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AEC RESEARCH AND
DEVELOPMENT REPORT
FEBRUARY 1971

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**SOUTHWEST EXPERIMENTAL
FAST OXIDE REACTOR
DEVELOPMENT PROGRAM
TWENTY-SEVENTH QUARTERLY REPORT
NOVEMBER 1970-JANUARY 1971**

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

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GEAP-10010-27
AEC Research and
Development Report
February 1971

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM

Twenty-Seventh Quarterly Report
November 1970—January 1971

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ABSTRACT

This report summarizes the work performed on the Research and Development Program for the Southwest Experimental Fast Oxide Reactor during the period from November 1, 1970 through January 31, 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 characteristics of power reactor operating conditions. SEFOR is particularly designed for the systematic determination of the Doppler coefficient of reactivity at temperatures up to the vicinity of fuel melting.

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

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

Phase I—Preoperational research and development,

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

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

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

1. "Southwest Experimental Fast Oxide Reactor Development Program, First Quarterly Report, April—June 1964," GEAP-4594 (July 1964).
2. "Southwest Experimental Fast Oxide Reactor Development Program, Second Quarterly Report, July—September 1964," GEAP-4742 (October 1964).
3. "Southwest Experimental Fast Oxide Reactor Development Program, Third Quarterly Report, October 1964—January 1965," GEAP-4799 (February 1965).
4. "Southwest Experimental Fast Oxide Reactor Development Program, Fourth Quarterly Report, February—April 1965," GEAP-4864 (May 1965).
5. "Southwest Experimental Fast Oxide Reactor Development Program, Fifth Quarterly Report, May—July 1965," GEAP-4934 (August 1965).
6. "Southwest Experimental Fast Oxide Reactor Development Program, Sixth Quarterly Report, August—October 1965," GEAP-4994 (November 1965).
7. "Southwest Experimental Fast Oxide Reactor Development Program, Seventh Quarterly Report, November 1965—January 1966," GEAP-5108 (February 1966).
8. "Southwest Experimental Fast Oxide Reactor Development Program, 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).

In addition, the following topical reports have been issued:

1. Novak, P. E., and Asamoto, R. R., "A Survey for a High Temperature Sensor for SEFOR," GEAP-4903 (July 1965).
2. Quinn, E. P., "Vibration of SEFOR Fuel Rods in Parallel Flow," GEAP-4966 (September 1965).
3. Johnson, M. L., "A Description of Fuel and Other Core Materials Procured for SEFOR Critical Experiments in ZPR-III," GEAP-5133 (December 1965).
4. Protsik, R., "Epithermal Capture and Fission Distribution in a Fuel Rod—The Computer Code REAX," GEAP-4973 (April 1966).
5. Novak, P. E., and Asamoto, R. R., "An Out-of-Pile Evaluation of W-Re Thermocouple Systems for Use to 4700°F in Mixed Oxide," GEAP-5166 (May 1966).
6. Pflasterer, G. R., "SEFOR Experimental Program Planning, Vols. I and II," GEAP-5092 (August 1966).
7. Morrell, R. P., "SEFOR β/λ 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. Cochran, J. T., "SEFOR Fuel Rod Sodium Thermal Shock Tests," GEAP-5467 (August 1968).
26. Craig, E. R., "SEFOR Instrumented Fuel Assembly Design and Development," GEAP-5615 (April 1968).
27. Steamer, A. G., Brynsvold, G. V., and Franckx, L. C., "Fuel Handling and Other Tests in the SEFOR Refueling Cell Mockup," GEAP-5701 (March 1969).
28. Johnson, M. L., Becker, R. A., Haar, J. M., "SEFOR Fuel Handling," GEAP-13565 (December 1969).
29. Noble, L. D., et al., "Results of SEFOR Zero Power Experiments," GEAP-13588 (March 1970).
30. Ikeuye, D. T., and Young, F. E., "Fabrication of Non-Fuel Core Components for SEFOR," GEAP-13595 (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).

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

Task 6 requires neutron absorber rods with three different reactivity worths. The rods have been designed; fabrication of the rod having the lowest reactivity worth is under way.

2.3 TASK 9 – FUEL FABRICATION

Nuclear Fuel Services (NFS) continued the fabrication of SEFOR fuel rods during the quarter. The production of fuel pellets was completed (five pellet batches) and approximately 60% of the fuel rods have been loaded and welded.

The O/M ratio of the last two batches of fuel pellets was reported to be out of specification (high) after the pellets were processed through the outgassing furnace.

Sample fuel pellets from the first three batches of fuel pellets were received by GE and analysis of the samples was started. The production activities at NFS were audited during the report period by the GE resident engineer.

Fabrication continued on the Core II tightener rods, and an initial shipment of tightener rods was received.

Fabrication of the final (6th) instrumented fuel assembly (IFA) for Core I continues. An information exchange meeting was held with AEC-DRDT, ANL, ORNL, HEDL representatives on the reactor operating experience and failure diagnosis.

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.

2.4 TASK 10 – STARTUP TESTING

Results of power, reactivity, and fuel temperature measurements obtained from static and oscillator tests during operation at power levels up to 17.2 MW indicate that the SEFOR Doppler coefficient $T dk/dT$ is -0.0081 with an estimated standard deviation of ± 0.0018 . The experimental value compares favorably with the value of -0.0082 that was predicted prior to the experiments.

Data from conventional oscillator tests at power levels up to 17.2 MW have been analyzed. The experimental frequency-dependent reactivity feedback is reproducible, and it shows good agreement with the theoretical model at all power levels and coolant flows. The reactivity feedback is independent of flow and the feedback effects appear to depend only on the fuel temperature and coolant temperature oscillations. The power reactivity coefficient obtained from these tests is in agreement with the static test data and with the calculated Doppler coefficient $T dk/dT$ of -0.0081 . The fuel time constant is of the order of 28 seconds at all power levels; which is in good agreement with the expected value only at low power levels. The reactivity energy coefficient that is inferred from the oscillator test data decreases more rapidly in magnitude at higher power levels than was expected however, and reasons for this difference are currently being investigated.

Analyses of experiments to determine the reflector temperature coefficient of reactivity at power levels of 10 and 15 MW show, in agreement with the earlier tests at 5 MW, that the reflector temperature coefficient of reactivity is essentially zero.

Analysis of successive sets of data at a power level of 5 MW indicates that the standard deviation in reactivity is ± 1.6 cents or 3% of the reactivity change, whichever is larger, when the reactor is at power; the estimated uncertainty in absolute power level is ± 0.1 MW or $\pm 5\%$ of the mean power level, whichever is larger. The random deviation in relative power level is estimated to be ± 0.1 MW or $\pm 2\%$.

Gamma emission scans along the length of guinea pig fuel rods obtained after operation at 10 MW were in good agreement with the calculated axial power distribution. Emission scans obtained after operation at higher power levels indicated that the volatile fission products may move toward the cooler regions of the pellets as they are operated at higher temperatures.

2.5 TASK 11 – TRANSIENT TESTS

The data-reading smoothing and integrating portions of the computer code which will be used in the analysis of transient data has been debugged. Checkout of the kinetic inversion portion of the code is continuing.

2.6 SAFETY AUDIT AND LICENSING

Changes required by the Technical Specifications for 20 MW operation and initiation of the transient test program were proposed by GE and reviewed by DRL. Partial approval was received in January; full approval is expected by early February.

Isotopic analyses of the reactor vessel cover gas were made to demonstrate the capability of detecting and measuring fission products which would be present in the event of failure of fuel cladding.

A report of test data through 10 MW was sent to the DRL to show that the Doppler coefficient and reactivity coefficients meet the requirements of the Technical Specifications.

The effects of sodium temperature and reflector segment position on calibration of the Wide Range Monitor (neutron flux measurement) were determined.

Modifications to the nitrogen cooling system were tested and found to provide the required system cooling improvement.

An audit inspection was made at the site to verify compliance with Technical Specifications.

Plant Modification Requests were reviewed and evaluated with respect to safety and licensing requirements.

The results of the natural circulation test were reviewed and found to be satisfactory.

2.7 SEFOR SUPPORT

During the report period, a planned outage was completed for installation of the NaK bubbler refueling cell purification system. During the same outage, modifications and improvements to the nitrogen cooling system were completed. Difficulties were experienced with excessive reactor core gas leakage. Retensioning of the outer head studs reduced the leakage to an acceptable level. Design of a backup seal for the reactor outer head was initiated.

2.8 SEFOR OPERATIONS

When the results of the quarterly pressure decay test of one containment penetration fell outside acceptable limits, an outage scheduled for the latter part of November was initiated on November 2, 1970. Major installation and modification activities included a replacement argon purification system, nitrogen cooling duct changes, the second

replacement reactor vessel sodium level probe, neutron shielding in the primary sodium equipment area, transient power detectors and energy probes, and automatic balanced oscillator control equipment. Maintenance and modification activities were completed and cell atmosphere returned to inert gases.

Reactor operation and program testing were resumed on December 3, 1970. Observations of the performance of the transient detector system were made during the increase-in-power and steady-state operation. Balanced oscillator measurements were made at 13, 15, and 17 MW at average primary coolant temperature of 760°F and 10 MW, 560°F. These measurements concluded all tests planned in the Experimental Program (13 and 17 MW were additions to the planned program) that could be performed without AEC approval of changes to Technical Specifications which permit operation above 17.5 MW. The proposed changes were first submitted to DRL on July 8, 1970 (change No. 3) and September 3, 1970 (change No. 4). Consequently, the reactor was shut down and program testing suspended until the approvals are received.

Procedures for the sub-prompt and super-prompt transient tests were issued for review and comment.

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 excursions during the experimental program.

3.1.2 Past Accomplishments

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

3.1.3 Poison Rods

The experimental program requires neutron absorber rods with three different reactivity worths for the FRED tests. The design of the three rods has been completed. The FRED poison rod design provides for worth adjustment and remote disassembly from the extension shaft should either prove necessary.

The FRED absorber rod design specification, development guide, and quality control plan were issued. All materials including the enriched B_4C for the Phase II ($\leq 1\$$) and Phase III ($\sim 1.3\$$) rods have been received.

Parts for the Phase I (~ 50 cents) rods are completed and fabrication is under way.

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 as well as the instrumented fuel assembly are part of this task.

4.2 PAST ACCOMPLISHMENTS

The complete fuel-rod order had been received at SEFOR in previous reporting periods. Twenty-two leaking fuel rods were returned to NFS for repair. Fabrication was started on 102 additional rods. Five instrumented fuel assemblies were shipped to the site.

4.3 CURRENT RESULTS

4.3.1 Fuel Rod Procurement

Nuclear Fuel Services 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

The repair of 19 of the 22 helium leaking rods was completed. The repair of two of the fuel rods was not possible because of mechanical damage. The end plug weld on one fuel rod was rejected because the weld thickness on the associated, destructively examined, sample weld was 0.001 inch less than that specified.

4.3.1.2 Fabrication of New Fuel Rods

The processing of fuel pellets through centerless grinding was completed on all pellet batches. There was a total of five pellet batches for this production run (701A, 802, 803, 804, and 805).

The center spacer was electron beam welded into a total of 93 tubes. Tubes and spacers to complete the required number of fuel rods are at Buehler Corp. ready for welding.

The top and bottom end plugs intended for use in the new fuel rods were thermal cycled to 825°F and helium-leak checked. The thermal cycling and leak check were intended to detect end plugs that might leak after they are welded to the fuel rods. The leak check after thermal cycling indicated two leaking end plugs out of a total of approximately 340 tested.

Loading of fuel pellets into tubes and welding end plugs to the rods was started. To date, 63 fuel rods have been loaded and 57 of these rods have been welded. NFS reported that during the fuel rod helium leak check some lower end plugs were found to leak. These end plugs had been thermal cycled and leak checked prior to welding. Evidently, thermal cycling and prior leak checking did not eliminate all defective end plugs. Additional bottom end plugs were machined from a new heat of bar stock, and these were used to replace the leaking end plugs.

The fuel pellet O/M ratio for Batches 804 and 805 was found to be out of specification (high) after the pellets were processed through the outgassing furnaces. This involved the last 5 of the 13 outgassing runs and the pellets from these runs were held as discrepant materials. Investigation into the cause of the problem revealed that the pellets had a high O/M ratio prior to being outgassed; the sintering furnace atmosphere was determined to be out of control during the sintering of the last two batches of pellets.

NFS will re-fire the affected pellets to adjust their O/M Ratio. GE has requested information from NFS on all fuel pellet batches to assure that they are within specification.

The gamma scan equipment to check the plutonium content of the fuel rods was assembled and checked out. Two standard rods were used to set the accept-reject criteria as specified in the NFS procedure.

The standard acceptable rod is numbered 016 and is loaded with fuel pellets from Batch 701A. Samples from this batch were analyzed for plutonium content by NFS, GE, and AEC New Brunswick Laboratory, with good agreement.

The standard defect rod is loaded with pellets from Batch 701A in the lower half only; this fuel stack has a UO₂ pellet in the center.

Gamma scanning of completed fuel rods was started.

4.3.1.3 Quality Assurance

A GE Resident Engineer audited all NFS production activities during the report period.

Fuel batch samples from Batches 701A, 802, and 803 were received by GE and verification analysis of these samples was started. The O/M ratio of the samples from Batches 701A, 802, and 803 were checked and found to be within specification (1.97 to 1.99). Inspection of the gamma-scan records for 55 completed fuel rods disclosed an apparent irregularity in the chart recorder speed for those scans performed on December 30, 1970. NFS will re-scan the several rods involved.

4.3.2 Tightener Rods

Fabrication continued on the Core II tightener rods, which are solid stainless steel. An initial shipment of tightener rods was received.

4.3.3 Instrumented Fuel Assemblies

Fabrication of the final (6th) IFA for Core I continues at the VNC-IPO shop.

One of the instrumented fuel rods (G8-1) as delivered for assembly was found (by x-ray) to have a reduced spring load on the pellet stack in the center segment. This will not adversely affect the IFA operation in the reactor if the pellets near the thermocouple hot junction are in the proper position when the IFA is inserted in the core. Verification of this proper relationship by an x-ray or gamma scan of the fuel rod center section is planned after the IFA has been moved to the refueling cell and raised to a vertical position.

The fuel rods were straightened and the leads were checked out for continuity and resistance of electrical insulation on the thermocouple leads.

The component parts (other than the fuel rods) of the IFA were retrieved from storage and the required reinspection completed. Sample weld specimens were machined for requalification of technicians on the critical weld joints. The tightener rod subassembly was set up in the final assembly fixture.

An interim report on the instrumented fuel assembly fabrication was compiled and sent to the AEC; this report contains a physical description of the IFA and a summary of the fabrication highlights related to the sensors. This information was sent in preparation for a presentation on Core I-IFA design and trouble diagnosis at the AEC-DRDT in Washington, D.C. The information presented at this meeting supplemented the written report and related the IFA operating data with diagnostic work on the sensor leads performed at the SEFOR site. In-core instrumentation experience was discussed and information exchanged between GE, ORNL, HEDL, ANL, and DRDT representatives.

The discussion served to confirm ideas on possible failure mechanisms, such as parts-per-million contamination of W/Re thermocouples. Information was exchanged on the use of a Time-domain Reflectometer for sensor lead trouble diagnosis.

4.3.4 Conceptual Design of Instrumented Fuel for Core II

The SEFOR Core II conceptual designs for instrumented fuel were initiated and completed. The conceptual design drawings and review comments will be used for final design of an IFA starting in February 1971.

The present conceptual design uses three instrumented fuel rods located at three separate through-head ports. Each rod is instrumented with one fuel thermocouple that is inserted axially five inches from top of core fuel. Two spares are planned as backups to the three instrumented fuel rods. All rods are designed to be interchangeable.

The intent is to place the Core II IFA's in the core only for the super-prompt critical tests. The planned fuel thermocouple location is such that the maximum thermocouple temperature is approximately 3,000°F, yet the thermocouple hot junction is in a position of well-defined power generation (determined from calculation, fission foil irradiation, and fuel rod gamma-scan).

Layout and scoping drawings were made of fuel thermocouple design options, instrumented fuel rod options, extension rod, adapter rod, through-head port penetration, in-cell handling of rods, in-cell storage for rods, and shipping container for all five rods.

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 all experimental data collected during performance of the experimental program at zero power was completed. Preliminary results of analysis of 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 $T dk/dT$ of -0.0081 .

5.3 CURRENT RESULTS

5.3.1 Comparison of Measured and Calculated Doppler Effects

Measurements of the reactivity feedback in SEFOR were performed at power levels up to 17 MW, and at average coolant temperatures between 350 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 maximum of approximately 4000°F. The time-dependent reactivity feedback was determined from the frequency response of the system to sinusoidal oscillations in input reactivity, and the static changes in reactivity were determined over the complete range of power and coolant temperature conditions. Analysis of the data and comparison of results with calculations indicates that the SEFOR Doppler coefficient $T dk/dT$ is -0.0081 with an estimated standard deviation of ± 0.0018 . This compares favorably with the value of -0.0082 that was predicted prior to the experiments.

The primary contributor to the estimated uncertainty in the SEFOR Doppler fuel temperature coefficient $T dk/dT$ is the uncertainty in the calculated conversion factor between power level and effective core average fuel temperatures. The deviations between calculated and measured Doppler power-reactivity effects are of the order of 7% (see below). However, the relationship between power level and fuel temperature involves the fuel thermal conductivity and the fuel-to-cladding heat transfer coefficient. The estimated uncertainty of $\pm 15\%$ in thermal conductivity causes a corresponding $\pm 15\%$ uncertainty in the Doppler temperature coefficient, and a large uncertainty in the fuel-to-cladding gap coefficient also causes an estimated $\pm 15\%$ uncertainty in the temperature coefficient. Analysis of the fuel thermocouple data obtained during these static and oscillator tests is under way, and the results may reduce these last two uncertainties. The combined estimated deviation in the Doppler coefficient is then the square root of the sum of the squares of all these deviations, or 22% of $0.0081 = 0.0018$.

All of the static test results up to 17.2 MW, as well as the experimental power reactivity transfer function and feedback data from the oscillator tests at power levels up to 10 MW were reported in GEAP-10010-26.² The oscillator test data at 15 and 17 MW are reported in Section 5.3.2. Since the SEFOR fuel was designed 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. The experiments have shown, in agreement with calculation, that the only other significant reactivity effects are those associated with changes in the average core coolant temperature.

The Doppler components of the measured reactivity feedback effects are summarized, and compared with effects calculated using a Doppler temperature coefficient $T dk/dT$ of -0.0081 in Table 5-1. The integrated reactivity feedback data was obtained from the static test data in GEAP-10010-26 using the new SEFOR reactivity conversion formulas that are also discussed in GEAP-10010-26.² The corresponding calculated effects were obtained from the WEDOP code, which applies a power-squared weighting factor to local reactivity effects throughout the core to modify (by ~5%) the results which would be obtained from a simple calculation using the logarithm of an average core fuel temperature. The power reactivity coefficients were obtained from the static tests by differentiating polynomial fits to the experimental static data. The fits to the oscillator test data (see Section 5.3.2) yielded the power coefficients directly, while the

Table 5-1
SEFOR DOPPLER MEASUREMENTS

Power Level (MW)	Total Integrated Doppler Effect from Zero Power		μ Doppler Power Reactivity Coefficient			τ Time Constants		η Doppler Energy Coefficient	
	Static(a)	Calculated(b)	Oscillator(a)	Static(a)	Calculated(b)	Oscillator	Calculated	Oscillator(a)	Calculated(b)
	Tests		Tests	Tests		Tests		Tests	
	(d)	(d)	(d /MW)	(d /MW)	(d /MW)	(sec)	(sec)	(d /MW-sec)	(d /MW-sec)
0.0	0.0	0.0	—	-16.5±2.0	—	—	—	—	—
2.0	-31±2	-36	-15.8±0.9	-14.5±1.0	-15.5	30.1±2.0	30.6	-0.56±0.05	-0.54
5.0	-71±5	-75	-10.9±0.7	-11.6±0.8	-11.1	28.0±2.5	25.9	-0.41±0.05	-0.45
10.0	-120±8	-120	- 8.3±0.7	- 8.5±0.7	- 7.6	28.7±2.5	21.4	-0.30±0.04	-0.37
15.0	-156±10	-154	- 5.8±0.8	- 7.0±0.6	- 6.0	27.7±2.5	19.6	-0.22±0.04	-0.33
17.2	-172±11	-167	- 5.0±0.8	- 6.2±0.5	- 5.6	26.7±2.5	19.0	-0.20±0.04	-0.31

$$\eta = \frac{\mu}{0.943\tau}$$

- a. Uncertainties include an estimate 5% uncertainty in absolute power level.
- b. Calculated using a Doppler coefficient $T \frac{dk}{dT} = -0.0081$ with the WEDOP code.

calculated coefficients were obtained from the WEDOP code. The "measured" energy coefficient was calculated (see Section 5.3.2) by dividing the Doppler component of the measured oscillator power coefficient by 0.943* times the measured time constant, while the calculated energy coefficient was obtained directly from the WEDOP code. The measured fuel time constants were obtained directly from analysis of the oscillator test data as described in Section 5.3.2 while the "calculated" time constants were determined by dividing the calculated power coefficient by 0.943* times the calculated energy coefficient.

The values in Table 5-1 show that there is good agreement at all power levels between the measured and calculated values of the integrated Doppler reactivity effect, and that there is generally good agreement among the power coefficients determined from the static tests, oscillator tests, and from calculations. At higher power levels the reactivity coefficients that were determined from the oscillator tests are smaller in magnitude than the static and calculated coefficients, however, and this apparent reduction in magnitude, combined with a larger measured than calculated fuel time constant causes the magnitude of the "measured" energy coefficients to be less than the calculated values at the higher power levels. Reasons for this difference are currently being investigated.

5.3.2 Conventional Oscillator Tests to 17 MW

Oscillator experiments are performed at several power levels and coolant flows to demonstrate reactor stability, to provide data for comparison with the power coefficients determined from the static tests, and to provide for the determination of the frequency response of the fuel. As expected, the reactor was quite stable, and, as will be shown, the experimental reactivity feedback was in good agreement with the model. The frequency response of the fuel temperature is currently being analyzed and will be reported at a later date. The conventional oscillator tests were performed at zero power, and at nominal power levels of 2, 5, 10, 15, and 17 MW. The zero power oscillator test results were reported in GEAP-10010-25.⁴ Data from the tests up to 10 MW were reported in GEAP-10010-26.² The data at 15 and 17.2 MW are given below along with the analysis of results.

5.3.2.1 Description of the Experiment

As with the zero power oscillator tests, the conventional oscillator tests were performed by oscillating a neutron absorber (boron carbide) rod in the central dry well 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.

In all of the measurements the lower end of the oscillator rod was at an indicated mean height of 5.5 inches above the core mid-plane. This location was selected because oscillator rod worth measurements² had indicated a nearly linear relationship between reactivity and rod position in this region of the core. To limit the amplitude of the power oscillations, the rod was oscillated about its mean position with an amplitude of ≤ 2 inches. The actual amplitudes that were used for the various tests are listed in GEAP-10010-26 and in Tables 5-2 through 5-4. All amplitudes were converted to reactivities using a constant conversion factor of 2.736 δ /in. that was determined from the analysis⁴ of the zero power oscillator tests.

*This factor corrects for the energy that is absorbed in non-fuel portions of the core.

Table 5-2
OSCILLATOR DATA
 15 MW, 60% Flow

Set 1 Tapes 41 & 47			Set 2 Tape 47		
Frequency	Rod Amplitude (inches)	File No.	Frequency	Rod Amplitude (inches)	File No.
F2	1.4	46	S6	1.4	4
F3	1.4	47	S5	1.4	5
F4	1.4	48	S4	1.4	6
F5	1.4	49	S3	1.4	7
F6	1.4	50	S2	1.4	8
S1	1.4	51	S1	1.4	9
S2	1.4	52	F6	1.4	10
S3	1.4	53	F5	1.4	11
S5	1.4	2	F4	1.4	12
S6	1.4	3	F3	1.4	13
			F2	1.4	14

Table 5-3
OSCILLATOR DATA
 15 MW, 75% Flow, Series A

Set 1 Tape 41			Set 2 Tape 41		
Frequency	Rod Amplitude (inches)	File No.	Frequency	Rod Amplitude (inches)	File No.
F2	1.4	12	F2	1.4	24
F3	1.4	13	F3	1.4	25
F4	1.4	14	F4	1.4	26
F5	1.4	15	F5	1.4	27
F6	1.4	16	F6	1.4	28
S1	1.4	17	S1	1.4	29
S2	1.4	18	S2	1.4	30
S3	1.4	20	S3	1.4	31
S4	1.4	21	S4	1.4	32
S5	1.4	22	S5	1.4	33
S6	1.4	23	S6	1.4	34

Table 5-4
OSCILLATOR DATA
17.2 MW, 75% Flow, Series A

Set 1			Set 2		
Tapes 65 & 55			Tape 55		
Frequency	Rod Amplitude	File No.	Frequency	Rod Amplitude	File No.
S6	1.45	53	S6	1.45	6
S5	1.45	54	S5	1.45	7
S4	1.45	55	S4	1.45	8
S3	1.45	56	S3	1.45	9
S2	1.45	57	S2	1.45	10
S1	1.45	58	S1	1.45	11
F5	1.45	2	F6	1.45	12
F4	1.45	3	F5	1.45	13
F3	1.45	4	F4	1.45	14
F2	1.45	5	F3	1.45	15
			F2	1.45	16

Up to 11 different frequencies of oscillation ranging between ~ 0.24 Hz and 0.0018 Hz were used for each set of oscillator data at a given power level and coolant flow, and the complete range of frequencies was repeated 2 or 3 times. Cyclic equilibrium was obtained at each frequency and several cycles of data were recorded on magnetic tape as one file of data. The number of cycles of data in each file ranged from a minimum of 3 cycles (in some instances) at low frequencies to a maximum of ~ 30 cycles at the higher frequencies. The nominal frequencies of oscillation used for the experiments are listed in Table 5-5.

The oscillator experiments were performed during the first extended period of operation at a given power level, and they were then repeated during the second period of extended operation at the same condition to verify the reproducibility of the data. The former measurements are designated as Series A while the latter are designated as Series B. The experimental data at power levels up to 10 MW are tabulated in GEAP-10010-26. Results of the 15, and 17.2 MW tests are listed in Tables 5-6 through 5-8.

Table 5-5
NOMINAL FREQUENCIES OF OSCILLATION

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

Table 5-6
**POWER REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
 15 MW, 60% Flow**

Frequency (rad/sec)	Power Reactivity Function		Feedback Kernel	
	Amplitude ($\frac{\text{percent power}}{\text{cents reactivity}}$)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
Set 1				
0.01110	0.912	27.31	0.07466	146.99
0.01764	1.034	25.31	0.06529	145.13
0.04886	1.489	27.42	0.04393	126.81
0.07807	1.896	19.52	0.03194	118.23
0.1290	2.136	3.30	0.02171	109.58
0.2196	1.865	-12.23	0.01327	104.23
0.3478	1.569	-16.57	0.00850	100.10
0.6052	1.324	-16.06	0.00450	100.45
0.9729	1.204	-13.76	0.00267	103.99
1.5545	1.124	-11.16	0.00174	108.76
Set 2				
0.0105	0.928	27.45	0.07342	146.78
0.0175	1.037	25.00	0.06508	145.46
0.0304	1.202	27.19	0.05403	136.71
0.0487	1.498	27.32	0.04367	126.80
0.0779	1.899	19.95	0.0329	117.87
0.1291	2.129	2.75	0.02147	110.12
0.2147	1.880	-11.58	0.01371	104.41
0.3440	1.579	-16.28	0.00879	99.60
0.6028	1.295	-16.16	0.00449	115.92
0.9726	1.211	-13.53	0.00292	96.89
1.5612	1.132	-11.27	0.00159	93.41

Table 5-7
 POWER REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
 15 MW, 75% Flow, Series A

Frequency (rad/sec)	Power Reactivity Function		Feedback Kernel	
	Amplitude ($\frac{\text{percent power}}{\text{cents reactivity}}$)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
Set 1				
0.01126	0.940	27.65	0.07249	146.41
0.01780	1.068	26.70	0.06349	143.39
0.03055	1.271	27.70	0.04102	135.27
0.04871	1.579	25.56	0.04102	127.02
0.07759	1.943	17.43	0.03047	119.01
0.1284	2.155	1.13	0.02073	111.76
0.2050	1.888	-11.91	0.01359	106.93
0.3351	1.573	-16.67	0.00862	103.23
0.5926	1.332	-16.07	0.00468	100.07
0.9597	1.205	-13.73	0.00281	104.88
1.5422	1.121	-11.36	0.00163	115.72
Set 2				
0.01149	0.951	27.98	0.07174	145.00
0.01807	1.068	26.38	0.06341	143.59
0.03095	1.279	27.79	0.05233	134.95
0.04917	1.575	26.09	0.04127	126.48
0.07847	1.947	17.05	0.03025	118.99
0.1295	2.117	1.36	0.02084	111.22
0.2143	1.866	-12.09	0.01341	105.75
0.3434	1.575	-16.39	0.00870	100.54
0.6014	1.328	-15.80	0.00479	99.84
0.9723	1.204	-13.64	0.00279	104.05
1.5545	1.130	-11.35	0.00154	98.62

Table 5-8
**POWER REACTIVITY TRANSFER FUNCTION AND FEEDBACK KERNEL AT
 17.5 MW, 75% Flow**

Frequency (rad/sec)	Power Reactivity Function		Feedback Kernel	
	Amplitude ($\frac{\text{percent power}}{\text{cents reactivity}}$)	Phase (degrees)	Amplitude (\$/MW)	Phase (degrees)
Set 1				
0.0111	0.922	29.52	0.06351	144.78
0.0175	1.050	27.87	0.05555	142.56
0.0304	1.266	28.02	0.04537	135.07
0.0487	1.557	25.32	0.03555	127.61
0.0779	1.890	16.59	0.02642	120.85
0.1291	2.035	2.10	0.01863	114.01
0.3536	1.549	-15.77	0.00768	102.32
0.6137	1.322	-15.33	0.00431	99.95
0.9836	1.205	-13.27	0.00263	100.68
1.5715	1.124	-11.02	0.00154	104.88
Set 2				
0.0114	0.929	29.66	0.06306	144.45
0.0179	1.058	28.13	0.05517	142.05
0.0308	1.259	27.96	0.04558	135.09
0.0492	1.567	25.26	0.03531	127.38
0.0784	1.889	16.48	0.02638	120.82
0.1295	2.031	2.01	0.01861	114.08
0.2145	1.838	-10.75	0.01226	107.59
0.3428	1.564	-15.56	0.00797	102.90
0.6032	1.331	-15.33	0.00449	98.93
0.9734	1.204	-13.39	0.00259	103.66
1.5621	1.126	-11.06	0.00154	104.42

5.3.2.2 Theoretical Model

The oscillator test data are analyzed by comparing the measured frequency-dependent reactivity feedback kernel, and the measured frequency-dependent average coolant temperature to power transfer function with a theoretical model. The "measured" reactivity feedback kernel, $h(iw)$ at frequency w , is determined from the calculated (based on experiment⁴) zero power reactivity transfer function G_0 , and the measured at-power reactivity transfer function, G , at frequency w through the following relation:

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

In this relation P is the mean power level at which the oscillator test was performed, while the other notation is given above. The transfer function between average core coolant temperature and power, $U(iw)$, is determined from the measured core inlet, $\Delta T_i(iw)$, core outlet, $\Delta T_o(iw)$, and power, $\Delta P(iw)$, oscillations according to the following relation

$$U(iw) = F_i(iw) \left[\frac{\Delta T_i(iw)}{\Delta P(iw)} \right] + [1 - F_i] \left[\frac{\Delta T_o(iw)}{\Delta P(iw)} \right]$$

In this expression F_i is a calculated function that depends on the frequency, the coolant flow, the shape of the axial power distribution in the core, the time constants of the fuel and coolant, etc. The expressions that are used for the calculation of this function (are defined in Reference 4 at low frequencies $F_i \sim 1/2$).

The theoretical model for the relation between the frequency dependent reactivity feedback and the average coolant temperature to power transfer function is described in GEAP-5576.⁵ If the inlet temperature coefficient of reactivity is set to zero in accordance with the results of the static tests,² the model may be written as follows:

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

In this model α is the power reactivity coefficient at a constant average coolant temperature, C_f and C_q are the fuel and core average coolant temperature reactivity coefficients respectively, $h(iw)$ is the feedback kernel discussed above, and the other quantities are defined as follows:

$F_{av}(iw, \gamma, \nu)$ = the frequency response of the average fuel temperature to a power oscillation at frequency w (normalized to unity at $w=0$).

$X_f(iw)$ = the frequency response of the ratio of the total power absorbed in the fuel to the "fission" power (normalized to unity at $w=0$). This function⁵ is used to account for the frequency dependence of the delayed gamma heating.

$F_s(iw, \gamma, \nu)$ = the frequency response of the average fuel temperature to a coolant temperature oscillation at frequency w (normalized to unity at $w=0$).

$U(iw)$ = the measured frequency response of the average coolant temperature to a power oscillation at frequency w .

The mean power level used in the determination of $h(iw)$ is measured in megawatts and the parameter α thus has units of $\$/MW$. Accordingly, the parameters C_f and C_q have units of $\$/^\circ F$, $U(iw)$ has units of $^\circ F/MW$ and the functions F_{av} , F_s and X_f are dimensionless.

The frequency response functions F_{av} and F_s may be written as follows:

$$F_s = \frac{2J_1(x)/x}{J_0(x) + iw\gamma 2J_1(x)/x}$$

$$F_{av} = \frac{1}{iw[\gamma + (\nu/8^2)]} [1 - F_s]$$

where $J_0(x)$ and $J_1(x)$ are Bessel functions of the first kind of order zero and order one, respectively.

The argument X is defined as $X = i^{3/2} \nu \sqrt{w}$ where w is the frequency of oscillation in radians per second, $i = \sqrt{-1}$, and the parameters γ and ν are defined in terms of fuel density, ρ , heat capacity, c , average conductivity, k , fuel pellet radius, R , and the effective heat transfer coefficient between fuel surface and coolant, h_f , as follows:

$$\gamma = \rho c R / 2 h_f$$

$$\nu^2 = \rho c R^2 / k$$

A good approximation for F_{av} is

$$F_{av} \cong \frac{1}{1 + iw\tau}$$

where the effective fuel time constant τ is defined as

$$\tau = \gamma + (\nu^2 / 8)$$

As noted in reference 5, the Doppler reactivity energy coefficient, η can be determined from the above parameters according to the following expression

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

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

5.3.2.3 Measured Feedback and Coolant Temperature Oscillations

To fit the theoretical model to the data, measured values of the feedback kernel, h , and the average coolant to power transfer function, U , are required at a number of frequencies. The steps involved in the determination of these functions are described in detail in Reference 5 and they will only be summarized here.

As discussed in Section 5.3.2.2 the feedback kernel is obtained from the measured power reactivity transfer function. To obtain this transfer function, the mean value of the flux signals and the harmonic components of the flux and oscillator rod position signals were determined from the recorded experimental data. The amplitude of the rod oscillation was converted to a reactivity oscillation by multiplying the measured displacements with the constant conversion factor of 2.736 ¢/in. that was determined from analysis⁴ of the zero power oscillator tests. The power

reactivity transfer function was then calculated at all frequencies by dividing the fundamental component of the flux signal by the product of the flux mean value and the fundamental component of the reactivity oscillation.

As discussed in Section 5.3.2.2, the determination of the transfer function, $U(i\omega)$, between average core coolant temperature and power involves the measured response of the core inlet and outlet coolant temperatures. Because of inlet and outlet thermocouple instrumentation problems, the correspondence between the true core average coolant temperature oscillations and the temperature oscillations indicated by thermocouples is questionable. Although no other* inlet thermocouple was available for comparison with the measured oscillations, the coolant inlet thermocouple in the central instrumented fuel assembly appeared to be working satisfactorily during the initial oscillator tests at power levels up to 10 MW. Subsequently, however, its performance, as demonstrated by the indicated mean temperature, appeared to deteriorate. However, the indicated oscillations about the mean value appeared to be reasonable, in general agreement with previous measurements, and since this was the only inlet thermocouple available, it was used in the determination of the average coolant temperature response.

The core coolant outlet temperature frequency response was usually determined from the temperature readings indicated by one of the coolant outlet thermocouples in the central IFA. The frequency response of all IFA outlet coolant temperature thermocouples were in general agreement at all frequencies, however, and on those occasions when the usual central IFA outlet thermocouple indicated an abnormally high mean value or unusual frequency response characteristics, another IFA outlet thermocouple was used instead.

The data from the measured power reactivity transfer functions, feedback kernels, and transfer functions, $U(i\omega)$, between reactivity and core average coolant temperature are tabulated in Tables 5-6 through 5-21 and in Reference 2. The coolant temperature oscillations were very small at 2 MW and they were treated as zero.

Table 5-9
AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
5 MW—30% Flow
Series A

Set 1			Set 2		
Frequency (rad/sec)	$\Delta T_c / \Delta P$ (° F/MW)		Frequency (rad/sec)	$\Delta T_c / \Delta P$ (° F/MW)	
	Real	Imaginary		Real	Imaginary
0.01116	4.6	-2.0	0.0112	4.6	-2.0
0.01764	4.7	-3.0	0.0176	4.8	-2.3
0.3058	2.8	-4.0	0.0306	2.6	-3.6
0.04881	0.9	-3.3	0.0494	1.2	-3.8
0.07819	-0.3	-2.4	0.0493	1.1	-3.5
0.1299	-0.7	-1.5	0.0789	0.0	-2.4
0.2280	-0.8	-0.5	0.1307	-1.2	0
0.3592	-0.3	-0.3	0.2192	2.0	-1.3
0.6201	-0.2	-0.2	0.3525	-0.7	-0.2
0.9908	-0.1	0.3	0.6138	1.7	-0.3
1.5790	0.0	0.0	0.9871	0	-0.1
			1.5751	-0.7	-1.4

Table 5-10
 AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
 5 MW—30% Flow
 Series B
 Sets 1 & 2

Frequency (rad/sec)	$\Delta T_c / \Delta P$ (° F/MW)	
	Real	Imaginary
0.01089	4.8	-3.0
0.01739	4.8	-3.2
0.01755	5.3	-2.9
0.03028	3.4	-3.7
0.04826	2.0	-3.5
0.07729	0.8	-2.8
0.1283	0.6	-1.8
0.1291	0.1	-2.0
0.2098	0.0	0.0
0.3378	0.0	0.0
0.5941	0.0	0.0
0.9613	0.0	0.0
0.9739	0.0	0.0
1.5397	0.0	0.0

Table 5-11
 AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
 5 MW—50% Flow
 Series A

Frequency (rad/sec)	Set 1 $\Delta T_c / \Delta P$ (° F/MW)		Frequency (rad/sec)	Set 2 $\Delta T_c / \Delta P$ (° F/MW)	
	Real	Imaginary		Real	Imaginary
0.01082	0.7	-6.5	0.009925	1.9	-4.0
0.02996	2.0	-4.4	0.01656	2.0	-3.0
0.03032	1.9	-3.4	0.02930	1.8	-2.9
0.04858	1.3	-3.2	0.04866	1.0	-3.2
0.07795	3.4	-3.2	0.07770	0.3	-2.7
0.12880	1.5	-2.7	0.1291	0.1	-2.1
0.21050	1.9	-5.2	0.2179	0.2	-2.4
0.97650	2.3	-2.5	0.3526	0.3	-2.2
1.5570	3.8	-3.0	0.5996	0.6	-1.7
			0.9691	0.6	-1.8
			1.5562	1.5	-1.6

Table 5-12
AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
5 MW—50% Flow
Series B
Sets 1 & 2

Frequency (rad/sec)	$\Delta T_c/\Delta P$ (°F/MW)	
	Real	Imaginary
0.01054	2.4	-3.0
0.01053	2.6	-3.0
0.01698	2.6	-2.3
0.02969	2.3	-2.0
0.02978	2.5	-2.0
0.04776	1.5	-2.0
0.07682	0.8	-1.9
0.1269	0.3	-1.3
0.2065	0.0	-1.0
0.2078	0.0	0.0
0.3343	-0.4	-0.7
0.5879	0.0	0.0
0.5947	0.0	0.0
0.9500	0.0	0.0
1.3198	0.0	0.0
1.5376	0.0	0.0

Table 5-13
AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
5 MW—100% Flow
Series A

Frequency (rad/sec)	$\Delta T_c/\Delta P$ (°F/MW)		Frequency (rad/sec)	$\Delta T_c/\Delta P$ (°F/MW)	
	Real	Imaginary		Real	Imaginary
0.01108	1.2	-2.4	0.01055	-0.2	-4.9
0.01750	0.9	-1.8	0.01773	-0.5	-2.9
0.03026	0.7	-1.2	0.03067	-0.4	-1.8
0.04352	0.3	-1.4	0.04887	-0.4	-1.5
0.07754	0.4	-0.7	0.07785	0.3	-1.6
0.1288	0.2	-1.0	0.1286	0.2	-1.7
0.2182	-0.1	-0.5	0.2024	0.7	-3.4
0.3472	0.1	-0.2	0.3509	1.0	-3.1
0.6050	-0.1	-0.1	0.6086	0.3	-3.0
0.9683	0.0	0.1	0.9768	0.4	-2.6
1.551	0.0	0.2			

Table 5-14
 AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
 5 MW-100% Flow
 Series B
 Sets 1 & 2

Frequency (rad/sec)	$\Delta T_c / \Delta P$ (° F/MW)	
	Real	Imaginary
0.01144	1.4	-5.5
0.01161	2.6	-5.1
0.01789	1.0	-3.2
0.03073	0.6	-2.1
0.03090	0.4	-2.2
0.07780	0.2	-1.2
0.07825	0.3	-1.1
0.1285	0.1	-0.8
0.2171	0.0	-0.4
0.2258	0.0	-0.5
0.3449	0.0	-0.3
0.6145	0.2	-0.4
1.5635	0.1	-0.1

Table 5-15
 AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
 10 MW-50% Flow
 Series A

Frequency (rad/sec)	$\Delta T_c / \Delta P$ (° F/MW)		Frequency (rad/sec)	$\Delta T_o / \Delta P$ (° F/MW)	
	Real	Imaginary		Real	Imaginary
0.01103	1.0	-4.1	0.11756	1.4	-3.8
0.01758	2.1	-2.2	0.01825	1.8	-2.6
0.03030	2.3	-1.9	0.03108	2.4	-1.7
0.04842	1.6	-2.1	0.04931	1.5	-2.2
0.07739	0.9	-1.9	0.07853	0.8	-1.9
0.1284	0.8	-1.0	0.1296	0.2	-1.4
0.2266	-0.3	-1.0	0.2250	-0.2	-1.1
0.3553	-0.5	-0.3	0.3559	-0.3	-0.5
0.9760	-0.2	-0.2	0.6146	-0.3	0
1.5550	-0.2	0.1	1.5707	0.2	0

Table 5-16
 AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
 10 MW-50% Flow
 Series B
 Sets 1 & 2

Frequency (rad/sec)	$\Delta T_c / \Delta P$ (°F/MW)	
	Real	Imaginary
0.01056	1.4	-4.0
0.01085	1.6	-4.0
0.01726	1.7	-2.0
0.02995	2.2	-1.6
0.02998	2.2	-1.6
0.04020	1.7	-1.8
0.07668	1.0	-1.8
0.07746	1.6	1.2
0.1271	-0.4	-1.5
0.2064	-0.0	-1.1
0.2095	0.1	-1.0
0.3333	-0.4	-0.8
0.5855	-0.3	-0.2
0.5974	-0.3	-0.2
0.9491	-0.1	0.1
1.5221	-0.1	0.2
1.5938	0.1	0.0

Table 5-17
 AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
 10 MW-100% Flow
 Series A

			Set* 2		
Frequency (rad/sec)	$\Delta T_c / \Delta P$ (°F/MW)		Frequency (rad/sec)	$\Delta T_c / \Delta P$ (°F/MW)	
	Real	Imaginary		Real	Imaginary
0.01071	0.8	-5.1	0.01128	0.8	-4.9
0.01729	0.0	-2.7	0.01773	0.2	-2.8
0.02990	0.2	-1.2	0.03046	0.4	-1.3
0.04794	0.6	-0.8	0.04866	0.6	-1.0
0.0767	0.6	-0.8	0.07803	0.6	-0.9
0.1273	0.5	-0.7	0.1291	0.5	-0.7
0.2099	0.1	-0.6	0.2125	0.0	-0.6
0.3354	0.0	-0.4	0.3426	-1.2	-0.8
0.5891	-0.2	-0.3	0.6003	-1.6	-0.8
0.9511	-0.2	-0.1	0.9690	-0.3	0
1.5223	-0.1	0.2	1.5525	-0.2	0

*Set 1 values were taken equal to zero.

Table 5-18
 AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
 10 MW-100% Flow
 Series B
 Set 1 & 2

Frequency (rad/sec)	$\Delta T_c/\Delta P$ (°F/MW)	
	Real	Imaginary
0.01665	0.5	-2.4
0.01667	0.4	-2.4
0.02964	0.5	-1.3
0.04785	0.4	-0.7
0.04795	0.5	-0.8
0.07706	0.5	-0.6
0.1287	0.4	-0.6
0.1979	0.2	-0.6
0.3281	0.1	-0.5
0.3299	0.0	-0.4
0.5856	-0.2	-0.2
0.9562	-0.3	-0.1
0.9607	0.0	-0.2
1.5441	-0.3	0.0

Table 5-19
 AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
 15 MW-60% Flow
 Series A

Frequency (rad/sec)	$\Delta T_c/\Delta P$ (°F/MW)		Frequency (rad/sec)	$\Delta T_c/\Delta P$ (°F/MW)	
	Real	Imaginary		Real	Imaginary
0.0111	1.5	-4.0	0.01105	1.3	-3.8
0.01764	1.2	-2.5	0.01751	1.2	-2.2
0.04886	1.6	-1.5	0.03039	1.7	-1.4
0.07807	1.1	-1.5	0.04869	1.6	-1.4
0.1290	0.4	-1.3	0.07789	1.0	-1.5
0.2196	-0.8	-1.0	0.1291	-0.5	-1.4
0.3477	-0.3	-0.6	0.2147	-0.1	-0.9
0.6052	-0.3	-0.2	0.3440	-0.2	-0.7
0.9729	-0.1	0.1	0.6028	-0.1	-0.3
1.5545	0.2	0.3	0.9726	-0.2	-0.1
			1.5612	-0.0	0.1

Table 5-20
AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
15 MW-75% Flow
Series A

Frequency (rad/sec)	$\Delta T_c/\Delta P$ (°F/MW)		Frequency (rad/sec)	$\Delta T_c/\Delta P$ (°F/MW)	
	Real	Imaginary		Real	Imaginary
0.01126	1.1	-4.1	0.01149	1.7	-2.6
0.01780	1.0	-2.3	0.01807	1.2	-1.4
0.03055	1.0	-1.1	0.03095	1.5	-1.4
0.04871	1.2	-1.1	0.04917	1.2	-1.2
0.07759	0.9	-1.1	0.07847	0.9	-1.1
0.1284	0.5	-0.7	0.1295	0.5	-1.0
0.2050	0.2	-0.6	0.2143	0.2	-0.8
0.3351	-0.1	-0.3	0.3434	0.0	-0.6
0.5926	-0.1	-0.1	0.6014	-0.2	-0.3
0.9597	-0.1	-0.1	0.9723	-0.2	0.0
1.5422	0.0	0.0	1.5545	-0.1	-0.1

Table 5-21
AVERAGE COOLANT TEMPERATURE TO POWER TRANSFER FUNCTIONS AT
17 MW-75% Flow
Series A

Frequency Rad/sec)	$\Delta T_c/\Delta P$ (°F/MW)		Frequency (rad/sec)	$\Delta T_c/\Delta P$ (°F/MW)	
	Real	Imaginary		Real	Imaginary
0.01106	0.6	-3.5	0.01140	0.6	-3.6
0.01748	0.5	-2.2	0.01792	0.5	-2.2
0.0304	0.6	-1.3	0.03079	0.6	-1.1
0.04872	0.8	-0.8	0.04917	0.7	-0.8
0.07787	0.7	-0.7	0.07843	0.7	-0.7
0.1291	0.5	-0.8	0.1295	0.5	-0.8
0.3536	0.0	-0.6	0.2145	0.2	-0.7
0.6137	-0.2	-0.2	0.3428	0.0	-0.5
0.9836	-0.1	-0.2	0.6032	-0.1	-0.3
1.5715	-0.2	0	0.9734	-0.2	0.0
			1.5621	-0.1	0.0

5.3.2.4 Comparison of Measured and Theoretical Oscillator Feedback

The theoretical model described in Section 5.3.2.2 was fitted to the experimental data to determine "best estimates" of the power reactivity coefficient at a constant coolant temperature, α , the fuel temperature frequency response parameters γ and ν , and the reactivity energy coefficient. These parameters were determined by adjusting one or more of the parameters α , γ , or ν in the theoretical model described in Section 5.3.2.2 so that the sum of the weighted squares of the real and imaginary parts of a complex function $h(i\omega)$, that is defined as follows,

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

were a minimum. Note that according to an equation in Section 5.3.2.2, $h(i\omega)$ would be identically zero if there were no statistical or other variations in the data, and if the model were an exact representation of the reactivity feedback.

The measured quantities in the above expression are the feedback kernel, $h(i\omega)$, and the average coolant temperature to power transfer function $u(i\omega)$. The other terms are calculated quantities that depend on the values of five parameters α , γ , ν , C_f , and C_q . In principle, "best" values for all five parameters can be determined by simultaneously minimizing the sum of the residuals defined above with respect to all the parameters. In practice a simpler approach was used.

First of all, the entire quantity $[C_f F_s(i\omega, \gamma_e, \nu_e) + C_q]$ was treated as a known function with known parameters by using estimated values (γ_e and ν_e) for the parameters in the function F_s , while for C_f and C_q calculated values that are in agreement with static test results were used. The indicated quantity was treated as known, rather than an unknown, for the following reasons: the coolant temperature oscillations were generally small, and the last term in the expression for h was thus generally smaller by a factor of 4 or 5 than the measured feedback kernel $h(i\omega)$ and the calculated function $\alpha F_{av}(i\omega, \gamma, \nu)$; there was a rather large uncertainty in $U(i\omega)$ (see Section 5.3.2.5) that might lead, if the quantity $[C_f F_s + C_q]$ were treated as an unknown function to be determined by fitting of the data, to a large uncertainty in the power coefficient α ; in addition, such a treatment simplifies the analysis.

Secondly, it was found during the analyses that even with the above simplifications, the values of α , and of the parameters γ and ν in F_{av} , could not all be simultaneously determined. The reason for this indeterminacy is that F_{av} is approximated quite well, as pointed out in Section 5.3.2.2 by $1/(1+i\omega\tau)$, where $\tau = \gamma + \nu^2/8$. Convergence was easily obtained, however, by treating one of the parameters in F_{av} , say γ , as known, and minimizing the residuals with respect to the other two parameters, α and ν , in F_{av} . The goodness of fit (i.e., the weighted sum of the squares of the residuals), visual comparisons of the fits to the data, and the "best" values of α and the time constant τ were all very insensitive to large changes in the parameter, either γ or ν in F_{av} , which was treated as a known parameter. This is as would be expected in the domain for which $F_{av} \sim 1/[1+i\omega(\gamma+\nu^2/8)]$ is a good approximation. Although γ was the parameter which was generally held constant, consistent results were obtained when ν was treated as the known parameter. During the fitting, a value of $-0.15 \text{ d}/^\circ\text{F}$ was used for C_f and a value of $-0.36 \text{ d}/^\circ\text{F}$ was used for C_q . The value of γ was generally about 12.0, but as indicated above the values of τ and α are relatively insensitive to the value of γ . Although an extensive analysis to determine the sensitivity of α and τ to changes in C_f and C_q has not been performed, some analyses on the 15 MW data indicate the sensitivity is small. For example, a $\sim 25\%$ decrease in C_q caused a $\sim 4\%$ increase in α and a $< 1\%$ change in τ , while a $\sim 25\%$ decrease in C_f caused a $\sim 2\%$ increase in both α and in τ .

The values of the parameters that were determined from the analysis are listed in Table 22. The tabulated Doppler energy coefficient was calculated from the fitted parameters using the expression in Section 5.3.2.2.

Representative fits of the theoretical model to the measured feedback kernel are illustrated in Figures 5-1 through 5-8. The fits are good at all power and coolant flow test conditions. The feedback kernel was consistently reproducible, but the average coolant temperature-to-power transfer functions appeared, because of the small magnitude of the

*The inlet thermocouple in the outer IFA deteriorated early, while the vessel inlet resistance temperature detectors that were used for heat balance determinations in the static tests respond too slowly to be of use in the oscillator experiments. Subsequent to the data reported here, it was determined that another thermocouple, located on the core structure near the core inlet, yields an apparently adequate measure of core inlet coolant temperature. It is planned to report the results of analysis of this data in a future quarterly report.

Table 5-22
CONVENTIONAL OSCILLATOR FITTED DATA BY INDIVIDUAL SET

Power (MW)	Flow (%)	Data Series and set	Power Coefficient (¢/MW)	Time Constant (sec)	Doppler Energy Coefficient (¢/MW-sec)
2	40	A1	-16.2	29.7	-0.56
2	40	A2	-16.8	29.3	-0.59
2	40	B1	-16.0	30.0	-0.55
2	40	B2	-16.3	31.3	-0.53
5	30	A1	-11.6	28.9	-0.41
5	30	A2	-10.9	27.0	-0.41
5	30	B	-11.0	29.0	-0.38
5	50	A1	-10.8	26.1	-0.42
5	50	A2	-11.6	28.2	-0.42
5	50	B	-11.4	28.8	-0.40
5	100	A1	-12.0	30.1	-0.41
5	100	A2	-11.7	26.9	-0.44
5	100	B	-11.3	26.9	-0.42

Table 5-22
CONVENTIONAL OSCILLATOR FITTED DATA BY INDIVIDUAL SET

Power (MW)	Flow (%)	Data Series and set	Power Coefficient (¢/MW)	Time Constant (sec)	Doppler Energy Coefficient (%/MW-sec)
10	50	A1	-8.67	30.9	-0.28
10	50	A2	-8.50	28.2	-0.30
10	50	B	-8.55	28.8	-0.30
10	100	A1	-9.06	28.2	-0.32
10	100	A2	-8.84	27.9	-0.32
10	100	B	-8.85	28.4	-0.31
15	60	A1	-6.26	26.8	-0.23
15	60	A2	-6.38	27.3	-0.23
15	75	A1	-6.26	27.6	-0.22
15	75	A2	-6.33	29.2	-0.21
17.5	95	A1	-5.50	26.8	-0.20
17.5	95	A2	-5.49	26.6	-0.20

Figure 5-1 Power Reactivity Feedback Kernel 2 MW, Series B, Set 1

Figure 5-2 Power Reactivity Feedback Kernel at 5 MW, 30% Flow, Series B

Figure 5-3 Power Reactivity Feedback Kernel at 5 MW, 100% Flow, Series B

Figure 5-4 Power Reactivity Feedback Kernel for 10 MW, 50% Flow, Series B

Figure 5-5 Power Reactivity Feedback Kernel for 10 MW, 100% Flow, Set No. 1, Series A

Figure 5-6 Power Reactivity Feedback Kernel at 15 MW, 60% Flow, Series A

Figure 5-7 Power Reactivity Feedback Kernel at 15 MW, 75% Flow, Series A.

Figure 5-8 Power Reactivity Feedback Kernel at 17.5 MW, 95% Flow, Series A

temperature oscillations, ≤ 3 to 5°F and perhaps because of instrumentation problem, to vary appreciably (see Tables 5-9 through 5-21). To determine the influence of the coolant temperature oscillations on the fitted parameters α and τ , the corrections due to the coolant temperature oscillations were arbitrarily set equal to zero during some of the fits. The resulting fit at 15 MW is compared in Figure 5-6 with the ordinary fit in which coolant temperature effects are included. As the figure indicates, the curves with no coolant temperature effects do not approximate the experimental data at lower frequency as well as the curves in which these coolant effects are incorporated.

The averaged power coefficient and fuel time constant that were obtained from all the data at a given power level are tabulated in Table 5-23 along with the Doppler energy coefficient that was obtained from these parameters using the formula in Section 5.3.2.2. The estimated standard deviations in the fitted parameters include a 5% uncertainty in absolute power level and, somewhat arbitrarily, they include one-half the change introduced in the parameters when the coolant temperature oscillations were set equal to zero. The fitted experimental values are compared with the measured static test results and with calculations in Section 5.3.1.

Table 5-23
AVERAGED CONVENTIONAL OSCILLATOR PARAMETERS

Power (MW)	Power Coefficient ($\text{\$/MW}$)	Time Constant (sec)	Doppler Energy Coefficient ($\text{\$/MW-sec}$)
2	-16.3 ± 0.9	30.1 ± 2.0	-0.56 ± 0.05
5	-11.4 ± 0.7	28.0 ± 2.5	-0.41 ± 0.05
10	-8.8 ± 0.7	28.7 ± 2.5	-0.30 ± 0.04
15	-6.3 ± 0.8	27.7 ± 2.5	-0.22 ± 0.04
17.5	-5.5 ± 0.8	26.7 ± 2.5	-0.20 ± 0.04

5.3.3 Uncertainties in Static Reactivity and Power Measurements

5.3.3.1 Deviations in Reactivity

Different measurements of the same critical configuration (the system conditions for which the reactor is exactly critical) of the reactor show small variations as a result of small variations in temperatures, power, instrumentation, core-reflector configurations, etc. This variation is demonstrated by the 5 MWt data (Table 14 of Reference 2 which has been rearranged in a more convenient form in Table 5-24). In this table the relative reactivity at a constant 760°F coolant temperature and a constant power level is tabulated for a number of measurements having the same nominal power and average coolant temperature conditions.

In one case the estimated standard deviation in reactivity is calculated using the power levels indicated by the thermal heat balances as a basis, while in the other the power levels indicated by the neutron instrumentation are used. As indicated in the table the choice of power level indicator does not appear to have a significant effect on the estimated standard deviation in reactivity.

Based on the above data, the estimated standard deviation in a single measurement of a reactivity change (at power) that is calculated from the difference between two measurements of the critical point is $1.1\sqrt{2}$ cents, or 1.6 cents. This deviation is probably reasonable for reactivity changes which are of the order of 30 cents or less. For larger changes the reflector configuration changes considerably, and the reflector interaction effects noted in Reference 6 may increase the uncertainties. Under the experimental conditions employed here (the coarse reflectors adjacent to fine reflector are always completely raised) the maximum previously⁶ observed interaction effect was an apparent change of $\sim 4\%$ in a total reflector worth of $\sim 1.3\%$. Since the nominal interaction effect would be expected to be less than the maximum, the estimated standard deviation for reactivity measurement at power is taken as ± 1.6 cents or $\pm 3\%$ of the reactivity change, whichever is larger.

Table 5-24
5 MW CRITICAL CHECKS

Measure- ment Number	Date (1970)	Reflector No. 3 Relative Worth (cents)	Average Coolant Temper- ature (°F)	Thermal Power (MW)	Neutron Instru- mentation (MW)	Relative Reactivity at 760°F Coolant and 5.10 MW using thermal power values (cents)	Relative Reactivity at 760°F Coolant and 5.06 MW using neutron instrumentation ^(a) (cents)
1	7/9	59.76	760.8	5.03	5.13	60.20	58.52
2	7/9	59.76	760.4	5.06	5.12	60.04	58.84
3	7/10	61.15	760.8	5.15	5.12	60.15	60.03
4	7/10	59.71	759.9	5.14	5.12	59.28	59.04
5	7/10	59.80	761.7	5.21	5.08	57.63	58.71
6	7/11	57.52	759.8	4.99	4.86	58.94	60.02
7	7/23	58.91	758.7	4.98	5.12	61.00	58.84
8	7/24	59.54	758.5	<u>5.22</u>	<u>4.92</u>	<u>58.85</u>	<u>61.97</u>
				$\bar{u} = 5.10$	$\bar{v} = 5.06$	$\bar{x} = 59.51$	$\bar{y} = 59.50$

Estimated standard deviation for $x = \left[\sum (x_i - \bar{x})^2 / 7 \right]^{1/2} = 1.05 \text{ ¢}$

Estimated standard deviation for $y = \left[\sum (y_i - \bar{y})^2 / 7 \right]^{1/2} = 1.15 \text{ ¢}$

Estimated standard deviation for $u = \left[\sum (u_i - \bar{u})^2 / 7 \right]^{1/2} = 0.095 \text{ MW}$

Estimated standard deviation for $v = \left[\sum (v_i - \bar{v})^2 / 7 \right]^{1/2} = 0.106 \text{ MW}$

(a) Temperature and power reactivity corrections made using a temperature coefficient of -0.50 ¢/°F and a power coefficient of -12.0 ¢/MW . Small changes in these coefficients do not significantly alter the estimated standard deviations.

5.3.3.2 Deviations in Power

The reactor power data that are tabulated in Table 5-24 can be used to obtain a measure of the standard deviation in power. A conservative (i.e., large) estimate of the standard deviation in power due to statistical fluctuations can be calculated by assuming that the true power level in each of the eight measurements in Table 5-24 is exactly the same. Actually the true power level is probably slightly different in each case, but since two inexact instruments (1, thermal heat balance; and 2, neutron instrumentation) are used, its true value cannot be known exactly, even though it is believed to be close to the measured values.

The deviations calculated using the above assumption are listed in Table 5-24 as 0.095 MW for the thermal heat balance values and 0.106 MW for the neutron instrumentation. Although the individual power values calculated from a heat balance appear to be rather randomly distributed about their mean, the neutron instrumentation values are clumped in two sets, one containing 6 points near 5.1 MW and another containing 2 points near 4.9 MW. The estimated standard deviations are essentially the same for both the heat balance and neutron instrumentation values, however, and the estimated standard deviation in power due to statistical fluctuations at 5 MW will be taken as 0.1 MW or 2% of the mean value. At higher power levels the deviation will also be taken as 2% of the mean power since the percentage deviations in power will be essentially the same at all power levels if the percentage fluctuations in flow are the same at all flow rates.

In addition to the uncertainty in power due to statistical fluctuations in instrumentation, etc., there is a possible bias because of uncertainties in the absolute calibration of coolant flow and temperatures, and because of uncertainties in coolant density and heat capacity. This possible bias will be treated as a random error, since its magnitude and direction are unknown. The estimated possible uncertainty due to this effect is $\pm 5\%$ of the mean power level.

For power levels $> 1\text{MW}$ the over-all estimated standard deviation in power level, including fluctuations and possible biases, is presently taken as $\pm 5\%$ of the mean power or $\pm 0.1\text{ MW}$ whichever is larger. Additional measurements may modify this preliminary estimate to some extent, but since this is not a primary contributor to the uncertainty in the Doppler effect (see Section 5.3.1.1), improvement of this uncertainty will not significantly alter the uncertainty in the Doppler effect.

5.3.4 Reflector Temperature Reactivity Coefficients

5.3.4.1 Summary

The SEFOR reflector temperature coefficient of reactivity was measured at power levels of 5, 10, and 15 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 (section 5.3.3.1) in reactivity at power, however, and for all practical purposes the reflector temperature coefficient of reactivity may be treated as zero.

5.3.4.2 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 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 (Section 5.3.3.2) 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 5-25. The data in each set are averages of 3 to 5 successive measurements with the data acquisition system static test program. The long wait between equilibrium conditions was necessitated by the slow response of reflector temperature to changes in gas coolant flow. The time-dependent response of thermocouple No. 56 between steady-state values of 290°F and 149°F is shown in Figure 2 of Reference 2 for a rapid increase from 40% to 100% coolant gas flow at time zero during the 5 MW tests. The response to a decrease in flow was even slower.

The reflector temperature readings shown in Table 5-25 were taken from two thermocouples located near the center of coarse reflector No. 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 5-25 indicate that the core reactivity decreases very slightly with increasing reflector temperature, and that the effect is about $-0.01\text{¢}/^\circ\text{F}$ or less when the temperature is monitored by these particular thermocouples. The measured reactivity changes are less than the estimated (Section 5.3.3) 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.

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

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.

5.3.5 Fuel Rod Emission Scans

As part of the fuel surveillance plan in SEFOR, the gamma emission from radioactive fission products is measured in guinea pig fuel rods that contain 25% fissile plutonium (as compared to the standard fuel rod which contains 18.7 fissile plutonium). The guinea pig rods are removed from the core for inspection after initial operation at 5, 10, 15, 17.5, and 20 MW and are generally reinserted prior to the next reactor startup. The gamma emission rates were measured as a function of axial position along the rod by means of a NaI crystal at the end of a collimator slit. The measured data are obtained as a continuous function of rod displacement and show fluctuations about the mean value at any axial location. The average intensity over a small segment of the rod was corrected for background* and compared with the calculated axial power distribution.

The results of measurements up to 17.2 MW are compared in Figures 5-9 through 5-12 with the calculated axial power distribution that is described in Reference 4. The agreement between the calculated axial power distribution and the measured emission rates at 10 MW is quite good. At higher power levels, and thus at higher fuel temperatures and longer irradiation times, the intensity of the gamma emission rate increased markedly at the axial fuel pellet interfaces in the hotter regions of the rod. This increase is believed to be a result of movement of volatile fission products toward the cooler regions of the pellets as the central fuel temperature becomes hotter. The extent of this apparent migration is illustrated in Figures 5-11 and 5-12 where the measured average axial emission rate differs markedly from the calculated power distribution, and thus from the earlier emission scans at 10 MW. The actual data are illustrated for two cases in Figures 5-13 and 5-14.

Table 5-25
REFLECTOR TEMPERATURE-REACTIVITY MEASUREMENTS

Time (1970)	Nominal Power (MW)	Data Set Number	Average Coolant Temperature (°F)	Reflector No. 3 Position (cm)	Reflector Worth (¢)	Reflector No. 56 (°F)	Thermocouple No. 60 (°F)	Relative ^(a) Reactivity at 760°F (¢)
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	-1.08
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
8/19-1400	15	2	759.8	50.96	-76.19	257	205	0.0
8/29-1055	15	1	759.5	51.42	-66.12	479	395	-1.07
8/29-1740	15	3	759.5	51.46	-77.11	441	362	-1.06

a. Reactivity corrections to a 760°F average coolant temperature were calculated using reactivity coefficients of -0.51, -0.48, and -0.46 ¢/°F at 5, 10 and 15 MW respectively. The relative values were adjusted at each power level so that the average of the values at the cooler reflector temperatures are zero.

*The background correction was approximately 10% of the intensity at the end of the rod for an emission scan at 10 MW.

Figure 5-9 Emission Gamma Scan in Guinea Pig Rod A01 after 10 MW Irradiation

Figure 5-10 Emission Gamma Scan in Guinea Pig Rod A17 after 10 MW Irradiation

Figure 5-11 Emission Gamma Scan in Guinea Pig Rod A17 after 15 MW Irradiation

Figure 5-12 Emission Gamma Scan in Guinea Pig Rod A17 after 17.2 MW Irradiation

Figure 5-13 Emission Scan Data from Rod A17 Inspection at 10 MW

Figure 5-14 Emission Scan Data from Rod A17 Inspection at 17 MW

6. TASK 11 – TRANSIENT TESTS

6.1 PURPOSE

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

6.2 PAST PERFORMANCE

An irradiated fuel shipping cask, rented with the authorization of the AEC, was modified for shipping 1, 2, or 3 SEFOR fuel rods, licensed by the AEC, approved by the Department of Transportation for transport, and shipped to SEFOR. Fabrication and installation of associated equipment for transferring fuel from the SEFOR refueling cell to the cask were completed at SEFOR.

6.3 CURRENT RESULTS

The digital computer code for processing the transient test data was written, and the code was partially debugged.

A simulated transient data tape was made at the SEFOR facility, the data was read by the GE computer in San Jose, and formatted for further processing. The code smooths the data, integrates the smoothed data, and determines the reactivity from the smoothed power data using a kinetic inversion routine.⁵ The smoothing and integrating portions of the code have been checked out with test problems, and they are working properly. A simulated transient tape was generated for testing the kinetic inversion portion of the code. A test showed there is an error in the kinetic inversion portion of the code. The coding is presently being analyzed to determine the source of this error.

7. SEFOR SAFETY AUDIT AND LICENSING

Proposed changes to the Technical Specifications have been sent to the DRL to establish limits for unexplained behavior of reactor parameters and to identify detection steps and criteria for judging the acceptability of continued steady-state reactor operation in the presence of a known loss of fuel cladding integrity. These changes are required by the Technical Specifications before initial operation at 20 MWt. Changes have also been proposed to permit use of the guinea pig rods under the innermost refueling ports throughout the test program as originally intended in the SEFOR design. The present Technical Specifications require that these guinea pig rods be removed for steady-state operations above 17.5 MWt.

Changes to the Technical Specifications, required for 20 MWt operation and initiation of the transient test program, were proposed by GE and reviewed by the DRL. Partial approval of these changes was received in January to permit operation up to 20 MWt. It is expected that full approval, permitting initiation of the transient (FRED) test program, will be received in early February.

Special apparatus was used to determine the isotopic analysis of the cover gas during reactor operation. The results of these tests demonstrated the capability of detecting and measuring fission products in the cover gas due to tramp fissile material in the sodium or plated on surfaces in the vicinity of the core and therefore also demonstrate the capability of detecting failed fuel. Analyses also showed that the reactor cover gas activity indicated by the cover gas monitor is due primarily to ^{23}Ne . At 10 MWt, the ^{23}Ne produced a response of 22 mR/h and ^{41}A produced a response of 4 mR/h, for a total response of 26 mR/h on the cover gas monitor.

A report of test results through 10 MWt which demonstrate that the Doppler coefficient and net non-fuel reactivity coefficient meet the requirements of the Technical Specifications was sent to the DRL. The report included additional information on reactivity coefficients and stability of the reactor.

Test data were obtained to determine the effect of sodium temperature on the calibration of the wide range monitor. The results of these tests showed that the calibration factor (WRM/Heat Bal), decreases by about 2.7%/100°F decrease in sodium temperature at constant power. The calibration factor also decreases by about 2% for each reflector segment lowered. These effects will be accounted for in determining the allowable high flux trip setting for all reactor operating conditions.

Operating data for the nitrogen cooling system were reviewed, and it was found that temperatures in two areas were higher than expected though not at unsafe levels. Changes were proposed to increase cooling of the upper primary sodium equipment cell and of the upper reactor vessel support region. These changes were made and test results showed that the desired reduction in temperatures had been achieved.

An audit inspection was made at the site in January to verify compliance with the Technical Specifications in the areas of surveillance tests and safety system trip settings. All items but one were found to meet the requirements of the Technical Specifications. The one item was the operating sodium level in the auxiliary secondary expansion tank. It was approximately 10 gallons higher than the amount permitted (Table 3.1-2 of the Technical Specifications). This would increase the possible loss of sodium in the auxiliary secondary loop from 14 gallons to 24 gallons before causing a reactor scram from the low-level trip. This amount of sodium would not cause a significant change in the consequences of such an accident. When the discrepancy was discovered, the reactor was shut down as required by the Technical Specifications, the excess sodium was removed, and reactor operation was resumed.

Modification requests were reviewed with regard to effects on plant safety and requirements of the operating license. Safety evaluations were provided where required.

The results of the natural circulation test were reviewed and found to be satisfactory. A steady-state reactor heat load of 600 KWt was removed under simulated conditions of plant power loss and diesel generator failure with a maximum sodium temperature of 650°F. This heat load corresponds to the estimated decay heat load about 3 minutes after reactor shutdown following infinite irradiation at rated power. Higher heat loads could be removed at higher sodium temperatures. The 650°F test value provides more than adequate margin with respect to the safety limit of 1050°F.

Natural circulation flow rates in both primary loops were about 40% of the predicted values, while flow rates in both secondary loops were 80 to 90% of the predicted values. The reduced flow in the primary loops is probably due to

effects of very low flow rates on pressure losses and thermal heads calculated for the reactor vessel under natural circulation conditions. The reduced flow does not have a significant effect on heat removal capability as evidenced by the above-mentioned test results.

The tests also demonstrated that flow stagnation in the main primary loop will not be a problem. This was demonstrated by allowing the temperature of the main secondary loop to increase until natural circulation flow in the main primary loop decreased to zero. Forced circulation was maintained in the auxiliary loop during this test. Under these conditions the main primary flow remained at zero until the reactor sodium temperature increased sufficiently to start flow in the loop. Flow then increased to about 50 gpm for a few minutes and then returned to zero. Repetition of this cycle maintained sodium temperatures at safe levels in the reactor vessel and throughout the loop.

A report of these test results is being prepared and will be sent to the DRL.

8. SEFOR SUPPORT

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

During November, a planned outage for installation of the argon cell replacement argon purification system was completed. Design and fabrication of the NaK bubbler system is covered in previous SEFOR Quarterly Reports, GEAP-10010-24 and -25. Upon completion of the modification, the refueling cell was placed on argon and the reactor systems prepared for operation. Initial satisfactory operation of the replacement purification system was demonstrated. Reactor operation continued through December and January, and the balanced oscillator tests were completed with reactor operation at power levels up to 17.5 MW. Successful operation at 20 MW was achieved on January 19, 1971.

Increased reactor cover gas leakage was noted during operation over an extended period of time. Inspection of the reactor head revealed that the leakage was occurring between the outer head and the vessel flange. A series of diagnostic tests were performed at the site to determine the cause of the leakage. Dial indicators were mounted to measure reactor flange movement during heat-up and cool-down cycles. Measurements made during the test and results of subsequent evaluations indicate the reactor flange and outer head are rotating slightly during heat-up and cool-down cycles. The rotation may be sufficient to allow the increased cover gas leakage. The following steps were undertaken to correct the leakage problem:

- a. The tension in the head bolts for the outer and inner heads was checked and the bolts were retensioned. This greatly reduced the leakage around the head. Subsequent operation has indicated only a very low leak rate from around the outer head.
- b. Design work was initiated on a backup seal design which would cover the annular gap between the reactor flange and the head bolts.

Engineering was completed for modifications to improve the nitrogen cooling system performance. Cooling ducts in the primary equipment zone were modified to provide more cooling in the upper regions. A remote actuator was provided in the nitrogen return damper from the reflector and primary shield regions. The actuator allows remote nitrogen flow adjustments to be made through the upper shield region. Results of tests performed following startup indicated that the nitrogen cooling was improved and that temperatures in the upper shield region were reduced.

The order was placed for procurement of the argon blower to replace the existing cooling fan for the refueling cell. The replacement blower is an axial flow unit which does not contain a shaft seal. The shaft seal on the existing blower has been a source of difficulty for some time. The design conditions for the replacement blower are as follows:

Blower:

Type	Ducted Axial Fan
Fluid	Argon Gas
Gas Temperature	70-150°F
Capacity	3680 cfm at 6 inches H ₂ O Static Head
Design Pressure	10 psig

Motor:

Type	Brushless
Electrical Rating	220V, 3 phase, 60 Hz
Power Rating	Not less than 125% of maximum brake horsepower required by blower
Insulation	Class H insulation. Temperature rise designed for Class A

A design modification to the fuel rod assembly-disassembly equipment in the refueling cell was completed which will accommodate remote removal of the FRED B₄C rod from the FRED extension shaft. New parts required for the modification were fabricated and sent to the SEFOR site.

Safety audits during 1970 regarding refueling cell access suit pressure transients determined that a suit exhaust system other than the current high vacuum type should be sought. Rescue of suited operators from the refueling cell has been carried out once following excessive suit depressurization and once following overinflation. Suit trials are currently in progress at the plant to determine if an enlarged exhaust hose between the suit and the air zone outside the cell will have adequate flow capacity utilizing only the suit's internal pressure. If an umbilical of manageable size will handle current flow rates, the existing modulating valves and high vacuum exhaust system will be replaced.

Scope drawings were completed on systems for recovery from a failure of the refueling equipment while handling a fuel element or of the reactor vessel closure. These recovery systems provide the means for placing a fuel element in a shielded enclosure or returning the vessel closure to the reactor opening, allowing man-entry for contact maintenance of the refueling equipment. The retrieval designs rely upon equipment external to the refueling cell to be placed into operation in the event of a malfunction which could not be repaired by normal access means.

9. SEFOR OPERATIONS

9.1 EXPERIMENTAL PROGRAM

During this reporting period balanced oscillator experiments (BOE) of the first and second kind were performed at 13, 15, and 17 MW. The BOE tests were originally planned at 5, 10, 15, and 19 MW. The BOE tests at 13 and 17 MW were added to augment the test sequence while awaiting permission from the Atomic Energy Commission's Division of Regulation and Licensing to proceed to full power. The BOE of the second kind at 10 MW was also performed. (The BOE of the first kind at 10 MW was completed prior to the scheduled shutdown.) In the BOE of the first kind the reactivity and the primary and secondary loop sodium flow rates are oscillated in a sinusoidal manner to achieve constant inlet and outlet coolant temperatures. The BOE of the second kind entails the simultaneous oscillation of the oscillator poison rod, main primary and main secondary flows in such a manner that the reactor power oscillations are balanced out or nulled.

Both manual techniques and the automatic balanced oscillator (ABO) were used in obtaining balanced conditions. The ABO did not function satisfactorily on experiments of the first kind due to noisiness or frequency response of the temperature sensing devices. The available vessel exit thermocouples exhibit a high-frequency noise (believed to be a result of flow mixing) and the resistance temperature detector (RTD) response is much too slow to be used in balancing except at the lowest frequency (0.00165 Hz).

In the experiments of the first kind the amplitudes and phase angles of oscillation for the coolant loops were not as predicted. In the experiments of the second kind, reactivity amplitudes much smaller than predicted (1 to 3 cents) and correspondingly only smaller power oscillation (~2.5 to 8%) could be balanced within the allowable flow oscillation at the test conditions. The data are being analyzed and the source of the discrepancy between prediction and measurement is being evaluated.

At 5 MW a BOE of the second kind was attempted using both manual and ABO control systems. From the results it was concluded that the feedback effects were not of sufficient magnitude to justify the tests at this level. During conventional oscillator tests no significant coolant temperature oscillations occurred at 5 MW, thus no BOE data of either kind were taken at 5 MW.

After completion of the balanced oscillator experiments, additional conventional oscillator tests with large reactivity amplitudes were performed at 15 MW and 560°F. The purpose of these tests was to investigate the possibility of fuel-cladding interaction which might produce unexpected reactivity effects. The temperature and power conditions were chosen to minimize the fuel-cladding gap. Data were taken at each frequency with the usual $\pm 5\%$ power oscillation and at the maximum oscillation permitted within the Technical Specifications. The amplitude and phase angle of the transfer function showed no significant difference for the two cases.

Absolute alpha counting of ^{239}Pu , ^{240}Pu , ^{238}U and ^{235}U source foils was initiated at the University of Arkansas. The alpha counting is part of the absolute fission determination to be performed using track-etch techniques and is necessary to make a precise weight determination of the source foils. Indications are that some self-absorption of alphas is occurring which could be due to the presence of material other than PuO_2 in the deposit. The uranium counting is complicated by the background in the low-geometry system being used.

Test Procedure II-11 (Fission Rate Determination) has been drafted and reviewed, and is being revised.

Reactivity loss due to fuel burnup has been determined as 12 to 16 cents for the interval September 3, 1970, to January 15, 1971. The determination of burnup has been inferred by analyzing reactor conditions and noting the changes in reflector positions as the power history of the core increases. The burnup has also been estimated based on the power history and the reactivity worth of the isotopic production — destruction in the core. The results of these analyses are summarized in Table 9-1.

The modified fuel leak check chamber has been installed in the fuel inspection area and has been checked out in preparation for the testing of the new fuel to be shipped from Nuclear Fuel Services. This modification allows hot leak checking of newly received fuel.

The five ^{238}U fission chambers, part of the transient detector system, were exposed to a neutron source at Vallecitos Nuclear Center by use of the Nuclear Test Reactor. The intent of the test was to ensure that all the chambers responded to neutrons. Voltage plateaus and current outputs were determined for each chamber. The results of these tests are summarized in Table 9-2. Note that the chambers were operated at 600 Vdc, while in actual operation 1000 Vdc has been recommended to ensure complete saturation and adequate response to the transient flux.

Table 9-1
SEFOR BURNUP: CALCULATED AND MEASURED

Date	EFPH	Calculated Burnup (¢)	Measured Burnup (¢)
9/3/70	190.671	5.91 ±1.2	5.91±1.2 ^(a)
10/28/70	360.618	11.183±2.2	9.83 ^(b)
12/6/70	426.974	13.24 ±2.6	10.4 ±6.0
1/17/71	760.719	23.58 ±4.7	16.51±4.9

- a. Measured assumed to be equal to calculated since previous burnup was immeasurably small compared to reactivity changes in fuel rearrangements prior to this period.
- b. No uncertainty can be ascribed to a single measurement. Other measured uncertainties are three standard deviations of multiple measurements.

Table 9-2
SEFOR ²³⁸U FISSION CHAMBER
RESPONSE TEST DATA AT N.T.R.

Detector Serial No. (a)	Collector Current (Amps) at Indicated Power				Time at Power
	30KW	3 KW ^(b)	0 KW ^(c)	0 KW ^(d)	
943	4.1×10^{-5}	-	2.8×10^{-6}	5.0×10^{-8}	13 minutes
944	4.2×10^{-5}	-	2.5×10^{-6}	4.5×10^{-8}	16 minutes
945	4.2×10^{-5}	-	5.1×10^{-6}	-	25 minutes
946	-	5.8×10^{-6}	1.9×10^{-6}	4.7×10^{-9}	13 minutes
947	4.0×10^{-5}	-	2.6×10^{-6}	-	16 minutes

All chambers were installed at the centerline of the Vertical Test Facility for response test. This required testing on a one-at-a-time basis. Detectors 943 and 945 were tested on 11/11/70 and Detectors 944, 946, and 947 on 11/12/70.

- a. All detectors operated with + and - H.V. set for 600 VDC.
- b. Detector 946 was tested at 3 kW to reduce activation levels as required to effect detector changeover. (only two sets of cables were available during testing).
- c. Readings in this column were taken immediately following reactor shutdown with chambers still in the vertical facility.
- d. Readings in this column were taken after chambers were removed from vertical facility following reactor shutdown.

After the response check at Vallecitos, the chambers were returned to SEFOR for installation during the outage scheduled for installation of the new NaK bubbler. Three chambers were installed, only two of which are connected through penetrations. Two SEFOR Energy Probes (SEP) were also installed during this outage. The SEP design called for relatively close-tolerance fitting inside a detector housing identical to the transient gamma detector housing. When assembly was attempted, it was found that the ceramic forming the outside insulator of the SEP's was so badly

distorted that the SEP's could not be inserted in the housing. The housing was modified by cutting the top section off and making the inside diameter of the housing tube large enough so that the SEP's could be installed inside. The housing was reassembled by inserting a splicing sleeve inside the tube and fixing the sleeve with set screws which were then back-welded to prevent inadvertent backing out after installation. It was also necessary to extend the signal lead wire from the SEP's since the as-received length was not sufficient to permit connection outside the detector housing. After modification of the SEP housing, the SEP's were installed in the 270 degree (i.e., west) gamma detector hole. The SEP's can be monitored at the R&D patch panel in the control room and were monitored with the fission chambers during initial power rise and periodically thereafter.

Plans called for installation of all the transient detectors, fission and gamma chambers. On checking the gamma chambers, it was found that the chambers' high voltage was shorted to the detector housing. Disassembly and inspection revealed that the sprayed-on ceramic insulation material which should prevent the high voltage from shorting to the detector housing had been either abraded or knocked off during shipping, which exposed the bare metal in numerous places on the tri-axial cable and chamber. Since some modification of the gamma transient detectors was necessary they were not installed.

During the initial power rise to the various Balanced Oscillator Test conditions, the transient detector system (two 238U fission chambers and two energy probes) was monitored at 2 MW power intervals. The fission chambers and energy probes responded linearly with power as expected. The energy probes are essentially a fueled thermocouple. The resulting temperatures in the energy probes were as expected. The fission chambers produced a current about eight times larger than design calculations predicted. The Transient System will be monitored periodically until the transient tests.

The Fast Reactivity Excursion Device (FRED) rupture discs from all lots to be used on the system during reactor tests were tested to failure, and all obsolete rupture discs were destroyed to avoid inadvertent use. Results of the tests were:

Expansion Chamber		
Lot No.	Rating (psig)	Rupture Pressure (psig)
260494	231	235
260494	231	225
267905	208	195
267905	208	193
Accumulator		
263712	1975	2225
263712	1975	2050
267904	1853	1875
Transfer Chamber		
80528	2313	2375
263713	2400	2400
267906	2366	2425

On the first disc tested, from lot 263712, the rupture pressure was significantly different from the specification (2225 vs 1975). Examination of the failed disc showed the dome to be off center, which may have given the material additional support. The disc was not carefully examined prior to testing. A second disc from the lot was tested and found to be as specified.

The irradiated fuel shipping cask was moved to the refueling cell vertical transfer port to check out the cask and fuel transfer procedures. With minor modifications of cask and procedures the checkout was satisfactory. It is intended to transfer both the 17.5 MW surveillance rod (A17) and the first transient surveillance rod (A16) in a single shipment. This shipment will be made after the subprompt transient tests from 15 MW.

Transient testing preparation continued throughout the quarter. The Familiarization Transient Test Procedure IV, is issued and approved. The Sub-Prompt Transient Test Procedures V and VI, have been distributed for review and comment. Ancillary preparations associated with the transient testing continued. Specific items involved in this category are: assembly-disassembly station modification to handle the FRED slug, test procedure preparation to measure initial worth of FRED slugs, and in-cell storage of irradiated orifices removed to install IFA's.

Test procedure preparation for the anticipated Core II tests was initiated during the period.

Work was initiated on the identification and scoping of tasks for plant decommissioning.

9.2 OPERATIONS AND MAINTENANCE

Operation Statistics (11/2/70 to 1/15/71)

	Quarter	Total to Date (1/15/71)
Hours Critical	625	2,361
Power Generated, MWh	7,532	15,476
Startups	17	419
Scrams	8	106
Maximum Power Level, MWt	17.5	17.5
Operating Efficiency, %	42.8	38.1

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

Slightly over half of the power generated to date was generated in the last quarter.

The longest continuous reactor operation to date was achieved over the period from December 27, 1970, through January 3, 1971. The reactor was critical for 153.5 hours. The previous longest run was 79 hours in October 1970.

The length of the operation was determined by the requirement for shutdown each Sunday to perform the required surveillance tests.

When the results of the quarterly pressure decay test of one containment penetration (auxiliary secondary sodium pipe — Mk 4) fell outside of acceptable limits, an outage scheduled for the latter part of November was initiated on November 2, 1970. Installation, modification, maintenance, and preventive maintenance were in progress during the remainder of November. The more significant of these items included:

- Installation of the replacement purification system for the Refueling Cell Argon,
- Installation of man-access suit auxiliary exhaust piping and new air supply and exhaust hoses,
- Installation of neutron shielding (boral sheets) in primary sodium equipment area,
- Replacement of the second reactor vessel sodium level probe assembly,
- Extension of nitrogen cooling ducts into upper regions of primary sodium equipment area,
- Installation of a remotely controlled damper actuator for primary shielding nitrogen cooling return, and
- Installation of the energy probe and fission transient detector system.

After completion of maintenance and installations, pressure decay and cell-to-cell leak tests were performed, the cell atmospheres were changed to argon and nitrogen, one of the two replacement NaK bubblers in the argon purification loop was filled with 1500 pounds of NaK and a trial run was made with the irradiated fuel shipping cask and refueling cell vertical transfer port shielding.

Reactor operation was resumed on December 3, 1970, but progress on program testing was limited during December because of the overhaul required on a freon compressor, malfunction of an electrical bus overcurrent relay lock-out device and interface and noise problems with the automatic balanced oscillator control system. Operation was further delayed by the necessity of obtaining analyses on additional sodium samples from the primary system when the results of analyses on the previous sample indicated an excessively high carbon content. The source of the high carbon content was determined to be external contamination of the sample. Analyses of the latter samples confirmed that the carbon content had remained in the 20 to 30 ppm range, unchanged from that obtained at the time sodium was charged into the systems in 1968. For the following 3 weeks reactor and plant availability was nearly 100%.

A total of 189 equipment malfunctions were corrected, distributed as follows:

Mechanical	70
Electrical	56
Instrumentation	<u>63</u>
	189

Significant malfunctions corrected included:

- Containment penetration
- Sodium piping heater circuits and thermocouples
- Reactor building ventilation valve
- Reactor vessel head stud tension
- Fuel rod disassembly station
- Reflector scram solenoid valves
- Distillation chamber gate valve
- Nitrogen blower motor windings
- Remote manipulator
- Nitrogen blower outlet RTD
- Electrical breaker overcurrent lock-out
- Argon compressor
- Freon compressor
- Argon vent vacuum pump
- 125 Vdc battery charger
- Freon piping
- Electrical transformer insulation
- Nitrogen radiation monitor
- Sodium piping heater circuit volt-pacs
- Reactor building air handling unit
- Air blast cooler door actuators

A total of 45 planned equipment modifications were completed including the following significant items:

- Hoisting structure for removal and installation of NaK bubblers
- Replacement argon purification loop
- Replacement man-access suit hoses
- Man-access suit auxiliary exhaust piping
- Neutron shielding — primary sodium equipment area
- Nitrogen cooling ducts and damper actuator
- Energy probe housing and connections

- Reactor vessel inner and outer head thermocouples
- Replacement reactor sodium level probe
- Freon compressor motor protection
- Thermocouples for balanced oscillator experiment
- Sodium piping heater circuits — gas phase temperature monitor
- Freon unit test instrumentation
- 480V bus overcurrent trip annunciator
- Fuel rod leak check chamber
- Main secondary pump power supply cooling
- Irradiated core orifice shielded storage
- Breathing air vacuum pump filter

Gamma ray spectrometry was used to identify and measure fission gas components of the reactor cover gas at the 10 MW power level. The results indicated a level of xenon and krypton commensurate with tramp uranium in the systems.

Routine environmental surveys were conducted to verify that no environmental radioactivity increase attributable to reactor operation occurred.

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