relative response times of bare wire, grounded, and ungrounded thermocouples in vacuum and helium atmospheres are nearing completion. The tests are performed using induction heating of a tungsten cylindrical block. The thermocouple is inserted through a hole into the center of the block. The thermocouple is spaced in the radial center of the hole by a ceramic bead. The block temperature is measured with an infra-scope sighting into a special small diameter hole located on the back of the block. The millivolt output of the thermocouple and infra-scope are traced out on a high-speed recorder. The recorder is equipped with time and special event markers. Special event markers are used to note power on and off conditions and time is scaled in 1-second intervals. The transient test runs are made for thermocouple temperatures between 1500° and 2100° F.

Preliminary test results do indicate, as expected, that the bare wire response time is faster than the grounded and ungrounded type thermocouples. The thermocouples respond about three times faster in a helium atmosphere compared to the vacuum for the geometrical setup of this test apparatus.

Material useful for the fuel rod cladding, extension rod, and spacers has been located. Purchase orders for thin- and thick-wall tubing are being initiated. The bar stock for the spacers and end plugs is presently on hand.

Figure 4-1. Core II Instrumented Fuel Rods Installation

Figure 4-2. Plan View of IFR Locations in Reactor Through Head Ports

Figure 4-3. IFR Reactor Penetration Assembly

CORE II
INSTRUMENTED FUEL ASSEMBLY

Figure 4-4. Instrumented Fuel Rod

Figure 4-5. Accuracy of Transient Fuel Temperature Rise

Figure 4-6. Fuel Thermocouple Installation

Table 4-5

FUEL THERMOCOUPLE DESIGN FEATURES

- Junction
 - Bare Wire
 - Stranded
 - Knotted
 - W/3% RE versus W/25% RE
 - Guide Tube
- Transition
 - Ceramic to Metal Seal
 - Fission Gas Containment
- Leadout Wire
 - Compensated Extension Wire
 - Swaged Sheath

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 with 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 experimental data collected during performance of the experimental program at zero power and up to 17.2 MW was completed, except for the balanced oscillator tests. In general, good agreement with prediction was observed. Data taken at power levels up to 17.2 MW in the power ascension phase of the experimental program have indicated that the observed reactivity effects and fuel temperature changes are consistent with a calculated Doppler coefficient $T(\text{dk}/\text{dT})$ of -0.0081 (-0.0085 predicted).

5.3 CURRENT RESULTS**5.3.1 Comparison of Calculated and Experimental Static and Oscillator Test Results****5.3.1.1 Discussion of Results**

Measurements of the reactivity feedback in SEFOR have been performed at power levels up to 20 MW and at average coolant temperatures between 350°F and 760°F . During these measurements the mixed oxide ($\text{PuO}_2\text{-UO}_2$) fuel temperatures ranged between a minimum of 350°F and a calculated maximum of approximately 4800°F . Analysis of the static and time-dependent experimental data has provided a measured power-Doppler reactivity relationship with an estimated standard deviation of $\pm 8\%$. Comparison of the results with calculations demonstrated that the Doppler component of the feedback can be described by a fuel temperature coefficient $T(\text{dk}/\text{dT})$ of -0.0081 with an estimated standard deviation of ± 0.0014 . This coefficient compares favorably with the value* of -0.0085 that was predicted² prior to the measurements.

* This is the estimated prediction for the cores, Assemblies I-I and I-J (the present core), on which these Doppler experiments were performed. The actual prediction, for a core with 19 B_4C rods, was -0.0082 . The predicted effect for a core with 12 B_4C rods was -0.0085 .

Since the SEFOR fuel was designed³ to suppress fuel axial expansion reactivity effects, most (~95%) of the reactivity effects due to fuel temperature changes are a result of Doppler broadening of the nuclear cross sections. All of the experimental results from steady-state and oscillatory tests at power levels up to 20 MW have shown, in agreement with calculation, that the only other significant reactivity effects are those associated with changes in the average core coolant temperature. Measurements to determine reactivity effects other than those associated with the fuel temperature and the average coolant temperature were those performed to determine the reactivity effects associated with changes in the coolant inlet temperature (at a constant power and average coolant) and in the reflector temperatures. The results of the measurements to determine reactivity effects of average coolant temperature changes have been previously⁴ reported. The measurements of reflector temperature effects are discussed below. The coefficients were, in agreement with calculations² nearly zero.

The results of the time-dependent oscillator tests were also used to provide a rough estimate of the Doppler energy coefficient to be expected from sub-prompt transients which will be performed at a later date. Because of the nonconstant fuel thermal conductivity, the temperature-dependent fuel-to-cladding heat transfer coefficient, and the $1/T$ behavior of the Doppler coefficient, these estimates are rather crude, however, and the results obtained from analysis of the fast transients are expected to yield larger reactivity energy coefficients than those estimated from the slowly varying temperature oscillations.

The oscillator, static, and calculated reactivity coefficients agree reasonably well at every power level, and they all demonstrate the same general decreasing magnitude with increasing power level. The magnitude of the coefficient obtained from the oscillator tests is about 10% larger than the static coefficient at low power, and decreases more rapidly with increasing power than the static coefficient, becoming 20% less than the static coefficient at high power. However, the magnitude and trend of the calculated coefficient lies between the coefficients obtained from the two different types of measurements. Within the experimental error, there is overall agreement throughout the complete power range between the experimental and predicted results for a Doppler $T(dk/dT)$ of -0.0081 . Further analysis of the power-reactivity data is planned to determine if fuel restructuring effects can be discerned.

5.3.1.2 Power Reactivity Measurements to 20 MW

The power levels, average coolant temperatures, and reflector positions measured at average coolant temperatures of $\sim 760^\circ\text{F}$ and power levels up to 20 MW during the performance of the recent static tests in the initial power ascension phase of the SEFOR experimental program⁵ are listed in Table 5-1. These tabulated reactivity values have been corrected for the variations in average sodium temperature using measured temperature reactivity 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 $-0.35 - 0.19 (1220)/(1220+65P)$ 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 20 MW. The thermal power levels and average coolant temperatures were calculated using the actual flow and temperature data in Table 5-3.

The reactivity effects shown in Table 5-1 are composed of two effects: the loss in reactivity due to the axial expansion of the fuel and cladding as it is heated, and a loss in reactivity due to Doppler broadening of the nuclear cross sections as the fuel temperature increases. SEFOR was designed³ with segmented fuel rods and dished fuel pellets to reduce the former effect, however, and the dominant component is the Doppler effect.

The experimental Doppler components of the power reactivity effects are listed in Table 5-4 along with total power reactivity effect and the corrections which were applied to the actual experimental data to account for the reactivity effects associated with small changes in core coolant temperature. The reactivity temperature corrections were made with measured⁴ temperature coefficients. The Doppler component of the reactivity effect was obtained by correcting the total reactivity change at a constant coolant temperature for the fuel and fuel-cladding axial expansion effects with a constant reactivity coefficient of -0.5 cent/MW that was obtained from calculations³ based on normalizations to the SEFOR mockup⁶ in ZPR-3. It is important to note that the corrections to the data are small, and that ~95% of the total measured reactivity change is the Doppler effect.

5.3.1.3 Conventional Oscillator Tests at Power

Oscillator experiments were performed at several power levels and coolant flows to demonstrate reactor stability, and to provide data for comparison with the power coefficients determined from the static tests. As expected, the reactor was quite stable, and the experimental reactivity feedback was in good agreement with the model.

Table 5-1
POWER REACTIVITY DATA TO 20 MW

Date (1971)	Reflector Number 3		Reflector Number 8		Average Core* Coolant Temperature (°F)	Thermal Power (MW)	Reactivity Change* at 760°F Coolant Temperature (cents)
	Position (cm)	Worth (cents)	Position (cm)	Worth (cents)			
1/25-2/2		42.40 [†]	0.00	0.00	760.0 [†]	0.0	0.0
1/25	40.00	55.63	38.79	52.65	759.3	5.1	66.2
1/26	40.00	55.63	67.94	104.56	760.4	10.1	117.6
1/26	59.84	91.71	70.00	107.54	761.8	14.7	156.0
1/26	62.17	95.62	70.00	107.54	755.6	15.6	162.8
1/26	64.41	99.33	70.00	107.54	760.5	15.7	164.2
1/26	73.20	111.78	70.00	107.54	756.7	17.7	178.4
1/26	79.98	119.41	70.00	107.54	759.8	18.7	184.6
1/28	40.00	55.63	39.38	53.77	758.0	5.2	68.0
1/28	40.00	55.63	70.00	107.54	764.4	10.0	118.7
1/28	60.83	93.38	70.00	107.54	761.0	14.8	158.1
1/28	74.18	112.95	70.00	107.54	753.2	18.0	181.2
1/29	95.07	129.18	66.70	102.71	754.9	19.8	191.7
1/29	95.07	129.18	67.00	103.16	750.8	19.6	194.0
1/30	95.07	129.18	67.64	104.11	752.1	19.6	194.4
1/30	95.07	129.18	66.96	103.10	751.2	19.6	193.8
1/30	95.07	129.18	67.20	103.46	751.9	19.6	193.8
2/2	40.00	55.63	66.20	100.40	741.6	10.1	122.3
2/2	62.62	96.36	70.00	107.54	756.5	15.1	163.1

*Reactivity temperature corrections made using a reactivity coefficient of $-0.35 \frac{0.19(1220)}{1220+65P} \text{ } \Delta/\text{ }^\circ\text{F}$ at power level P (MW). Estimated deviation in coolant temperature is $\pm 0.25^\circ\text{F}$. The estimated standard deviation in the relative reactivity is $\pm 1.6\%$ or $\pm 3\%$ of the reactivity, whichever is larger. The estimated deviation in absolute power level is $\pm 5\%$.

[†]Average values obtained from Table 5-2.

Table 5-2
LOW POWER (0.06 MW) CRITICAL POINTS
(FEBRUARY 25, 1971–FEBRUARY 2, 1971)

Date	Reflector Number 3		Reactor Coolant Temperature (°F)	Relative Reactivity* at 760°F and Zero Power (cents)
	Position (cm)	Worth (cents)		
1/25/71	30.57	38.11	750	42.43
1/27/71	30.43	37.87	748	43.27
2/02/71	8.52	6.40	693	41.50
Average =				42.40

*Reactivity temperature corrections were made with a coefficient of $-0.54 \Delta/\text{ }^\circ\text{F}$, and a reactivity correction of 1.08 cents to the zero power point was made with a power reactivity coefficient at low power of $-18 \Delta/\text{MW}$. The estimated standard deviation in the relative reactivity is $\pm 1.6\%$. The estimated deviation in temperature is $\pm 2.5^\circ\text{F}$.

**Table 5-3
HEAT BALANCE DATA**

Date (1971)	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)		
1/25	741.2	777.3	3519	745	773.7	240	5.1	759.3
1/26	725.1	795.6	3525	718	788.0	239	10.1	760.4
1/26	711.0	813.1	3539	713	802.2	239	14.7	761.8
1/26	702.2	809.8	3552	700	798.0	237	15.6	755.6
1/26	700.0	821.6	3163	704	809.0	239	15.7	760.5
1/26	703.0	811.1	4061	701	799.3	235	17.7	756.7
1/26	703.0	817.3	4049	704	804.1	236	18.7	759.8
1/28	742.0	775.0	3786	730	770.4	234	5.2	758.0
1/28	732.0	797.4	3742	728	790.0	236	10.0	764.4
1/28	712.9	809.6	3768	714	799.6	235	14.8	761.0
1/28	694.7	812.5	3760	693	799.8	235	18.0	753.2
1/29	705.5	804.8	5001	705	792.5	228	19.8	754.9
1/29	699.7	802.0	4787	707	791.7	229	19.6	750.8
1/30	701.0	803.6	4778	704	792.2	226	19.6	752.1
1/30	700.2	802.5	4784	703	791.2	226	19.6	751.2
1/30	701.1	803.3	4788	701	792.2	229	19.6	751.9
2/2	703.5	774.9	3753	709	767.1	236	10.1	741.5
2/2	707.4	806.2	3771	707	796.2	234	15.1	756.5

⁺To account for observed differences between inlet and outlet coolant temperature readings at zero power, the actual recorded outlet coolant temperatures were increased by 1.2°F.

*Auxiliary inlet temperature readings taken from operations instrumentation.

Table 5-4

DOPPLER POWER REACTIVITY MEASUREMENTS
JANUARY 25, 1971 TO FEBRUARY 2, 1971

Date (1971)	Thermal [†] Power (MW)	Coolant Temperature Reactivity Correction (cents)	Reactivity Change at 760°F Average Coolant Temperature (cents)	Doppler Effect* (cents)
1/25-2/2	0.0	-	0.0	0.0
1/25	5.1	-0.3	-66.2	-63.7
1/26	10.1	+0.2	-117.6	-112.6
1/26	14.7	+0.8	-156.0	-148.7
1/26	15.6	-2.0	-162.8	-155.0
1/26	15.7	-0.2	-164.2	-156.4
1/26	17.7	-1.5	-178.4	-169.6
1/26	18.7	-0.1	-184.6	-175.3
1/25-2/2	0.0	-	0.0	0.0
1/28	5.2	-1.0	-68.0	-65.4
1/28	10.0	+2.1	-118.7	-113.7
1/28	14.8	+0.5	-158.1	-150.7
1/28	18.0	-3.1	-181.2	-172.2
1/29	19.8	-2.2	-191.7	-181.8
1/29	19.6	-4.1	-194.0	-184.2
1/30	19.6	-3.5	-194.4	-184.6
1/30	19.6	-3.9	-193.8	-184.0
1/30	19.6	-3.6	-193.8	-184.0
1/25-2/2	0.0	-	0.0	0.0
2/2	10.1	-8.7	-122.3	-117.3
2/2	15.1	-1.6	-163.1	-155.6

[†]The estimated standard deviation in power is ± 0.1 MW or $\pm 5\%$ of the power, whichever is larger. ⁽⁴⁾

*Obtained from the adjacent column by accounting for the calculated fuel-fuel cladding axial expansion reactivity effect of -0.5¢/MW . The estimated standard deviation in the Doppler effect is $\pm 2\text{¢}$ or $\pm 4\%$ of the effect, whichever is larger.

The conventional oscillator tests were performed by oscillating a boron-carbide rod in the central drywell to produce sinusoidal oscillations in the reactivity, and thus in the neutron flux, fuel temperatures, coolant temperatures, etc. The time-dependent rod position, signals from two (channels 27 and 28) of the operational neutron monitors located below the vessel, the time-dependent fuel and coolant temperature oscillations indicated by the instrumented fuel assemblies (IFA's) and some of the vessel thermocouple signals were recorded on magnetic tape with the computerized data acquisition system. The sampling rate was controlled by the oscillator rod position so that 32 nearly equally spaced (in time) samples were recorded in each cycle of oscillation.

The conventional oscillator tests were performed at zero power, and at nominal power levels of 2 MW, 5 MW, 10 MW, 15 MW, 17 MW, and 19 MW. The zero power oscillator tests results are described in Reference 7 and test results up to 17 MW are reported in References 4 and 8. The measured power and coolant temperature responses at 19 MW and at 15 MW (following 20-MW operation) are shown in Tables 5-5 through 5-8. The nominal frequencies of oscillation are listed in Table 5-9, and the amplitudes of the oscillator rod are listed in Tables 5-10 and 5-11.

The measurements procedures are described in further detail in Reference 8. The theoretical model that was used in analyzing the data is also described in Reference 8.

Table 5-5
POWER REACTIVITY TRANSFER FUNCTION AND
FEEDBACK KERNEL AT 19 MW

Frequency (rad/sec)	Power Reactivity Transfer Function		Feedback Kernel	
	Amplitude (% Power/ Δ Reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
0.0114	0.913	27.57	0.05887	146.6
0.0179	1.031	25.94	0.05174	144.4
0.0309	1.215	25.53	0.04289	138.0
0.0489	1.482	23.89	0.03387	130.2
0.0781	1.798	17.21	0.02559	122.4
0.1283	2.007	3.22	0.01778	114.5
0.2228	1.805	-11.22	0.01103	108.0
0.2241	1.821	-10.85	0.01117	106.2
0.2244	1.862	-11.31	0.01090	102.9
0.3500	1.590	-15.92	0.00712	96.3
0.3506	1.530	-15.75	0.00711	106.6
0.5590	1.349	-15.39	0.00418	92.3
0.6053	1.320	15.49	0.00393	102.8
0.6128	1.324	-15.04	0.00418	100.0
0.9539	1.225	-13.66	0.00239	89.0
0.9648	1.195	-13.45	0.00243	112.7
1.5153	1.130	-11.24	0.00145	104.8
1.5393	1.119	-11.33	0.00136	120.2
1.5622	1.121	-11.29	0.00126	114.4

Table 5-6
 AVERAGE COOLANT TEMPERATURE TO POWER
 TRANSFER FUNCTION AT 19 MW

Frequency (rad/sec)	$\Delta T_c / \Delta P$ ($^{\circ}\text{F}/\text{MW}$)	
	Real	Imaginary
0.0114	1.5	3.8
0.0179	1.2	-2.4
0.0309	1.1	-1.2
0.0489	1.1	-0.6
0.0781	0.1	-0.8
0.1283	0.7	1.0
0.2228	0.2	-1.1
0.2241	0.1	-1.1
0.2244	-0.1	-1.2
0.3500	-0.1	-0.9
0.3506	-0.1	-0.8
0.5990	-0.2	-0.4
0.6053	-0.1	-0.4
0.6128	-0.1	-0.4
0.9539	-0.2	-0.1
0.9648	0.0	1.0
1.5153	0.0	0.0
1.5393	0.0	0.0
1.5622	0.2	0.0

Table 5-7
POWER REACTIVITY TRANSFER FUNCTION AT 15 MW,
75% FLOW, SERIES B

Frequency (rad/sec)	Power Reactivity Transfer Function		Feedback Kernel	
	Amplitude (% Power/ Δ Reactivity)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
0.01133	0.951	28.75	0.07182	145.23
0.01170	0.965	27.67	0.07069	146.03
0.01771	1.087	26.20	0.06237	143.75
0.03058	1.296	27.00	0.05151	135.60
0.03117	1.301	26.81	0.05121	135.51
0.04867	1.593	25.11	0.04056	127.21
0.07763	1.971	16.46	0.02973	119.15
0.07845	1.973	16.13	0.02955	119.13
0.2168	1.861	-12.97	0.01284	105.50
0.2191	1.839	-13.04	0.01279	106.67
0.3445	1.566	-16.91	0.00828	101.52
0.6039	1.337	-16.07	0.00458	94.94
0.6155	1.322	-15.89	0.00451	99.38
0.9685	1.205	-13.81	0.00265	103.59
0.9828	1.198	-13.41	0.00294	106.93
0.9839	1.197	-13.56	0.00278	107.85
1.5503	1.130	-11.43	0.00147	97.41
1.5686	1.118	-11.04	0.00185	115.80

Table 5-8
AVERAGE COOLANT TEMPERATURE TO POWER
TRANSFER FUNCTION AT 15 MW,
75% FLOW, SERIES B

Frequency (rad/sec)	$\Delta T_c / \Delta P$ ($^{\circ}F/MW$)	
	Real	Imaginary
0.01133	1.0	-4.3
0.01170	1.2	-4.3
0.01771	0.8	-2.4
0.03058	1.0	-1.3
0.03117	1.0	-1.1
0.04867	1.1	-0.9
0.07763	1.0	-1.1
0.07845	1.0	-1.1
0.2168	0.2	-0.9
0.2191	0.2	-0.8
0.3445	-0.1	-0.6
0.6039	-0.1	-0.3
0.6155	-0.2	-0.3
0.9685	-0.2	-0.1
0.9828	-0.2	-0.1
0.9839	-0.2	-0.1
1.5503	0.0	0.0
1.5686	0.0	0.0

Table 5-9
NOMINAL FREQUENCIES OF OSCILLATION

Designation	Radians/Second	Hertz
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 5-10
OSCILLATOR AMPLITUDES AND RECORDS
19 MW

Tape Number 94

Frequency	Rod Amplitude (Inches)	File Number
F2	1.00	1
F3	0.86	2
F4	0.80	3
F5	0.65	4
F6	0.58	5
F2	1.20	6
F3	1.20	7
F4	1.20	8
F5	0.85	9
F6	0.85	10
S1	0.85	11
S2	0.85	12
S3	0.85	13
S4	1.25	14
S5	1.25	15
S6	1.25	16
F2	1.20	17
F4	1.20	18
F6	0.85	19

Table 5-11
OSCILLATOR AMPLITUDES AND RECORDS
15 MW, 75% FLOW, SERIES B

Tape Number 115

Frequency	Rod Amplitude (Inches)	File Number
F2	1.40	12
F3	1.40	13
F4	1.40	14
F5	1.40	15
F6	1.40	16
S2	1.40	18
S3	1.40	19
S4	1.40	20
S5	1.40	21
S6	1.40	22
F2	1.40	23
F3	1.40	24
F4	1.40	25
F6	1.40	26
S2	1.40	27
S4	1.40	28
S6	1.40	29
F3	1.40	30

5.3.1.4 Integral Doppler Reactivity Effects

The data in Table 5-4 demonstrate the dependence of the more recent measured Doppler reactivity changes on the reactor power level. Earlier results to 17.2 MW are reported in Reference 4. 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 Reference 4 were used in the WEDOP⁹ 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 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 $T(dk/dT)$). Since previous analyses⁷ 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 $T(dk/dT)$ of -0.0081 that were obtained from the WEDOP⁹ calculations are compared with the measured Doppler reactivity effects in Figure 5-1. It is seen that the agreement between measured and calculated effects as a function of reactor power is quite good.

5.3.1.5 Doppler Power Reactivity Coefficients

The Doppler components of the power reactivity coefficients obtained from analysis of the static and oscillator tests are summarized and compared with calculations in Table 5-12. The tabulated static test values were obtained from fits to the static test data in Table 5-4 and in Reference 4, while the oscillator values were obtained from the conventional oscillator data in Tables 5-5 to 5-8, and in References 4 and 8 using the models discussed in Reference 8. The calculated coefficients were obtained from results of the WEDOP⁹ code.

Figure 5-1. Doppler Power Reactivity Measurements

Table 5-12

DOPPLER POWER REACTIVITY COEFFICIENTS

Power Level Power Level (MW)	Oscillator Tests (d/MW)	Doppler Coefficient	
		Static Tests (d/MW)	Calculated $T dk/dT = -0.0081$ (d/MW)
0	—	-16.8 ± 1.7	—
2	-15.8 ± 1.3	-14.4 ± 1.2	-15.5
5	-10.9 ± 0.9	-11.6 ± 0.8	-11.1
10	-8.3 ± 0.7	-8.5 ± 0.6	-7.6
15	-5.7 ± 0.7	-6.5 ± 0.6	-6.0
17	-5.0 ± 0.6	-5.9 ± 0.6	-5.6
19	-4.2 ± 0.6	-5.4 ± 0.7	-5.1
20	—	-5.2 ± 0.8	-5.0

5.3.1.6 Fuel Time Constants

An effective core average fuel time constant is determined from analysis of the feedback reactivity measured during the oscillator tests. In addition, a local time constant appropriate to a fuel rod central temperature may be estimated from analysis of the measured fuel temperature oscillations. Since these time constants are definable in terms of fuel rod heat transfer characteristics, they may be independently calculated, and compared to the measured values.

To facilitate the calculations of these parameters from the nonlinear equations involving the temperature dependent heat transfer characteristics, they were related to calculated fuel temperatures as described below. These relations allow the determination of the time constants from temperatures that are readily calculated from the heat conduction equations.

A local fuel time constant, $\tau(u, r)$, at a radial position u within a fuel rod located at core location r , may be defined as follows:

$$\tau(u, r) = \frac{\rho C(u, r) R^2}{2h(r) R} + \frac{\rho C(u, r) R^2}{4 K(u, r)} [1 - u^2/R^2]$$

In this expression ρ is the fuel density (assumed constant), $C(u, r)$ is the fuel heat capacity [which is dependent on the local fuel temperature at location (u, r)], R is the fuel radius, and $h(r)$ is the effective heat transfer coefficient between the fuel surface temperature and the coolant temperature at location r . The effective fuel thermal conductivity, $K(u, r)$, is formally defined in terms of the temperature-dependent local thermal conductivity, $\lambda(\theta)$, the fuel surface temperature, $T_s(r)$, and the local fuel temperature, $T(u, r)$, by the following relation:

$$K(u, r) = \int_{T_s(r)}^{T(u, r)} \lambda(\theta) d\theta / [T(u, r) - T_s(r)]$$

The fuel surface temperature, $T_s(r)$, is related to the coolant temperature, $T_c(r)$, and the total reactor power, P , by the conventional steady-state relation:

$$T_s(r) = T_c(r) + f \frac{\rho}{M} P \frac{q(r) R^2}{2h(r) R}$$

where f is the fraction of the total power which is absorbed in the fuel, M is the total fuel ($\text{PuO}_2\text{-UO}_2$) mass, $q(r)$ is the normalized (to a core average value of 1.0) power distribution, and the other parameters have been previously defined.

Using the above relations, the local fuel temperature, $T(u,r)$, obtained from the steady-state heat conduction equations may be written as follows:

$$T(u,r) - T_c(r) = \frac{f P q(r)}{MC(u,r)} \tau(u,r)$$

Accordingly, the local central fuel time constant, $\tau(O,r)$, estimated from steady-state conditions may be written as:

$$\begin{aligned} \tau(O,r) &= \frac{\rho C(O,r) R^2}{2h(r) R} + \frac{\rho C(O,r) R^2}{4 K(O,r)} \\ &= \frac{MC(O,r)}{f P q(r)} [T(O,r) - T_c(r)] \end{aligned}$$

The second relation is the one which is used to determine calculated values of the central fuel time constants from calculated fuel temperatures.

In an analogous manner, the core average fuel temperature, $\bar{T}$, may be used to approximate the calculation of the core average fuel time constant, $\bar{\tau}$, thusly:

$$\bar{\tau} = \frac{\bar{MC}}{fP} [\bar{T} - \bar{T}_c]$$

where $\bar{C}$ is the core average fuel heat capacity, and $\bar{T}_c$ is the core average coolant temperature.

The above expression was used to calculate the fuel time constants at different reactor power levels. The core fuel loading was 2120 kg of $\text{PuO}_2\text{-UO}_2$. The expression for the heat capacity that was used in the calculation is $C = 0.165 + 1.31 \times 10^{-6} (T - 700)$ Joules/g-°F where T is the fuel temperature in °F. The "effective" average fuel temperature rise, $\bar{T}$, for use in the expression for $\bar{\tau}$ was defined so that $-0.0081 \ln [1 + \bar{T}/1220]$ was equal to the reactivity change calculated using the WEDOP⁹ code. At the higher power levels, the calculated effective temperature rise was about 3% larger than the calculated actual core average fuel temperature rise. The calculated and measured average fuel time constants are tabulated in Table 5-13. As the results in the table indicate, the calculated and measured times constants agree within about 10%.

Table 5-13
AVERAGE FUEL TIME CONSTANTS

Power (MW)	Fuel Time Constants	
	Oscillator	
	Tests (sec)	Calculated (sec)
2	30.1 ± 2.2	32.6
5	28.0 ± 2.2	31.6
10	28.7 ± 2.2	29.0
15	28.0 ± 2.2	27.5
17	26.7 ± 2.2	27.0
19	24.5 ± 2.2	26.5

5.3.1.7 Doppler Energy Coefficient

At low power levels the transient Doppler energy coefficient may be estimated using an approximation to a more exact formula. The approximate formula provides a simple relationship between the energy coefficient and parameters that are determined from analysis of the oscillator test data.

The coefficients obtained using calculated coefficients in the simple approximation agree with those obtained using measured parameters; thus demonstrating consistency between measured parameters and calculational models. The approximation itself appears to be valid only at low power levels, however, and, as will be shown below, the more detailed model is required for determining the energy coefficients at the higher power levels.

The model for the determination of the Doppler energy coefficient and the approximation to this model may be derived through the following considerations:

When the local absolute fuel temperature, $T(u,r)$, defined in the previous section, is changed from an initial value of $T_o(u,r)$ to a new value $T_o(u,r) + \Delta T(u,r)$, the local change in reactivity, $\Delta K_D(u,r)$, resulting from the local Doppler effect, may be written as:

$$\Delta K_D(u,r) = A \ln \left[1 + \frac{\Delta T(u,r)}{T_o(u,r)} \right]$$

where A is the value of the Doppler reactivity coefficient $T(dk/dT)$.

For small temperature changes the Doppler effect may be approximated as:

$$\Delta K_D(u,r) \simeq A \Delta T(u,r)/T_o(u,r)$$

The total change in reactivity for the entire core is the weighted sum of the local contributions. The total change may thus be expressed as:

$$\bar{\Delta K}_D \cong \int W(r) d^3r \left(\frac{2u du}{R^2} \right) [\Delta K_D(u,r)]$$

where $W(r)$ is a normalized spatial weighting function, and the other parameters have been previously defined.

When the energy deposition in the fuel occurs during a time interval that is short compared to the fuel time constant (e.g., during transients), little heat is lost by conduction to the coolant, and the local change in fuel temperature, for small temperature changes, may be approximated by $\Delta T(u,r) \cong \Delta E q(r)/MC(u,r)$ where ΔE is the total energy deposited in all the fuel during the transient.

The local Doppler energy coefficient $\eta(u,r)$ may be expressed as

$$\eta(u,r) = \frac{\Delta K_D(u,r)}{\Delta E} = \frac{A q(r)}{MC(u,r) T_o(u,r)}$$

In a similar manner, for small changes ΔP in total reactor power level at a constant coolant temperature, the local Doppler power coefficient $\alpha_D(u,r)$ is

$$\alpha_D(u,r) = \frac{\Delta K_D(u,r)}{\Delta P} \cong \frac{A f q(r) \tau(u,r)}{MC(u,r) T_o(u,r)}$$

From these relations it is seen that

$$\eta(u,r) = \frac{\alpha_D(u,r)}{f \tau(u,r)}$$

The core average Doppler energy coefficient, $\bar{\eta}$, is defined as $\bar{\Delta K}_D/\Delta E$, while the core average Doppler power coefficient, α_D is defined as $\bar{\Delta K}_D/\Delta P$. From the above relations it is seen that

$$\bar{\eta} = \frac{1}{f} \langle \alpha_D/\tau \rangle$$

where $\langle \alpha_D/\tau \rangle$ denotes the core average of the indicated ratio.

In general, $\bar{\eta} \neq \bar{\alpha}_D/(\bar{f}\tau)$. However, it can be shown by direct substitution into the above equations, that at low power levels, where the fuel and coolant temperature (and thus the fuel rod heat transfer characteristics) are relatively constant across a fuel rod and across the core, the relation $\bar{\eta} = \bar{\alpha}_D/(\bar{f}\tau)$ is valid.

The differences between the core average Doppler energy coefficients obtained from direct calculations using the WEDOP⁹ program, and those calculated using the approximate formula given above are illustrated in Table 5-14. It is seen that they agree at the low power levels, as pointed out above, but diverge as the power level is increased.

Also shown in Table 5-14 are the Doppler energy coefficients estimated by using the Doppler power coefficients and fuel time constants that were experimentally determined from the analysis of the oscillator test data. The agreement values obtained from the approximate formula is, of course, a result of the previously noted agreement between calculated and measured values of the power coefficient and the fuel time constants; thus demonstrating consistency between the calculational model and the experimental results.

Table 5-14
DOPPLER ENERGY COEFFICIENTS

Calculated From Definition* (d/MW-sec)	Calculated From Average Values (d/MW-sec)	Estimated From Oscillator Tests (d/MW-sec)
-0.54	-0.51	-0.56 ± 0.06
-0.45	-0.38	-0.41 ± 0.05
-0.37	-0.28	-0.30 ± 0.03
-0.33	-0.24	-0.22 ± 0.03
-0.31	-0.22	-0.20 ± 0.02
-0.29	-0.20	-0.18 ± 0.03

*See text for a discussion of the differences.

5.3.1.8 Uncertainties in the Doppler Temperature Coefficient

The overall estimated standard deviation in the measured Doppler reactivity effects as a function of reactor power level is ±8%. This deviation is composed of the ±4% standard deviation⁸ of the Doppler component of the static reactivity measurements, an estimated ±5% uncertainty⁸ in the absolute value of the indicated power levels, and one-half (±5%) of the indicated (see Table 5-12) average deviation (±10%) in the power reactivity coefficients. Only a portion of the deviations in the power coefficients was included, because part of these deviations are attributable to power and reactivity deviations that are incorporated (above) into the other components of the overall uncertainty. The ±8% deviation is the square root of the sum of the squares of the above individual components (i.e., random deviations are assumed).

The overall estimated standard deviation in the Doppler temperature coefficient $T(dk/dT)$ is ±17%. One component of this deviation is the ±8% deviation in the power-Doppler reactivity relationship. The other component, and the primary contributor to the deviation, is an estimated ±15% uncertainty in the calculated relationship between power level and effective core average fuel temperature, resulting from an estimated ±15% deviation in the fuel thermal conductivity and the fuel-to-cladding heat transfer coefficient. Since the model for the fuel-cladding gap coefficient was normalized⁴ to the fuel temperature measurements using the best estimate for the fuel thermal conductivity, these two

parameters are correlated, and the deviation was estimated for the combined effect rather than from individual contributions from each parameter, calculations indicate the uncertainty would have been of the order of $\pm 18\%$ (instead of $\pm 15\%$) if the fuel temperature measurements had not been available.

Although the generally good early agreement between calculated and measured fuel temperatures in the central instrumented fuel assembly would indicate that the uncertainty in the fuel temperature-power relationship might be as small as $\pm 10\%$ or less, the large deviation between measured and calculated temperatures in the peripheral instrumented fuel assembly at power levels above 5 MW would appear to negate such a reduction in the uncertainty.

The contributions of other deviations, such as a $\pm 3\%$ deviation in local power density³ and a $\pm 2\%$ variation in pellet density (allowed by fuel specification) are small compared to the $\pm 15\%$ uncertainty in the fuel thermal conductivity and may be neglected. The various contributions to the total estimated standard deviation are summarized in Table 5-15.

5.3.2 In-Core Temperature Measurements

5.3.2.1 Fuel Temperature Measurements

Fuel temperatures have been measured at different core locations at power levels up to 20 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-cladding heat transfer coefficient to provide agreement between measured and calculated temperatures. Since the adjusted fuel-cladding gap coefficient agreed reasonably well with the calculated value, the normalized model was then used to predict fuel temperatures at higher power levels.

Only one fuel-cladding 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 (up to 17 MW) is discussed in Reference 4.

The fuel temperatures were measured using W/Re thermocouples (T/C's) located along the centerline of four fuel rods in two instrumented fuel assemblies (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 centerline, and one near the periphery. Each IFA contains an "inner fuel rod" and an "outer fuel rod," with the inner rod being that closest to the core centerline.

The measured fuel temperature changes with power during early 1971 are tabulated in Table 5-16. Except for T/C 20, which continues, as previously noted,⁴ to indicate higher and higher temperature readings, all thermocouple readings have decreased (see the earlier data in Reference 4) over a period of time.

None of the thermocouples have recorded a temperature higher than $\sim 3000^\circ\text{F}$ to $\sim 3500^\circ\text{F}$, even though calculations indicate local fuel temperatures at some of the T/C locations may have reached $\sim 4000^\circ\text{F}$, and all thermocouples have shown a deterioration in performance over a period of time. These effects may have been caused by the thermal cycling, a high temperature irradiation, or both. In any event, the practical upper limit for these T/C's appear to be $\sim 3000^\circ\text{F}$.

5.3.3 Reflector Temperature Reactivity Coefficients

5.3.3.1 Discussion of Results

The SEFOR reflector temperature coefficient of reactivity was measured at power levels of 5, 10, 15, and 20 MW. The maximum observed change in reactivity at any power level due to temperature changes* in excess of 100°F was 1.1 cents. At all power levels the reactivity appeared to decrease slightly with increasing reflector temperature. The magnitudes of the reactivity changes were less than the estimated standard deviation (± 1.6 cents) in reactivity changes at power, however, and for all practical purposes the reflector temperature coefficient of reactivity may be treated as zero.

5.3.3.2 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

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

Table 5-15

STANDARD DEVIATION IN DOPPLER COEFFICIENT

Parameter	Effect on Doppler Coefficient
Absolute Reactivity Measurement	±4%
Absolute Power Level Measurement	±5%
Deviations in Power Coefficients	±5%
Total Power Reactivity Deviation	±8%
Uncertainty in Fuel Thermal Conductivity and Fuel-to-Cladding Heat Transfer	±15%
Uncertainty in Local Power Density	±3%
Variations in Fuel Pellet Density	±2%
Uncertainties in Coolant and Cladding Heat Transfer Properties	0†
Total Power Fuel Temperature Deviation	±15%
Total Deviation	±17%

† Large uncertainties in the parameters have little influence on fuel temperatures.

Table 5-16

IFA FUEL TEMPERATURES TO 20 MW

Date (1971)	Thermal Power (MW)	Fuel Temperature Rise Above Sodium Coolant Temperature of ~760°F			
		Central IFA		Peripheral IFA	
		Inner Fuel Rod Upper T/C No. 1 (°F)	Lower T/C No. 2 (°F)	Outer Fuel Rod Lower T/C No. 4 (°F)	Outer Fuel Rod Lower T/C No. 20 (°F)
1/25	5.1	365	310	599	864
1/26	10.1	764	655	1243	1579
1/26	14.7	1194	1008	1845	2099
1/26	15.6	1255	1078	1952	2186
1/26	15.7	1270	1097	1977	2201
1/26	17.7	1479	1253	2167	2363
1/26	18.7	1599	1307	2218	2451
1/28	5.2	258	313	641	860
1/28	10.0	572	655	1301	1559
1/28	14.8	871	1015	1717	2080
1/28	18.0	1304	1268	1715	2373
1/29	19.6	1587	1307	2797	2092
1/30	19.6	1397	1286	2656	2151
1/30	19.6	1367	1285	2577	2152
1/30	19.6	1206	1277	2487	2182
2/2	10.1	536	657	1456	1660
2/2	15.1	736	1013	2023	2196

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 or 2 parts in 500, or less, 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 were less than the estimated deviation (± 0.1 MW) in flux level and reactivity corrections for these variations were neglected.

The averages of data taken at low and high equilibrium reflector temperature readings are shown in Table 5-17. The measurements at 5, 10, and 15 MW were reported earlier,⁸ but are repeated here for comparison purposes. The data in each set are averages of 3 to 5 successive measurements with the data acquisition system static test program.*

The reflector temperature readings shown in Table 5-17 were taken from two thermocouples located near the center of coarse reflector 6. 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 5-17 indicate that the core reactivity decreases very slightly with increasing reflector temperature, and that the effect is about -0.01 cent/°F or less when the temperature is monitored by these particular thermocouples. The measured reactivity changes are less than the estimated standard deviation (± 1.6 cents) in reactivity measurements at power, however, and since the normal changes in reflector temperature between zero power and full power are $\sim 200^\circ\text{F}$ or less, the reflector temperature coefficient may be treated as zero.

5.3.4 Balanced Oscillator Tests

Balanced oscillator tests of the first and second kind (BOE I and BOE II) were performed at nominal power levels of 10, 13, 15, 17, and 19 MW. In these tests the reactivity and flow are simultaneously oscillated and their relative amplitude and phase are adjusted in an attempt to maintain either the average coolant temperature constant (this is BOE I), or the power level constant (this is BOE II). Harmonic analysis of the data has been completed, and the BOE I transfer function data are presented in Tables 5-18 through 5-24. The data shown in the tables are (1) measured power/reactivity (%power/cent); (2) inferred total feedback reactivity/power (cent/MW); and (3) measured average core coolant temperature/power (°F/MW). The BOE II data are still being analyzed, and will be reported in a subsequent progress report.

Preliminary analysis of the BOE I data shows that the overall power coefficient (cent/MW), after correction for slight coolant temperature oscillations is approximately 10% larger in magnitude than that derived from the conventional oscillator tests. Further analysis planned to investigate possible causes for these small discrepancies.

* The data for reflector thermocouple No. 60 were taken from Test Plan Procedure II-8 data sheets for all three sets of measurements at 5 MW.

Table 5-17

REFLECTOR TEMPERATURE-REACTIVITY MEASUREMENTS

Time (1970-1971)	Nominal Power (MW)	Data Set Number	Average Coolant Temperature (°F)	Reflector Number 3 Position (cm)	Worth (cents)	Reflector Thermocouple Number 56 (°F)	Reflector Thermocouple Number 60 (°F)	Relative Reactivity at 760°F* (cents)
7/9-1400	5	1	760.4	42.20	- 59.76	152	130	+0.10
7/10-0250	5	3	759.9	42.17	- 59.71	149	128	- 0.10
7/10-0035	5	2	760.8	42.94	- 61.15	290	252	1.08
8/14-1615	10	1	760.9	70.00	-107.75	215	181	- 0.44
8/14-1920	10	2	761.7	69.85	-107.54	199	167	+0.16
8/14-2115	10	3	761.8	69.91	-107.62	231	194	+0.12
8/15-0450	10	5	761.6	69.80	-107.48	197	164	+0.17
8/15-0230	10	4	761.7	70.50	108.40	322	273	- 0.70
8/29-1400	15	2	759.8	50.96	- 76.19	257	205	0.0
8/29-1055	15	1	759.5	51.42	- 77.12	479	395	- 1.07
8/29-1740	15	3	759.5	51.46	- 77.11	441	362	- 1.06
1/29-2220	20	1	750.8	67.00†	103.16†	309	240	- 0.14
1/30-0640	20	3	751.2	66.96†	-103.10†	315	247	+0.10
1/30-1050	20	4	751.9	67.20†	-103.46†	308	241	+0.05
1/30-0450	20	2	752.1	67.64†	-104.11†	472	377	- 0.52

* Reactivity corrections to a 760°F average coolant temperature were calculated using reactivity coefficients of -0.50, -0.47, -0.46, and -0.44 cents/°F at 5, 10, 15, and 20 MW respectively.

The relative values were adjusted at each power level so that the average of the values at the cooler reflector temperatures is zero.

† Values for Reflector Number 8.

Table 5-18
First Balanced Oscillator Data
10 MW - 560° F - BOE I

Frequency (Rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel		$\Delta \bar{T}_c / \Delta P$ (°F/MW)	
	Amplitude (%Power/Cents)	Phase (degrees)	Amplitude (Cents/MW)	Phase (degrees)	Real	Imaginary
0.01111	0.941	16.74	10.651	157.19	1.41	-1.85
0.01120	0.934	19.10	10.777	154.86	0.41	-3.30
0.01734	1.163	21.37	8.629	147.92	-0.55	-1.32
0.03060	1.325	23.87	7.438	138.20	0.00	0.00
0.03085	1.324	23.77	7.438	138.22	0.00	0.00
0.04902	1.721	23.20	5.575	126.63	0.00	0.00
0.07854	2.102	13.91	4.079	117.83	0.00	0.00
0.07870	2.203	14.40	0.397	115.07	0.00	0.00
0.07872	2.089	20.52	4.504	113.09	-0.52	-1.32
0.07873	2.180	13.38	3.936	116.35	-0.60	-0.30
0.1309	2.224	-4.71	2.589	109.61	0.00	0.00
0.2197	1.882	15.74	1.665	102.51	0.08	-0.36
0.2234	1.848	15.89	1.641	104.51	0.07	-0.31

Table 5-19
First Balanced Oscillator Data
~10 MW - 760° F - BOE I

Frequency (Rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel		$\Delta \bar{T}_c / \Delta P$ (°F/MW)	
	Amplitude (%Power/Cents)	Phase (degrees)	Amplitude (Cents/MW)	Phase (degrees)	Real	Imaginary
0.01124	1.084	21.23	9.471	152.03	-0.40	-2.4
0.01774	1.229	21.23	8.258	147.36	0.00	0.00
0.01781	1.200	25.84	8.537	142.90	0.00	0.00
0.04928	1.858	22.05	5.156	125.26	0.00	0.00
0.07659	2.245	12.37	3.801	116.39	0.00	0.00
0.07759	2.311	10.96	3.675	116.32	0.00	0.00
0.1291	2.247	-7.81	2.365	112.38	0.00	0.00
0.1295	2.192	-7.32	2.427	113.21	0.00	0.00
0.1296	2.192	-7.90	2.376	113.56	0.00	0.00
0.2029	1.890	-16.56	1.638	108.87	0.00	0.00
0.2031	1.892	-16.45	1.632	108.55	0.00	0.00
0.2222	1.865	-17.29	1.511	105.33	0.00	0.00

Table 5-20
First Balanced Oscillator Data
~13 MW - 760°F - BOE I

Frequency (Rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel		$\Delta \bar{T}_c / \Delta P$ (°F/MW)	
	Amplitude (%Power/Cents)	Phase (degrees)	Amplitude (Cents/MW)	Phase (degrees)	Real	Imaginary
0.01112	0.998	18.78	7.761	154.8	0.50	-0.71
0.01130	0.980	32.57	8.106	141.4	0.02	-3.8
0.01742	1.083	24.86	7.194	145.2	-0.03	-1.55
0.03032	1.356	25.83	5.660	136.0	-0.43	-0.09
0.04878	1.642	22.68	4.456	128.6	0.48	0.37
0.07821	2.070	12.94	3.130	119.5	0.23	-0.29
0.07826	2.032	11.64	3.115	121.4	0.36	0.60
0.1305	2.160	-4.52	2.032	112.5	-0.34	0.19
0.1307	2.168	-4.65	2.020	112.1	-0.37	0.06
0.2337	1.746	-16.22	1.229	112.0	0.05	-0.20

Table 5-21
First Balanced Oscillator Data
~15 MW - 760°F - BOE I

Frequency (Rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel		$\Delta \bar{T}_c / \Delta P$ (°F/MW)	
	Amplitude (%Power/Cents)	Phase (degrees)	Amplitude (Cents/MW)	Phase (degrees)	Real	Imaginary
0.07725	1.899	12.29	2.892	124.5	0.85	1.00
0.07741	1.937	13.23	2.881	122.7	0.37	0.54
0.07743	1.942	12.89	2.861	122.0	0.34	0.52
0.07745	1.940	12.98	2.867	122.9	0.43	0.60
0.07751	1.947	13.19	2.867	122.5	0.42	0.43
0.1282	2.137	1.64	1.920	112.6	0.0	0.0
0.1998	1.888	-12.45	1.333	108.8	0.0	0.0
0.1999	1.884	-12.34	1.340	109.0	0.0	0.0
0.2005	1.876	-12.35	1.341	109.5	0.0	0.0

Table 5-22
First Balanced Oscillator Data
~15 MW - 760°F - BOE I (continued)

Frequency (Rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel		$\Delta \bar{T}_c / \Delta P$ (°F/MW)	
	Amplitude (%Power/Cents)	Phase (degrees)	Amplitude (Cents/MW)	Phase (degrees)	Real	Imaginary
0.01096	0.853	19.18	7.864	155.4	1.75	-1.59
0.01099	0.965	19.27	6.959	154.6	-0.88	-1.71
0.01104	0.924	34.86	7.462	139.6	-0.42	-5.85
0.01104	0.927	19.00	7.242	155.1	0.80	-1.26
0.01115	0.966	27.78	7.060	146.2	0.44	-3.03
0.01132	0.920	25.96	7.382	148.2	-0.55	-3.28
0.01764	1.098	26.57	6.181	143.3	-1.24	-2.45
0.01766	1.089	27.23	6.245	142.8	-1.37	-2.66
0.01770	1.061	27.33	6.402	142.9	-0.83	-1.54
0.01770	1.055	26.62	6.405	144.6	-0.58	-1.16
0.01772	1.110	29.46	6.169	140.4	-1.11	-1.88
0.01774	1.088	27.50	6.253	142.5	-0.62	-1.77
0.01775	1.075	27.03	6.316	143.0	-0.85	-1.60
0.01781	1.110	27.86	6.138	141.9	-0.95	-1.72
0.03042	1.280	25.66	5.177	137.1	0.14	-0.26
0.03057	1.288	27.76	5.200	135.0	-0.45	0.95
0.03057	1.253	26.23	5.299	136.9	0.0	0.0
0.03062	1.246	26.03	5.322	137.2	0.16	-0.46
0.03066	1.244	25.02	5.300	138.2	0.68	-0.31
0.03067	1.247	27.25	5.347	136.0	-0.27	-0.53
0.04858	1.602	25.45	4.046	126.8	0.0	0.0
0.07667	1.993	16.37	2.943	119.0	0.0	0.0
0.07722	1.940	11.61	2.811	124.1	0.22	1.15

Table 5-23
First Balanced Oscillator Data
~17 MW - 760°F - BOE I

Frequency (Rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel		$\Delta \bar{T}_c / \Delta P$ (°F/MW)	
	Amplitude (%Power/Cents)	Phase (degrees)	Amplitude (Cents/MW)	Phase (degrees)	Real	Imaginary
0.01086	0.860	20.69	6.462	153.5	0.40	-1.83
0.01162	0.943	20.35	7.120	153.4	0.35	-1.78
0.01715	0.928	25.11	5.813	145.6	0.40	-1.82
0.03019	1.100	24.87	4.796	138.9	0.0	0.0
0.03112	1.239	25.64	4.706	137.5	0.0	0.0
0.03117	1.302	23.92	4.443	138.3	0.0	0.0
0.03120	1.282	24.52	4.527	138.0	0.0	0.0
0.04939	1.558	24.56	3.632	128.1	0.0	0.0
0.0774	1.907	15.20	2.649	121.7	0.65	0.20
0.07795	1.964	9.31	2.338	124.6	-0.76	0.99
0.1291	2.034	-0.96	1.779	116.1	0.0	0.0
0.2150	1.805	-11.37	1.226	107.4	0.0	0.0
0.2280	1.815	-13.03	1.122	106.0	0.11	-0.42

Table 5-24
First Balanced Oscillator Data
~19 MW - 760°F - BOE I

Frequency (Rad/sec)	Power-Reactivity Transfer Function		Feedback Kernel		$\Delta \bar{T}_c / \Delta P$ (°F/MW)	
	Amplitude (%Power/Cents)	Phase (degrees)	Amplitude (Cents/MW)	Phase (degrees)	Real	Imaginary
0.01114	0.9910	28.17	5.616	145.65	0.54	-1.60
0.01125	0.9194	28.44	6.044	145.77	0.79	-1.25
0.01763	1.033	22.43	5.282	147.96	0.45	-0.55
0.03060	1.207	23.40	4.412	140.26	0.0	-0.30
0.03080	1.213	23.35	4.386	140.16	0.0	0.0
0.04897	1.499	22.29	3.415	131.42	0.0	0.0
0.04904	1.493	22.09	3.422	131.69	-0.25	0.0
0.07784	1.898	16.67	2.506	120.62	0.0	0.0
0.07781	1.871	16.93	2.545	121.04	0.0	0.0
0.07786	1.919	16.64	2.481	120.13	0.0	0.0
0.07831	1.894	16.22	2.494	120.97	0.0	0.0
0.1296	2.005	0.35	1.715	116.25	0.0	0.0
0.1297	1.998	0.80	1.737	116.15	0.0	0.0
0.2255	1.796	-12.26	1.082	108.29	0.0	0.0
0.2306	1.773	-12.35	1.075	108.90	0.0	0.0

6. TASK 11 - TRANSIENT TESTS

6.1 TRANSIENT TESTS RESULTS

Transients involving rapid insertion of positive reactivities up to 50 cents (the familiarization transients) have been completed. A preliminary analysis of the flux data recorded during the transient was performed to determine the time-dependent reactivity. This "dynamic" reactivity agreed well with the results which had been predicted from the previous measurements of FRED worth vs position, and position vs time.

6.2 DYNAMIC CALIBRATION

The time-dependent reactivity, $k(t)$, was determined from the following relation:

$$k(t) = 1 + (k_0 - 1) \frac{P_0}{P(t)} + \left(\frac{\ell}{\beta}\right) \frac{1}{P(t)} \frac{dP}{dt} - \frac{1}{P(t)} \int_0^t [P(u) - P_0] \sum_i \sum_j a_{ij} \lambda_{ij} e^{-\lambda_{ij}(t-u)} du$$

where P is the flux, P_0 in the initial steady-state value of the flux at time zero, (ℓ/β) is the ratio of the prompt neutron lifetime to the effective delayed neutron fraction, a_{ij} is the fraction of delayed neutrons in delay group i from fissionable isotope j , and λ_{ij} is decay constants for neutrons in delay group i from isotope j .

The power, P , and the derivative dP/dt in the above relation were obtained from the raw flux data with a smoothing routine similar to that in Reference 10.

The time-dependent behavior of the flux, and the time-dependent reactivity obtained from preliminary analysis of the signal from one of the boron-lined detectors below the core during a 50 cent transient from an initial power level of 10 MW are shown in Figures 6-1 and 6-2.

The agreement between the time-dependent reactivity in Figure 6-2 with that determined from the "static" calibration (see below) indicates that the point reactor kinetic model provides an adequate description of the SEFOR transient behavior.

6.3 STATIC CALIBRATION

The reactivity worth of the FRED familiarization slug was measured as a function of axial position. These measurements were combined with position vs time measurements to generate the reactivity worth as a function of time. This "statically" determined relationship will be compared, at a later date, with the "dynamic" worth vs time data determined from the power trace during an actual transient.

The worth vs position measurements were made with a calibrated fine reflector as the rod was lowered by increments into the core, and they were repeated as the rod was withdrawn. Static equilibrium was attained after each change in the FRED position. The measurements were performed at low power with a constant coolant temperature of 751°F.

The actual worth vs position data are tabulated in Table 6-1. The two measurements at each axial position were averaged to give the worth vs position data tabulated in Table 6-2 and plotted in Figure 6-3. Averaged position vs time data for a FRED test firing with an accumulator pressure of 1385 psi (at or near maximum pressure) are illustrated in Figure 6-4.

The FRED slug can be fired from any initial position in the core. The distance vs time characteristics are independent of the initial position. The reactivity change as a function of FRED slug displacement is dependent on the starting slug position, however, and the total reactivity change for a firing of the familiarization slug can vary between a maximum of 49.5 cents for a starting position of zero, to a minimum of ~ 0 cent for a starting position ≥ 25 inches.

The data in Figures 6-3 and 6-4 were used to generate FRED worth vs time curves for three initial positions: 0.0 inch, 3.0 inches, and 4.0 inches. The 0- and 4-inch starting positions correspond to the nominal 50-cent and 40-cent transients called for in the experimental program, while the 3-inch position corresponds to the expected starting position for the 96-cent and larger transients that will be performed at a later date with larger worth slugs.

The FRED reactivity worth - time characteristics for these three different initial starting conditions are illustrated in Figures 6-5, 6-6, and 6-7. The maximum reactivity inserted, and the maximum rate of insertion are tabulated on the curves.

The maximum reactivity insertion rate for 96-cent, 118-cent, and 130-cent slugs is estimated to be 13.2 dollar/sec, 16.3 dollar/sec, and 18.0 dollar/sec, respectively. These three values were calculated by multiplying the insertion rate at the 3-inch starting position (this is the expected location for these slugs) with the ratio of the slug worth in question (96 cents, 118 cents, or 130 cents) to the maximum reactivity (42.7 cents) inserted by the familiarization slug when fired from the 3-inch position.

Table 6-1

FAMILIARIZATION SLUG REACTIVITY WORTH MEASUREMENTS

FRED Position*	Reflector No. 3		Reflector No. 8		Relative† FRED Worth
	Position (cm)	Worth (cents)	Position (cm)	Worth (cents)	
29.69	34.38	45.06	0.41	-0.43	0.27
24.69	34.38	45.06	1.03	0.01	0.71
19.64	34.38	45.06	5.20	3.33	4.03
17.64	34.38	45.06	8.08	6.01	6.71
15.64	34.38	45.06	11.70	9.85	10.55
13.64	34.38	45.06	15.46	14.36	15.06
11.64	34.38	45.06	19.60	20.01	20.71
9.64	34.38	45.06	22.77	24.74	25.44
7.64	34.38	45.06	26.00	29.82	30.52
5.64	34.38	45.06	29.14	35.03	35.73
3.64	34.38	45.06	32.12	40.29	40.99
1.64	34.38	45.06	34.84	45.26	45.96
0.01	34.38	45.06	36.72	48.75	49.45
1.64	34.38	45.06	34.87	45.31	46.01
3.64	34.38	45.06	32.15	40.34	41.04
5.64	34.38	45.06	29.25	35.21	35.91
7.64	34.38	45.06	26.15	30.06	30.76
9.64	34.38	45.06	22.88	24.91	25.61
11.64	34.38	45.06	19.27	19.53	20.23
13.64	34.38	45.06	15.59	14.53	15.23
15.64	34.38	45.06	11.59	9.72	10.42
17.64	34.38	45.06	7.84	5.78	6.48
19.64	34.38	45.06	4.71	2.90	3.60
24.69	34.38	45.06	0.41	-0.43	0.27
29.69	34.09	44.52	0.41	-0.43	-0.27

*Indicated position above the core midplane.

At an indicated position of 0.0 inch, the lower tip of the B₄C column in this slug is estimated to be 6.0 inches below the core midplane.

†Relative to an average value at the completely out position (29.69 inches) of zero.

Table 6-2

FAMILIARIZATION SLUG WORTH VERSUS POSITION

FRED Position* (inches)	Average Measured† Worth (cents)
0.01	0.00
1.64	3.47
3.64	8.44
5.64	13.63
7.64	18.81
9.64	23.93
11.64	28.98
13.64	34.31
15.64	38.97
17.64	42.86
19.64	45.64
24.69	48.96
29.69	49.45

*Indicated position above the core position.

At an indicated position of 0.0 inch the lower tip of the B₄C column in this slug is estimated to be 6.0 inches below the core midplane.

†Relative to zero at the completely inserted position (0.01 inch).

Average values obtained from last column of previous table.

Figure 6-1. Flux Data From Wide Range Monitor No. 2 for 50-Cent Transient From 10 MW

Figure 6-2. Reactivity Data From Wide Range Monitor No. 2 for 50-Cent Transient From 10 MW

Figure 6-3. FRED Familiarization Slug

Figure 6-4. FRED Rod Transit Time

Figure 6-5. 49.5-Cent Slug-Reactivity Worth as a Function of Time

Figure 6-6. 42.7-Cent Slug--Reactivity Worth as a Function of Time

Figure 6-7. 40.5-Cent Slug—Reactivity Worth as a Function of Time

7. SEFOR SAFETY AUDIT AND LICENSING

An audit trip was made to SEFOR during February 16-19 to review checkout testing of the FRED, and to review the recently approved Change No. 2 to the Technical Specifications with site personnel.

Final copies of the SEFOR Technical Specifications pages affected by Changes No. 2 and No. 3 were received from the DRL. Change No. 2 (which completed action on Proposed Changes 3 and 4) authorizes the use of guinea pig rods under the innermost refueling ports during steady-state operation up to 20 MWt and during excursion testing including the maximum planned transient. It also provides additional specifications to detect anomalous performance of the reactor. Change No. 3 completed action on the requirements for personnel qualifications submitted in Proposed Change No. 2. Approval of the change in pump-around flow requested in Proposed Change No. 2 has not yet been obtained.

The following Modification Requests were reviewed and approved:

No. 687—Lift-Off and Proximity Switches to the DAS

No. 690—SEFOR Excursion Mode Circuit Modifications

No. 666—Three Phase Volt-Pack Neutral Legs

No. 669—Sodium Level Switch Fuse Relocation

Malfunction Reports were sent to the DRL regarding:

1. Malfunction of K5 Relay (initiates scram signal for high temperature indication in the core outlet upper region) which occurred during a surveillance test,
2. Failure of the instrument nitrogen line to the operator for the valve in the reactor vessel overflow line,
3. Overcurrent damage to safety system relays (provides control logic for the cover gas supply valve) which occurred during a surveillance test.

Reports required by the Technical Specifications entitled, "Summary of Results from Testing of the Fast Reactivity Excursion Device" and "Natural Circulation Testing of the SEFOR Reactor Coolant System" were issued.

A response to three items shown on a Form 592 concerning increased pretension values used for the reactor vessel head bolts, low heat transfer coefficient for the auxiliary IHX, and FRED modifications was sent to the AEC.

An increase in reactor vessel cover gas leakage was noted and found to be related to the tension in the outer head bolts. Analysis and measurements demonstrated that the problem was caused by rotation and displacement of the reactor vessel flange relative to the reactor vessel head. The rotation and displacement was caused by temperature gradients resulting from changes in the reactor vessel sodium temperature. The analysis that was done and the proposed corrective action were discussed with the DRL in meetings on March 25, April 27 and 28, 1971. It was determined that the outer head bolt preload should be reduced from the previous stress level of approximately 8500 psi to 4000 psi to keep the stress levels below code-allowable values during all expected transients. In addition, the reactor vessel sodium temperature transient rates will be limited to 10°F/hr for changes greater than 125°F, and yearly surveillance will be done on a sample of head bolts. The Technical Specifications will be changed to incorporate the temperature transient limits, bolt preload, and yearly surveillance.

8. SEFOR SUPPORT

The plant engineering effort is a supporting function to the SEFOR Facility. During February and March, the familiarization transients were completed. During the latter half of March and April, reactor operation was suspended while investigation of the reactor head bolt condition was completed.

8.1 REACTOR VESSEL HEAD BOLTS

A test and evaluation program concerning the reactor head bolts was completed during the report period. The reactor remained shut down for the period March 15, 1971, through April 28, 1971, while diagnostic tests and analyses were completed to measure the head bolt stresses and reactor flange rotations. As a result of the tests, examinations, and analyses, the bolt tension was adjusted to a lower value, and the bolt stresses were determined to be acceptable for the intended service. Preparations for resumption of reactor operation was initiated on April 29, 1971.

The reactor vessel (Figure 8-1) contains an inner and outer head which are held with bolts and sealed by metal O-rings. The inner head is bolted to the annular outer head by means of 24 bolts. The outer head is bolted to the reactor flange with 48 bolts (see Figure 8-2).

During the initial period of reactor testing following initiation of the power ascension, an increase in reactor cover gas leakage was noted. An investigation into the cause of this leakage was made during the November 1970 outage, and it was discovered that gas leakage past the outer head seal was occurring. Sodium vapor had been carried with this gas

Figure 8-1. SEFOR Reactor Vessel

Figure 8-2. Reactor Head Bolt

and had condensed on top of the outer head beneath the insulation. This sodium was removed, and the tension in the outer head bolts was checked and found to be lower than expected, and the bolts were then retensioned. In order to investigate the possible causes of cover gas leakage, dial indicators and thermocouples were installed to measure any flange movement and temperatures during a change in sodium temperature at the normal heatup rate.

Measurement by dial indicators indicated the vessel flange was rotating with sodium temperature changes (see Figure 8-3). Subsequent analysis indicated the rotation was increasing the stresses in the bolts, reducing their tension, and allowing gas leakage to occur.

As a result of the analyses and measurements, it was concluded that flange rotation resulting from system temperature changes of more than 200°F at a rate of 20°F/h could result in excessive bolt stresses.

The bolt tension was rechecked during February 1971, and the bolt tension was found to be low, which provided an indirect verification of the hypothetical bolt strain conditions. The bolts were then retensioned to the proper value and the allowable heatup/cooldown rate was reduced to 10°F/h.

Subsequently, the reactor was operated over several thermal cycles, and then to provide a direct verification of the condition of the head bolts, three bolts were removed and examined. Direct visual, dye penetrant, and ultrasonic examinations were performed. The results of these examinations did not reveal any deleterious condition in the bolts.

In order to provide a direct measurement of bolt stresses, strain gages were applied to the bolts as shown in Figure 8-4. The bolts were installed in the head, and direct strain indication was recorded over a full temperature cycle, including heatup and cooldown at 10°F/h and simulated scram.

As a result of these tests and further analyses, it was concluded that a reduction in bolt pre-tension loading would provide the necessary reduction in bolt stresses to maintain the stresses below limits for the normal steady-state operation and transient conditions. The bolts were retensioned to a level less than 4,000 psi and subsequently preparations for reactor operation were begun on April 29, 1971.

8.2 REACTOR VESSEL HEAD SEAL

In connection with the reduction in reactor head bolt loadings, design work was initiated on a vessel head seal. The seal is designed as a backup to prevent reactor cover gas from leaking past the outer head O-ring seal. The seal will be installed should cover gas leakage become excessive. The seal provides two major sealing functions:

1. A backup seal is made between the outer head surface and the reactor vessel skirt. The seal made with an elastomer O-ring retained by a backing strip.
2. A backup seal is provided by a cover located over each of the 48 outer head bolts. The cover is sealed by an elastomer O-ring. The seal is designed such that the seal components can be replaced, should it be required.

8.3 CRANE RETRIEVAL

Methods of retrieving refueling cell crane loads in the event of system malfunctions or failures have been under development, and the design of retrieval equipment has been completed which allows efficient recovery in the event of major unforeseen equipment malfunctions.

The retrieval equipment is designed for the following functions:

1. **Bridge E-W movement, Figure 8-5).** Bridge motion is obtained through the use of a cable hoist which is employed to move the bridge. The cable is permanently installed along the bridge rail, and ports are installed at the end of the cell to provide penetrations for the cable. Bridge movement is achieved by actuating a cable hoist fixed externally to the end of the cell.
2. **Trolley Retrieval (N-S movement, Figure 8-6).** Trolley motion is achieved through a push/pull motion of a cable which is looped over a hook supplied on either end of the carriage. Existing penetrations may be used to insert the cables as required.
3. **Ten-Ton Hoist.** In the event the reactor head is attached to the load block at the time of malfunction, the head may be lowered manually by means of a chain hoist permanently installed as a part of the lifting sling. This operation required man access to the refueling cell. N-S and E-W movement is achieved as described above.

In the event that a lighter load is attached to the 10-ton hoist, vertical motion is obtained through the use of a "fish line" cable. The line is installed through either a horizontal or vertical port in the refueling cell (see Figure 8-7). The port for the fish line may be installed in preselected locations, or core drilling may be accomplished when the need arises. The fish line is threaded through the open port, passes through a guide ring attached to the trolley and latches onto the grapple. The load may then be raised by the cable as required.

Figure 8-3. Flange and Head Rotations

Figure 8-4. SEFOR Head Bolts—Strain Gages

Figure 8-5A. SEFOR Support—Bridge Retrieval System, Refueling Cell Plan View

Figure 8-5B. SEFOR Support-DETAIL-Bridge Emergency Retrieval System

GEAP-10010-18

Figure 8-6. SEFOR Support—Plan View Refueling Cell

4. **One-Half-Ton Hoist (Figure 8-7).** Loads attached to the 1/2-ton hoist are handled in a similar fashion as described for the 10-ton hoist, using the "fish line" concept installed through a port. Guide rings and latching locations are provided on the trolley and grapple, respectively.

8.4 FRED ROD ASSEMBLY MODIFICATION

The FRED (fast reactivity excursion devise) rod was modified to permit remote assembly and disassembly within the refueling cell. Assembly/disassembly is performed by means of pinned connections, utilizing the modified fuel rod disassembly station.

8.5 ARGON GAS METER

The requirements of an argon gas use-rate meter were established, and an integrating type of meter chosen to monitor the usage of argon supplied to reactor cover gas.

9.0 SEFOR OPERATION

9.1 EXPERIMENTAL PROGRAM

With the receipt of AEC-DRL approval of proposed changes PC-3 and PC-4 to the Technical Specifications on January 22, testing above 17.5 MWt was undertaken. The Group II (Power Ascension) and Group III (Static and Balanced Oscillator) tests that remained, were integrated into one sequence. This sequence is shown in Table 9-1.

With guinea pig fuel remaining in the inner through head port locations, the high flux trip level was reduced to 101.5% of full power (20.3 MW) to limit the maximum linear power density to 25.2 kW/ft. This resulted in extremely delicate maneuvering of reactor power in performance of the oscillator tests at 19 MW because the amplitude of oscillation closely approached the trip level. It also resulted in delicate maneuvering to attain full power due to the proximity of the overpower trip. In order not to compromise the experiment and still not scram the reactor, the mean power level of the balanced oscillator test at the nominal 19 MWt point was reduced about 0.5 MW.

The testing and power ascension were performed with little difficulty. Preliminary evaluation of the plant operating characteristics compare favorably with those predicted by the design and revealed no major problems. The Wide Range Monitors (WRM's) were calibrated at 19 MW because of the very restrictive overpower limit. The phase margin of the point at which the Nyquist diagram intercepts the unit circle for 19 MW was 80 degrees, indicating a very stable system.

Fuel surveillance after 20 MW operation showed only a small evidence of fuel element change associated with full power operation. These data are tabulated in Table 9-2.

Table 9-1
REVISED POWER ASCENSION TEST SEQUENCE

Power Level (MWt)	Test	Main Primary Flow Rate (gpm)
18	Static	4065
19	Static	3750
	Static	4750
	Conventional Oscillation	4750
	Balanced Oscillation	
	Type I	3750
	Type II	3750
20	Static	~5000
	Reflector Temperature	
	Coefficient	~5000
	Plant Heat Balance	~5000
	Nuclear Instrument Calibration	~5000
	Radiation Surveillance	~5000
Shutdown	Fuel Surveillance	
15	Static	3000
	Conventional Oscillation	3000

Table 9-2

POST 20 MW OPERATION FUEL SURVEILLANCE RESULTS

RESULTS

	Allowable	Actual			
		A16	A02	806	807
Pellet Column Length Change (in.)	1/2	+1/16	+1/16	+1/16	+1/16
Cladding Diameter Change (in.)	0.010	+0.001	+0.001	None	None
Cladding Rupture or Observable Defects	None	None	None	None	None
Bowing (in.)	Less than an amount preventing reinsertion	0.046 T.I.R.	0.016 T.I.R.	0.006 T.I.R.	0.020 T.I.R.

Results of all examinations to date on these rods are:

Rod A16

	Preirradiation	5 MW	15 MW	17.5 MW	20 MW
Length (in.)	-0.007	-0.012	+0.040	+0.030	+0.033
Temperature Corrected Length	-0.007		+0.008	-0.002	-0.012
Diameter Spiral Trace (in.)	+0.0030	+0.0040	+0.0045	+0.0045	+0.0055
Temperature Corrected Diameter	+0.0030		+0.0035	+0.0035	+0.0040
Bow Spiral (in.)	0.008 T.I.R.	0.043 T.I.R.	0.046 T.I.R.	0.056 T.I.R.	0.045 T.I.R.
Bottom Pellet Column Length (in.)	21-1/2	21-1/2	21-1/2	21-7/16	21-1/2
Top Pellet Column Length (in.)	13-15/16	13-15/16	13-15/16	13-15/16	14

Table 9-2 (Continued)

Rod A02

	Preirradiation	20 MW
Length (in.)	-0.008	+0.059
Temperature Corrected Length	-0.008	+0.009
Diameter Spiral Trace (in.)	+0.0030	+0.055
Temperature Corrected Diameter	+0.0030	+0.0040
Bow Spiral (in.)	0.016 T.I.R.	0.016 T.I.R.
Bottom Pellet Column Length (in.)	21-3/8	21-7/16
Top Pellet Column Length (in.)	13-15/16	13-15/16

Rod 806

	Preirradiation	20 MW
Length (in.)	-0.001	+0.071
Temperature Corrected Length	-0.001	+0.020
Diameter Spiral Trace (in.)	+0.0035	+0.0055
Temperature Corrected Diameter	+0.0035	+0.0040
Bow Spiral (in.)	0.016 T.I.R.	0.006 T.I.R.
Bottom Pellet Column Length (in.)	13-15/16	14
Top Pellet Column Length (in.)	21-3/8	21-3/8

Table 9-2 (Continued)

Rod 807		
	Preirradiation	20 MW
Length (in.)	-0.006	+0.031
Temperature Corrected Length	-0.006	-0.009
Diameter Spiral Trace (in.)	+0.0035	+0.0055
Temperature Corrected Diameter	+0.0035	+0.0040
Bow Spiral (in.)	0.019 T.I.R.	0.020 T.I.R.
Bottom Pellet Column Length (in.)	21-7/16	21-1/2
Top Pellet Column Length (in.)	14	14

General Comments to Table 9-2

Fuel rod temperatures in-cell one day after shutdown were about 225°F at the core midplane position due to fission product decay. Sodium distillation is improved, apparently due to the slightly higher fuel rod temperature during distillation. With the chamber at 700°F, the fuel rod surface is about 800-825°F.

The reported 1-mil increase in diameter may not be real, but it is a general impression from the appearance of the overall recorder trace. There is no way of repeatedly measuring the same point on the fuel rod. Also, the readout of the reference diameter (machine spindle) now is 0.0003 to 0.0006 inch higher than most of the early measurements. Therefore, the 0.001-inch reported diameter change is uncertain but is considered to be negligible. Both emission and transmission gamma scans results were good. Insulator pellets are still indicated distinctly on the transmission gamma scan, providing a very good measurement of pellet column length. Activity peaks are present at the pellet interfaces over the full length of the guinea pig rods. On the standard rods, activity from pellets near the end of the rods still show valleys at the pellet interfaces. On the guinea pig rods, pellet interfaces in the peak flux region are less distinct, and more random activity peaks occur. This is interpreted to be indications of pellet cracking and/or fuel relocation.

Spectrum measurements at the center of pellets and at the interfaces were repeated on two rods. There is no detectable difference (except in magnitude) in the spectrum at the two points. Only rod A02 was mass spectrometer leak checked due to failure of the diffusion pump. However, the chamber exhaust line was monitored for gamma activity during distillation of all rods. No leaks were detected.

Comments, Rod A02

The top pellet in the lower stack is significantly higher in activity than other fuel pellets, in both the emission and transmission gamma scans. The emission gamma scans made during the low Pu investigation showed this pellet to be normal in enrichment.

The anomaly possibly can be explained by pellet cracking reducing its gamma self-shielding.

At the conclusion of operation at the 15 MWt power ascension repeat point, a short period of power operation was completed to gather data to better understand and define reactor cover gas activity. When these cover gas activity tests were complete, the reactor was shut down and preparations for transient tests began.

Transient test preparation involved modification to the reactor safety system, checkout of the assembly-disassembly station which had been modified for remote assembly-disassembly of the FRED slugs; and further tests of the transient detector system. In May 1970, FRED system was checked both in-cell and out-of-cell. At that time a grounding incompatibility between the FRED lift-off switches and the data acquisition system (DAS) was found. In the interim, this problem had been corrected. Prior to moving FRED into the refueling cell some adjustments and measurements were made on the lift-off switches and the distance detector, then FRED was installed on the reactor head. Over forty test firings were made with the nominal 50-cent familiarization slug and FRED at the "secured" position (slug out of the core). These tests were performed to gather in situ firing statistics, provide the final readiness testing prior to transient tests and check the lift-off modification. Comparison shows that slug ejection times compare well with early preoperation tests done in May of 1970 at SEFOR and are very repeatable.

During the final firings of FRED, the hose connecting the accumulator and the expansion chamber was inadvertently snared on the junction box of the through-head Na level probes when the positioner was moved to verify and adjust the limit switches. This resulted in deformation of the hose and the firings done in this condition were noticeably slower. The hose was replaced and several additional firings were made to confirm the same performance. Also, during this sequence of performance tests, one lift-off switch malfunctioned and required the disassembly of FRED to repair the lift-off switch. A similar sequence of about 10 firings demonstrated that the characteristics remained unchanged.

As a part of test preparation, redundant signals for the two lift-off switches and both the 10-inch and 20-inch travel proximity switches were wired into the DAS. These vital signals are obtained, as well, on the visicorder (high speed oscillograph).

The pretest FRED firings performed in-cell were analyzed in detail. The 20-inch ejection time for a nominal 2000 psig transfer chamber pressure was about 3 milliseconds slower for the SEFOR tests when compared to a selection of early FRED development tests. The results of the early development tests were obtained by reducing the data taken on the first 50 FRED development tests. This data reduction was performed at SEFOR which should preclude one potential systematic data reduction error. The difference between the two sets of data is not considered to be significant because both sets of data exhibited about the same repeatability and agree within two standard deviations. SEFOR composite data had an approximate ± 1.6 millisecond standard deviation while the FRED development data had a reported standard deviation of ± 1.5 milliseconds.¹¹

To provide test standards to assess the proper functioning of FRED during each test, the pretest data were reviewed and the tests for which the transfer chamber was most carefully pressurized to 2000 psig were selected. These tests were then analyzed to determine the mean ejection time from liftoff switch No. 2 to the closure of proximity No. 2 (20 inches of travel). These values were:

Accumulative Pressure Range (psig)	Mean Travel Time for 20-Inch Travel (milliseconds)	Standard Deviation (milliseconds)
1360 1410	105.6	± 1.2

For checking the performance of FRED after each test, an ejection time band was established within which the determined ejection time must fall. This band was 102.0 - 109.2 milliseconds (± 3 sigma). If the ejection time is outside this interval, it will be considered to be the indicator of some incipient problem and will require explanation before proceeding with the next test.

The transient detector system pretest checkout was completed on the amplifiers and installed detectors. Frequency response curves were completed on the detector system. The tests were performed by taking the signal cable leads loose at the penetration into the reflector drive cell and adding a section of identical cabling to account for the cabling from the penetration to the detector inside the nitrogen zone. A signal generator was used to input a signal at the mock detector terminal and the output was determined on an oscilloscope connected at the data acquisition system (DAS) patch panel. The response data is tabulated in Table 9-3.

During the operation to 20 MW, the transient detector system which was comprised at that time of two U-238 fission chambers and the SEFOR energy probes (SEP's) was monitored and power-calibrated. Careful evaluation of the detector checkout and calibration revealed that with the present amplifier gains, the output signal into the data acquisition system (DAS) would be unacceptably small in a DAS 0-to-4-volt channel on the linear ranges.

A Sanborn, model 8875A, differential amplifier was added on the output of the transient detector amplifiers to provide a larger flexibility in gain selection. Gains of 3, 10, and 30 may be selected and allow a better output level into

Table 9-3

**TRANSIENT DETECTOR SYSTEM
FREQUENCY RESPONSE TESTS**

Gain: 100V/mA

Transient Detector Amplifier No. 1

Frequency (Hz)	Output—Relative Units		
	Input Amplitude (Amperes)	10^{-6} – 10^{-5}	10^{-7} – 10^{-6}
10		5.0	5.0
100		5.0	5.0
500		5.0	5.0
1K		5.0	5.0
5K		5.0	4.9
10K		4.9	4.6
20K		4.7	3.8
30K		4.4	3.0
40K		4.0	2.5
50K		3.8	2.0
60K		3.5	1.8
70K		3.4	—
80K		3.3	1.5
90K		3.1	—
100K		3.0	1.4
200K		2.8	1.3

Transient Detector Amplifier No.2

Gain: * 100 V/mA

Frequency (Hz)	Output—Relative Units	
	Input Amplitude (Amperes)	10^{-7} – 10^{-5}
10		5.0
100		5.0
1K		5.0
5K		5.0
10K		4.4
20K		3.0
30K		2.0
40K		1.2
50K		0.9
60K		0.7
80K		0.5
100K		0.3

Table 9-3 (Continued)

Transient Detector Amplifier No. 3

Gain: * 100 V/mA

Frequency (Hz)	Output—Relative Units	
	Input Amplitude (Amperes)	
	10^{-7} — 10^{-6}	10^{-6} — 10^{-5}
10	5.0	5.0
100	5.0	5.0
1K	5.0	5.0
5K	4.8	5.0
10K	4.2	4.9
20K	3.0	4.5
30K	2.0	4.1
40K	1.6	3.6
50K	1.2	3.2
60K	1.0	2.8
80K	0.7	2.2
100K	0.5	1.9

*Note: Data at gain settings of 1 V/mA and 10 V/mA were obtained, but not included since all tests on the linear range will be performed at maximum gain.

the DAS 0- to 4-volt channels. Frequency response characteristics with the Sanborn amplifiers in place were checked. The high speed Sanborn amplifier has no apparent effect on the frequency response.

The transient amplifiers exhibit a zero drift on the linear ranges. The drift is believed to be due predominantly to temperature sensitivity of the amplifier. To avoid uncertainties from zero drifting at low input levels, the power levels on the nominal 0.1-MW tests were increased to 1 MW and additional cooling has been added to the amplifier cabinet.

During the familiarization transient, a high noise content was evident on both fission chamber channels which will be referred to in subsequent discussions as Channel A and Channel B. The chamber for Channel A was located at $342^{\circ}30'$ relative to reactor 0° (North) and Channel B was located at $162^{\circ}30'$. Both channels had a background of low amplitude, few-kilocycle noise which, although contributed to the untidy appearance of the signal, was not a problem in the interpretation of the data. Channel A, in addition to the high frequency noise component, had a relatively large amplitude (approximately 200 mV) 200 Hz oscillations which also appeared to "beat." Channel B had a component of what appeared to be 60-Hz noise which was not detected until the test data were reduced and analyzed. Systematic diagnostic testing has consistently pointed to the origin of the noise on Channel A (200 Hz) to be inside the nitrogen zone. The excitation source appears to be mechanically induced vibration caused by the nitrogen blowers. The fission chambers appear to be "microphonically" sensitive and that possibly the 200 Hz is the natural frequency of the signal electrode. Diagnostic testing indicated that four of the five fission chambers exhibit the characteristic 200 Hz, ostensibly, mechanically induced noise. The apparent 60-Hz noise on Channel B may result from the same microphonic source—fabrication variation could result in a different natural frequency. (Subsequent examination of one chamber indicated that an improved internal support for the electrodes might eliminate this noise problem. This modification is underway on one chamber.) However, the frequency is so close to normal plant ac that induced electromagnetic noise is the obvious suspect. To preclude ac-induced noise, the cable leads for the fission chambers have been encased in grounded conduit inside the reflector drive cell to the penetration and will be similarly shielded from the exterior penetrations to the amplifiers.

During this period, the transient gamma chambers were completed and installed and will be calibrated against the fission chambers during the sub-prompt transient tests.

As a part of the transient test preparation, the safety system was modified, in the intervening period, to include (a) a permissive that requires the flow to be above 4300 gpm to perform transient tests; and (b) scram delay of nominal 400 milliseconds. The scram delay time was specified to be 370 ± 20 milliseconds and is in addition to the normal safety system reaction time. The scram signal is initiated by the FRED fire switch. Taking into account the respective relay delays and function delays in the safety system, it appears that about 330 milliseconds exist from the time FRED begins to move out of the core until the reflectors begin motion from the scram signal. This was substantiated by flux traces made during the familiarization transients. The total elapsed time from the initiation of scram signal until reflector motion begins in the "excursion" mode is a nominal 460 milliseconds.

When testing of the safety system with the reactor "secured" was successfully completed, the final check was completed by establishing reactor power at 5 MWt and initiating a scram by the FRED firing switch (FRED was not armed) and the flux, operation of the scram contactors and programmed primary flow reduction were followed. This test was performed as a provisional test procedure and confirmed the proper operation of the safety system.

The preliminary stage of the familiarization transient testing began by calibrating the signals from the transient detector system against power up to 17 MW and back to zero power. When the detector calibration was completed, the familiarization slug was calibrated and the worth of the slug was determined acceptable. After the slug calibration and prior to the first transient test, the mechanical checkout of FRED (FRED firing with reactor "secured") was performed. One of the galvanometers on the Visicorder had failed and was replaced. The mechanical checkout was repeated successfully.

The first familiarization transient test, 40-cent reactivity ejection from an initial power level of 1 MW, was completed on February 28, 1971. Immediately after the test, the reactor was secured and a cooldown for fuel surveillance was initiated. Preliminary onsite data analysis showed that the transient detector signals were not recorded. This problem was traced to the DAS. In order to disable the crossover between low sensitivity (gamma) and high sensitivity (fission) chambers, the inputs to the different amplifiers for Channels A and B (low sensitivity) were disconnected from their analog inputs and grounded. The high or low sensitivity channels may be selected by program control. This function is used during calibration in order to disable the switch-over and provide a compulsory logic level to monitor the high or low sensitivity channels only. Two "flip-flops" determine high or low sensitivity which are set or reset by a differentiator. A noise spike had apparently triggered the "flip-flop" just prior to test initiation which caused the DAS to monitor the low sensitive (gamma) channels. The low sensitive chamber, of course, is not installed and the signal lead had been intentionally grounded. As a result, no fast transient detector data were obtained for familiarization Transient Test No. 1. The power response was followed, however, on two of the three wide range monitors (WRM) which are also recorded by the DAS.

This DAS noise fault has been eliminated by requiring the "flip-flop" to be triggered by the logic signal itself rather than the pulses' leading edge so that noise cannot "set" the control. Also a permissive light has been added so that the DAS operator can visually interrogate the channel to determine if they are ready just prior to the test. This modification has proven to be adequate in all subsequent tests.

Fuel surveillances have been performed after Tests No. 1 and No. 5 and then at the conclusion of the familiarization tests. The results of the fuel surveillances performed during and following the familiarization transient tests are summarized in Table 9-4. In general, no deleterious effects to the fuel as a result of any of the testing to date has been detected. The transmission and emission scans of guinea pig rods have shown no appreciable increase in either the upper or lower fuel column length. The emission gamma scans have lost the activity spikes at the pellet interfaces in the central portion of the rods as noted on the earlier power ascension emission scans. This is believed to be due to the short lived activity of the volatile fission products postulated to be the cause of these spikes.¹³ The general activity of the fuel rods is considerably reduced due to the lower power levels and short durations at these levels which has allowed decay of the earlier deposited activity. A further item of interest is noted on the transmission gamma scans of the guinea pig rods. Pellet definition, by transmission gamma scan, in the central region of guinea pig rods has been lost. Individual pellets at the ends of the rods, however, can still be clearly discerned. In the standard fuel rods, the pellets are distinct over the entire length of rod by transmission gamma scanning.

Table 9-4

FAMILIARIZATION TRANSIENT TEST FUEL SURVEILLANCES

Rod A02							
		Preirradiation	20 MW	10 MW, 0.4\$			
Length (in.)		-0.008	+0.059	-0.005			
Temperature Corrected Length		-0.008	+0.009	-0.030			
Diameter Spiral Trace (in.)		+0.0030	+0.0055	+0.0042			
Temperature Corrected Diameter		+0.0030	+0.0040	+0.0036			
Bow Spiral (in.)		0.016 T.I.R.	0.016 T.I.R.	0.036 T.I.R.			
Bottom Pellet Column Length (in.)		21-3/8	21-7/16	21-3/8			
Top Pellet Column Length (in.)		13-15/16	13-15/16	13-15/16			
Rod A16							
		Preirradiation	5 MW	15 MW	17.5 MW	20 MW	1 MW, 0.4\$ 10 MW, 0.5\$
Length (in.)		-0.007	-0.012	+0.040	+0.030	+0.033	+0.002 +0.013
Temperature Corrected Length		-0.007	----	+0.008	-0.002	-0.012	-0.023 -0.009
Diameter Spiral Trace (in.)		+0.0030	+0.0040	+0.0045	+0.0045	+0.0055	+0.0042 +0.0041
Temperature Corrected Diameter		+0.0030	----	+0.0035	+0.0035	+0.040	+0.0035 +0.0036
Bow Spiral (in.)		0.008 T.I.R.	0.043 T.I.R.	0.046 T.I.R.	0.036 T.I.R.	0.046 T.I.R.	0.042 T.I.R. 0.042 T.I.R.
Bottom Pellet Column Length (in.)		21-1/2	21-1/2	21-1/2	21-7/16	21-1/2	21-7/16 21-7/16
Top Pellet Column Length (in.)		13-15/16	13-15/16	13-15/16	13-15/16	14	14 13-15/16

Table 9-4 (Continued)

Rod 806		Preirradiation	20 MW	1 MW, 0.4\$	10 MW, 0.5\$
Length (in.)		-0.001	+0.071	+0.034	+0.068
Temperature Corrected Length		-0.001	+0.020	+0.004	-0.016
Diameter Spiral Trace (in.)		+0.0035	+0.0055	+0.0047	+0.0045
Temperature Corrected Diameter		+0.0035	+0.0040	+0.0039	+0.0038
Bow Spiral (in.)		0.016 T.I.R.	0.006 T.I.R.	0.006 T.I.R.	0.008 T.I.R.
Bottom Pellet Column Length (in.)		21 3/8	21 3/8	21 7/16	21 7/16
Top Pellet Column Length (in.)		13-15/16	14	14	14-1/16
Rod 807		Preirradiation	20 MW	10 MW, 0.4\$	
Length (in.)		0.006	+0.031	+0.18	
Temperature Corrected Length		-0.006	-0.009	-0.006	
Diameter Spiral Trace (in.)		+0.0035	+0.0055	+0.0048	
Temperature Corrected Diameter		+0.0035	+0.0040	+0.0042	
Bow Spiral (in.)		0.019 T.I.R.	0.020 T.I.R.	0.027 T.I.R.	
Bottom Pellet Column Length (in.)		21 7/16	21 1/2	21 7/16	
Top Pellet Column Length (in.)		14	14	14	

General Comments to Table 9-4

The emission gamma scans did not have the "spikes" occurring at the pellet interfaces which occurred previously in surveillance following high power operation.

The transmission gamma scan on rod 807 showed every pellet interface very distinctly. On rod A02, the pellet interfaces could only be seen on the lower six pellets in the bottom stack and the uppermost 10 pellets in the top stack.

The "hot" pellet at the top of the bottom stack in rod A02, reported following 20 MW surveillance, no longer stands out from the adjacent pellet. Apparently a crack was contributing to the high activity measured earlier.

Helium mass spectrometer leak checks were omitted because the mass spectrometer is still inoperable.

After fuel surveillance, Tests 2, 3, 4, and 5 were completed. Test 2 was a duplicate 40-cent slug test from a nominal 1 MWt power level. Tests 3 and 4 were duplicate 49-cent slug tests from a nominal 1 MWt power level. Test 5 was a 40-cent slug test from a nominal 10 MWt power level.

Evaluation of transient fission detector performance indicated an inordinate amount of high frequency noise in both detectors, as noted earlier. Diagnostic testing to this point established that the noise was endemic to the cabling and/or detectors inside the nitrogen zone. In order to provide as well conditioned a signal as possible to check out data reduction and analysis, filters were installed on the two transient detector channels for the remainder of the familiarization tests. A 5-kHz filter was installed on one channel and a simple approximate 100-Hz filter was installed on the channel which exhibited the 200-Hz noise.

Familiarization Tests 6, 7, and 8 were completed. All of these tests commenced from an initial power level of 10 MWt. On Test 6, both channels were in the linear mode. For Tests 7 and 8, one channel was in the linear mode and one channel was in the log mode to check out log mode data collection which will be necessary in later tests. Table 9-5 summarizes the static slug worth measurements made just prior to each transient test.

Table 9-5
TRANSIENT TEST SLUG WORTH MEASUREMENTS

Test Number	Initial Power Level (MWt)	Nominal Slug Worth (cents)	Measured Static Slug Worth At Critical (cents)	At Power (cents)
1	1.0	40	39.5	39.5
2	1.0	40	39.7	40.0
3	1.0	50	49.1	48.5
4	1.0	50	48.9	48.9
5	10.0	40	39.9	39.5
6	10.0	40	----	39.5
7	10.0	50	----	49.0
8	10.0	50	----	49.6

In conjunction with the fuel surveillance after Test 5, the pilot irradiation for the fission rate measurement utilizing track-etch techniques was performed. Source foils of Pu-239, Pu-240, U-235, and U-238 were irradiated in core position W2.5N1.0F. The source foils were in contact with mica solid-state track recorders (SSTR). The irradiation was performed for 1-1/2 hours at 100 watts. The SSTR foils were counted by the biology department at the University of Arkansas. From the results of the pilot run, it appears that the track density for the Pu-239 foils was too high at the 100-watt power level and that the actual irradiation will have to be performed at a power level considerably lower.

When the familiarization transient tests were completed, the familiarization FRED slug was removed and stored in the refueling cell. As soon as shielded cask storage is completed, the familiarization slug will be removed and stored outside the refueling cell. The sub-prompt transient slug was received from California and its worth measured. Since the allowable worth was very narrowly defined—less than 98 cents by the technical specifications and greater than 95 cents from program restriction—it was anticipated that the slug might likely have to be returned to California for adjustment. The initial measurement was outside the allowable limits and was returned for adjustment. It was adjusted, returned, and remeasured. The worth of the adjusted slug for the sub-prompt transient tests was measured to be 96.6 cents.

Test procedure preparation was initiated for Core II. The test procedure to replace the BeO Tightener Rods for Core II has already been issued for review. Noise measurements to be made on both Core I and Core II (just before and immediately after changing to Core II) will be made by Oak Ridge National Laboratory (ORNL). Draft procedures for these tests were forwarded to ORNL for comment. The ORNL comments have been received and incorporated into a draft procedure which has been issued for review. Work continues on the preparation of Core II procedures.

One hundred twenty-six fuel rods were delivered to the SEFOR site. Included were 104 new fuel rods, 19 reworked fuel rods which had been rejected by GE during receiving inspection in 1968, two mechanically damaged fuel rods which were damaged during handling by GE in 1969, and one dummy UO₂ rod.

Receiving inspection is in progress on these rods. Dimensional inspection has been completed, with one reworked rod being rejected for misalignment of the end plugs. This defect can be corrected at SEFOR. Helium mass spectrometer leak checks at room temperature resulted in seven additional rejects. Four of these were reworked fuel rods. At present the rods are being helium leak checked in a 750°F chamber.

9.2 OPERATION AND MAINTENANCE

Reactor Operation Statistics	Month Ending April 18, 1971	Quarter (January 18, 1971, to April 18, 1971)	Total to April 18, 1971
Hours Critical (hr)	33.9	407	2,769
Power Generated (MWh)	121	4115	19,592
Startups	3	33	452
Scrams	1	17	123
Operating Efficiency ^(a) (%)	5.7	52.1	39.5

(a) This 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

All possible test activity was concluded on January 18, 1971, when the repeatability tests were performed at 15 MW. Operation was resumed on January 22, 1971, at 1548 hours after permission was received from DRL to proceed to 20 MW with guinea pig rods under the innermost ports. The intervening time was used to perform preventive maintenance work on facility systems.

Power levels of 18 and 19 MW were achieved at 2200 and 2240 hours, respectively, on January 22, 1971, and 20 MW was achieved at 1520 hours on January 29, 1971.

All Group II and Group III Tests were completed on February 3, 1971. Provisional Test Procedure 45 "Isotopic Analysis of Reactor Vessel Cover Gas in SEFOR" was completed on February 6, 1971, at which time an outage was initiated to perform plant modifications in preparation for transient tests.

The refueling cell and nitrogen zone atmospheres were converted to air to permit modification and maintenance activities.

The oscillator was removed and the fast reactivity excursion device (FRED) installed on the positioner on the reactor vessel head. Modifications to the fuel rod assembly/disassembly station were made to permit assembly/disassembly of FRED extension rod and slug. The reactor vessel outer head studs were retensioned. Approximately 70 test firings of FRED were made to verify proper operation and to obtain statistical data on performance. Modifications were made to the reactor safety system to increase the time delay for scram during excursion testing and to provide interlocks related to the FRED firing.

The transient detector systems were calibrated and response characteristics measured. Modifications to in-cell equipment were completed; tension of the reactor vessel outer-head studs was checked; the refueling cell atmosphere was changed to argon; preventive maintenance on nitrogen blowers, gaseous radwaste compressors, and argon cooler blower was accomplished; sodium piping heater circuits were repaired; auxiliary primary reactor inlet resistance temperature detector (RTD) was replaced; instrument nitrogen compressor was overhauled; and the cell atmosphere was returned to nitrogen. The plant outage was terminated on February 25, 1971, and reactor operation and testing resumed. A total of 67 plant modifications was completed.

All eight planned tests in the Familiarization Transient test series and the associated fuel inspections were performed with the 0.5 dollar (nominal) slug. The FRED slug for the sub-prompt transient test series was received, calibrated, returned to California for adjustment, and recalibrated. After adjustment the slug worth was 0.966 dollar.

On March 14, 1971, a reactor outage was initiated to investigate the loading on the reactor vessel outer-head studs from differential movement of vessel flange and head during temperature changes of the primary coolant. Strain gages and dial indicators have provided detailed data which are currently being evaluated to determine appropriate values of pre-tension for the studs and rates of primary coolant temperature changes.

In vivo whole-body counting was performed on the SEFOR site personnel. No radioactivity attributable to the operation of the facility was detected.

A total of 198 equipment malfunctions were corrected, distributed as follows:

Mechanical	54
Electrical	71
Instrumentation	73
	198

Significant malfunctions included:

Deep well pump	Auxiliary primary reactor inlet RTD
Sodium systems heaters	Nitrogen radiation monitor
Fuel grapple load cell	Auxiliary sodium expansion tank level switch
Multi-ton load cell	Oscillator positioner position indicator
Main primary EM pump generator exciter regulator	Reactor building air handling unit fan bearing
Reactor vessel outer-head flange rotation and loading on head studs	Instrument nitrogen compressor
Safety system auxiliary relays	Freon condenser
	Nitrogen blowers time delay relay

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