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**SOUTHWEST EXPERIMENTAL
FAST OXIDE REACTOR
DEVELOPMENT PROGRAM
TWENTY-THIRD QUARTERLY REPORT
NOVEMBER 1969-JANUARY 1970**

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

BREEDER REACTOR DEVELOPMENT OPERATION

GENERAL ELECTRIC

SUNNYVALE, CALIFORNIA

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SOUTHWEST EXPERIMENTAL FAST OXIDE
REACTOR DEVELOPMENT PROGRAM

Twenty-Third Quarterly Report
November 1969 – January 1970

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GENERAL ELECTRIC
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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, 1969 through January 31, 1970. The work was performed under contract AT(04-3)-540 with the United States Atomic Energy Commission.

1. INTRODUCTION

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

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

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

Phase I --Preoperational research and development,

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

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

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

1. "Southwest Experimental Fast Oxide Reactor Development Program, First Quarterly Report, April--June 1964," GEAP-4594 (July 1964).

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

12. "Southwest Experimental Fast Oxide Reactor Development Program, Twelfth Quarterly Report, February–April 1967," GEAP-5498 (May 1967).
13. "Southwest Experimental Fast Oxide Reactor Development Program, Thirteenth Quarterly Report, May–July 1967," GEAP-5533 (August 1967).
14. "Southwest Experimental Fast Oxide Reactor Development Program, Fourteenth Quarterly Report, August–October 1967," GEAP-5561 (November 1967).
15. "Southwest Experimental Fast Oxide Reactor Development Program, Fifteenth Quarterly Report, November 1967–January 1968," GEAP-5595 (February 1968).
16. "Southwest Experimental Fast Oxide Reactor Development Program, Sixteenth Quarterly Report, February–April 1968," GEAP-5629 (May 1968).
17. "Southwest Experimental Fast Oxide Reactor Development Program, Seventeenth Quarterly Report, May–July 1968," GEAP-5673 (August 1968).
18. "Southwest Experimental Fast Oxide Reactor Development Program, Eighteenth Quarterly Report, August–October 1968," GEAP-5705 (November 1968).
19. "Southwest Experimental Fast Oxide Reactor Development Program, Nineteenth Quarterly Report, November 1968–January 1969," GEAP-5754.
20. "Southwest Experimental Fast Oxide Reactor Development Program, Twentieth Quarterly Report, February–April 1969," GEAP-10010 (May 1969).
21. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-First Quarterly Report, May–July 1969," GEAP-10010-21 (August 1969).
22. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Second Quarterly Report, August–October 1969," GEAP-10010-22 (November 1969).

In addition, the following topical reports have been issued:

1. Novak, P. E., and Asamoto, R. R., "A Survey for a High Temperature Sensor for SEFOR," GEAP-4903 (July 1965).
2. Quinn, E. P., "Vibration of SEFOR Fuel Rods in Parallel Flow," GEAP-4966 (September 1965).
3. Johnson, M. L., "A Description of Fuel and Other Core Materials Procured for SEFOR Critical Experiments in ZPR-III," GEAP-5133 (December 1965).
4. Protsik, R., "Epithermal Capture and Fission Distribution in a Fuel Rod—The Computer Code REAX," GEAP-4973 (April 1966).
5. Novak, P. E., and Asamoto, R. R., "An Out-of-Pile Evaluation of W-Re Thermocouple Systems for Use to 4700°F in Mixed Oxide," GEAP-5166 (May 1966).
6. Pfisterer, G. R., "SEFOR Experimental Program Planning, Vols. I and II," GEAP-5092 (August 1966).
7. Morrell, R. P., "SEFOR β/ℓ Measurements by Noise Analysis Techniques," IDO-17210 (August 1966).
8. Williams, R. F., and Polomik, E. E., "Pressure Drop and Dye Tests with Water on the SEFOR Prototype Fuel Bundle," GEAP-5129 (February 1966).
9. Craig, C. N., Ryer, C. M., and Thompson, M. L., "Plutonium-Uranium Mixed Oxide Fuel Pellet Fabrication Development for the Southwest Experimental Fast Oxide Reactor," GEAP-5285 (October 1966).
10. Polomik, E. E., and Williams, R. F., "Coolant Surface Turbulence Study, SEFOR Reactor Vessel," GEAP-5099 (December 1966).
11. Wilkinson, L., "Use of Neutron Filters for Fast Reactor Fuel Irradiations in a Thermal Core," GEAP-5035 (April 1967).
12. Wilkinson, L., et al., "Devices for Cyclic and Transient Power Control in a Steady-State Neutron Flux Gradient," GEAP-5036 (May 1967).

13. Reynolds, A. B., and Steward, S. L., "Analysis of the SEFOR Mockup Critical Experiments in ZPR-3," GEAP-5294 (March 1967).
14. Wilkinson, D., "Weighted Doppler Analysis Code—WEDOP," GEAP-5513 (October 1967).
15. Craig, E. R., Fulton, R. H., Damon, J. J., "Design and Testing of Solenoid and Instrument Lead Connector for Use in SEFOR Reactor Vessel," GEAP-5307 (March 1967).
16. Asamoto, R. R., Perry, K. J., and Zebroski, E. L., "The Effective and Diametral Thermal Expansion of UO_2 Under a Radial Temperature Gradient," GEAP-5284 (October 1966).
17. McVean, R. L., Weitzberg, A., Kremser, J. A., Leridon, A. M., and Long, J. K., "Critical Studies of the Southwest Experimental Fast Oxide Reactor (SEFOR) in ZPR-3," ANL-7348.
18. Polomik, E., "Fluid Flow and Vibration Tests, 1/2-Scale Model of SEFOR Reactor Vessel," GEAP-5440 (April 1967).
19. Field, J. H., Johnson, M. L., and Novak, P. E., "An Evaluation of the Effect of Design and Operating Variables on SEFOR Fuel and Fuel Cladding," GEAP-5309 (December 1967).
20. Noble, L. D., "Final Specifications for the SEFOR Experimental Program," GEAP-5576 (April 1968).
21. Russell, C. E., "Flow Distribution Measurements in a 3-Scale Model of the SEFOR Fuel Bundle," GEAP-5184 (May 1968).
22. Cochran, J., "SEFOR Core Clamping Model Tests, One-Sixth Segment," GEAP-5465 (May 1968).
23. Asamoto, R. R., and Novak, P. E., "An In-Pile Evaluation of W-Re Thermocouple Systems for Use to 4700°F in SEFOR Fuel," GEAP-5466 (October 1968).
24. Cochran, J. T., "SEFOR Fuel Assembly Sodium Flow Tests," GEAP-5466 (September 1968).
25. Cochran, J. T., "SEFOR Fuel Rod Sodium Thermal Shock Tests," GEAP-5467 (August 1968).
26. Craig, E. R., "SEFOR Instrumented Fuel Assembly Design and Development," GEAP-5615 (April 1968).
27. Steamer, A. G., Brynsvold, G. V., and Franckx, L. C., "Fuel Handling and Other Tests in the SEFOR Refueling Cell Mockup," GEAP-5701 (March 1969).
28. Johnson, M. L., Becker, R. A., Haar, J. M., "SEFOR Fuel Handling," GEAP-13565 (December 1969).

2. SUMMARY

2.1 COMPLETED TASKS

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

TASK	TITLE
1.1	Specifications for the Reactor and Experimental Program
1.2	Final Specifications of the Experimental Program
2.0	Experimental Program in ZPR-III
3.0	Fuel Development

4.0 Special Instrumentation

5.0 Steady-State and Transient Characteristics of the Core Assembly

6.0 Reactivity Controls Required for Experimentation

7.0 Fuel Handling

8.0 Core Design

2.2 TASK 9—FUEL FABRICATION

Negotiations with Nuclear Fuel Services (NFS) were begun concerning fabrication of additional and replacement fuel rods.

The fourth and fifth Instrumented Fuel Assemblies (IFA) were completed and delivered to the reactor site.

Two IFA's were installed in the reactor core January 25, 1970. The assemblies were checked out and were operating in accordance with the design objectives.

IFA fabrication sequence photos are included in this report.

2.3 TASK 10—START-UP TESTING

The reflectors were calibrated in Assembly I-E at a temperature of 650°F. As with the calibration at 350°F in the same assembly,³ these measurements indicated asymmetric control strengths that were consistent with an asymmetric core loading pattern. A comparison of the worths of the individual reflector segments measured at the two different temperature (and thus reflector) conditions clearly illustrates the extent (up to 40%) of the influence of reflector position on the worth of adjacent reflectors.

The measured flow and pressure coefficients of reactivity in SEFOR are essentially zero. The reactivity effects that are associated with changes over the normal operating range of reactor vessel hydraulic pressure and up to 80% of maximum primary coolant flow are the order of the standard deviation (0.5 cent) in reactivity difference measurements.

The temperature reactivity data between 350°F and 760°F in Assembly I-E shows good agreement with the reactivity effects calculated using the predicted Doppler $T(dk/dT)$ of -0.0082 and the combined density-expansion coefficient of -0.36 cent/°F.

Fission rate distributions for U-235, U-238, and Pu-239 obtained from activation measurements in Assembly I-D were compared with diffusion calculations. Although the agreement was generally good, when normalized at the core center the fission rates of all three isotopes near the core boundaries about 30% higher than the calculated values. This indicates a flatter power distribution than predicted by diffusion calculations.

Noise measurements were performed at different power levels in Assembly I-D. Due to saturation of the detectors above 1 kW reactor power and interference with induced background at very low reactor power, a reliable value of λ/β could only be obtained from the one measurement of ~20 watts reactor power. The experimental result,

$$\lambda/\beta = 0.205 \times 10^{-3} \text{ sec, is about 11\% higher than the calculated value,}$$

$$\lambda/\beta = 0.184 \times 10^{-3} \text{ sec.}$$

2.4 TASK 11—TRANSIENT TESTS

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

A draft of the license application including the structural and thermal analysis has been completed, reviewed and the final copy is being prepared.

The design of the fuel transfer equipment and the shielding associated with the fuel transfer has been completed. Fabrication of approximately 90% of the fuel transfer equipment has been completed.

2.5 SEFOR SAFETY AUDIT

Three trips were made to the site for audit purposes this quarter. Procedures and special equipment used for IFA installation were observed in use. License test procedures and standard operating procedures were audited.

2.6 SEFOR OPERATIONS

The zero-power phase of the experimental program was completed, and a planned outage was initiated on November 1, 1969 to perform maintenance and modifications prior to the ascension to power. During this period, an audit of the sodium coolant system construction radiography was completed. As a result of the audit, re-radiography of a portion of the construction welds was accomplished followed by repair of nine coolant system welds.

Maintenance and modifications of major significance included: replacement of two primary cold traps, one reactor-vessel sodium-level probe, and the main IHX level probe; installation of the reactor-cover gas monitor, emergency diesel enclosure, and a liquid argon supply system; and rewinding of the stator on one nitrogen-blower motor. Two instrumented fuel assemblies were installed in the reactor core.

Test procedures for a number of experiments, at power, were reviewed, approved and issued. Preliminary analysis and reduction of data for the zero-power tests continued. A license which permits operation at steady-state up to 20 MWt in the pulsed mode was received on January 2, 1970. Operation was resumed on January 4th with the initiation of preparations for the ascension to power.

3. TASK 9—FUEL FABRICATION

3.1 PURPOSE

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

3.2 PAST ACCOMPLISHMENTS

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

3.3 INSTRUMENTED FUEL RODS

Fuel rod fabrication was delayed pending delivery of a thermocouple weldment. Delivery of this weldment is expected in the near future which will allow completion of the final fuel rod for the sixth IFA in the next report period. The other fuel rod required for the sixth IFA was approximately 30% completed.

3.4 IFA ASSEMBLY OPERATIONS

The fourth IFA was completed and delivered to the reactor site December 2, 1969. This is the first central-core-position (E 0.5-S1) IFA to be fabricated. On January 26, 1970, the fifth IFA was completed and delivered to the reactor site. This mid-radius core position (E 3) IFA is the second of this type to be fabricated.

The final thermocouple weldment in the process of fabrication was delayed due to several component failures. The W/5 Re wire failed several times during the sine-wave-forming operation, and the W/26 Re sheath failed in the final jog-bending operation. More rigorous material inspection and close control on the forming operations were used to overcome these problems.

A modification was made on the lead support brackets for the fourth and later fabricated IFA's. The modification resulted in one welded bracket replacing two smaller brackets, with a reduced tendency to develop fretting or chafing during operation. The Design Engineer and an IPO Shop Technician made a trip to the reactor site where the first three delivered IFA's were also modified.

The flowmeter for the sixth IFA was rebuilt after a failure to make a successful braze at the point where the lead enters the lower end of the tightener rod. The braze was made successfully on the second try.

The final report on IFA fabrication was started. This report will cover development of fabrication techniques, as well as the sequence of making parts and the assembly of components. Quality control and testing will also be included.

3.5 REACTOR INSTALLATION OF IFA's

Two IFA's were inserted into the SEFOR core January 25, 1970. The installation was accomplished under the direction of Reactor Operations, Reactor Program and Analysis, and Design Engineering.

A number of special tools and fixtures were used in the operation. A platform was used in the vessel at the elevation of the shield ring, approximately 2 feet below the vessel flange. This was used by a suited operator to enable hand contact with the IFA connector flange area.

Wire brushes and high pressure argon were used to clean the sodium oxide from the flange and vessel area, to prevent contamination of the electrical connector when it was open during the insertion operation. A hand-operated torque wrench was used to make up the "conoseal" gasketed connector flange.

A taper-type of feeler gauge was used to measure the gap between flange faces. This tool was hand held and used from the platform work station.

The lifting fixture was used to grapple and support the IFA when it was removed from the transport fixture on the refueling cell wall and inserted into the reactor core position. This fixture uses two grapples (1/2-ton grapple type) to hold the IFA from lifting lugs, which are located at a point above the channel part of the IFA and a point above the electrical connector flange.

The operations of preparing the vessel, fuel channel arrangement, grapping the IFA from the refueling cell wall position, and installing the IFA in the core were accomplished with minimum use of the suited operator in the refueling cell. All work possible was accomplished outside the cell. The total operation was successful. The only unplanned time delay involved grapping the IFA on the wall and releasing the IFA from the grapples after insertion in the core. These operations were accomplished with some difficulty. A minor modification to the lifting fixture was planned to prevent recurrence of this problem.

After each installation of an IFA in the SEFOR core, the sensors (thermocouples and flowmeter) were checked-out through the data acquisition system and were found to be operative.

3.6 FUEL RODS

Negotiations were begun with Nuclear Fuel Services (NFS) for fabrication of additional and replacement fuel rods. An agreement-in-principle was reached between NFS and GE, under which NFS would: (1) repair 23 rods and replace 17 rods which are now unusable for mechanical reasons, (2) replace 30 rods which are now unusable because of plutonium deficiency, and (3) fabricate 55 addi-

tional rods under terms of an option in the original fuel order contract.

3.7 PHOTOGRAPHS ILLUSTRATING ASSEMBLY OF IFA's

Figures 1 through 9 show photographs illustrating various aspects of assembly of IFA's.

4. TASK 10—STARTUP TESTING

4.1 PURPOSE

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

4.2 PAST PERFORMANCE

Initial criticality in SEFOR was attained with a fuel loading of 550 fuel rods. Investigation revealed that a number of these rods were low in reactivity worth compared to a standard rod. The low-worth rods were removed and a minimum critical loading of 522 fuel rods was attained, in good agreement with the predicted¹ value of 519 rods.

The control strength of all the reflector segments was estimated from the initial loading data to be 9.7 dollars. This compared well with the predicted¹ value of 11.2 ± 1.8 dollars. The reactivity worths of fuel, guinea pig, B_4C , UO_2 , stainless steel, and BeO tightener rods were measured in two different core loadings. Experimental and calculated results were in reasonable agreement near the core boundary, and the calculated central worths were generally 10 to 20% higher in magnitude than the measured values. The average temperature coefficient of reactivity between 350°F and 400°F was measured to be -0.59 cent/°F in Assembly I-D.

4.3 REFLECTOR CALIBRATION

In Assembly I-E (Figure 10), the reflectors were calibrated with the reactor at a temperature of 350°F, and again with the reactor at a temperature of 650°F. In both cases the two fine reflectors, No. 3 and No. 8, were calibrated in increments by means of period measurements. The total worth of each of the other reflector segments was determined by lowering a coarse reflector, raising one fine reflector, and adjusting the other calibrated fine reflector to maintain criticality by compensating for the differences in reflector worths.

The fine reflector calibrations are performed in the following manner.

1. The reactor was brought to criticality with the fine reflector to be calibrated, in the completely lowered position and the other fine reflector completely raised.
2. A coarse reflector configuration was selected to maintain criticality and to minimize the influence of a lowered coarse reflector on the fine reflector calibration (i.e., the coarse reflectors immediately adjacent to the fine reflectors, maintained in the completely raised position).
3. The reflector to be calibrated was raised a few centimeters to produce a period of the order of 100 seconds, the period was measured, and the other fine reflector was then lowered to bring the reactor back to critical at the original flux level.
4. This process was repeated until the entire length of the fine reflector had been calibrated. Since Assembly I-E was critical at 350°F with 2-1/2 reflectors lowered, fine reflector No. 3 was calibrated with reflector No. 5 lowered, No. 10 raised 41%, and the other coarse reflectors completely raised, while No. 8 was calibrated with No. 10 down and No. 5 raised 36%. At 650°F the reactor was critical with slightly more than one reflector lowered, and so No. 3 was calibrated with No. 10 raised 85%, while No. 8 was calibrated with No. 5 raised 74%.

The fine reflector calibration curves for Assembly I-E at 350°F are given in Reference 3. The worth of the fine reflectors as a function of position at 650°F are illustrated in Figures 11 and 12. The total worth of each of the reflectors at the two different temperatures are summarized in Table 1.

As previously noted,³ the asymmetrical loading pattern in Assembly I-E (see Figure 10) is consistent with the indicated asymmetric reflector worths in Table 1. In addition to this asymmetry, the tabulated worths clearly demonstrate that the worth of a reflector is influenced by

A CLOSEUP PICTURE OF THE TUNGSTEN/RHENIUM THERMOCOUPLE SHEATH TUBING AFTER FORMING. THE SPLITTING OCCURRED ON BOTH SIDES OF THE TUBING IN BOTH AREAS OF BENDING. THIS PROBLEM WAS CORRECTED BY ADDITIONAL RIGOROUS INSPECTION BEFORE FORMING.

Figure 1. Rejected Tungsten-Rhenium Thermocouple Sheath.

AN IN-PROCESS QUALITY CONTROL CHECK BEING MADE OF THE THERMOCOUPLE LEADOUT-FORMING PROCESS. THE LOWER PORTION HAS BEEN COMPLETED INCLUDING LEAK CHECKING AND HELIUM BACK FILLING.

Figure 2. Assembly Jig Set Up for "Ribbon" Lead Forming.

THE THERMOCOUPLE LEAD WIRES EXITING FROM THE SEAL BLOCK BRAZE HAVE BEEN FORMED SO THAT THEY WILL FIT IN-BETWEEN FUEL BUNDLES OR OVER THE WALLS OF THE HEX CHANNELS.

Figure 3. Sheathed Leads After Forming Operation.

THE ADDITIONAL STRUCTURAL BRACKET HAS BEEN ADDED IN THE FIGURE "S" AREA OF THE THERMOCOUPLE LEADOUT WIRES TO REDUCE POSSIBLE FRETTING. THIS WAS DEVELOPED DURING THE FABRICATION SEQUENCE AS A RESULT OF CONTINUED ENGINEERING ANALYSIS.

Figure 4. Assembly Jig Arrangement for Upper Part of an IFA.

PREPARATION FOR THE SPLICE IS BEING MADE BETWEEN THE PRESSURE VESSEL PORTION AND THE LOWER SECTION. THIS IS ACCOMPLISHED BY CUTTING TO LENGTH AND REMOVING THE INCONEL SHEATHING. THIS SECTION OF PIPE IS SPLIT FOR ACCESS DURING THE SPLICE OPERATION.

Figure 5. Preparation for the Lead Splices.

THE SHIELD OR GUARD FOR THE THERMOCOUPLE LEADS FIGURE "S" EXPANSION SHAPE IS BEING FITTED INTO PLACE JUST PRIOR TO WELDING.

Figure 6. Protective Guard Over Expansion Leg of Leads.

THE ARRANGEMENT AND EQUIPMENT USED FOR CONDUCTING THE FUEL THERMOCOUPLE QUALITY CONTROL CHECK. EACH INDIVIDUAL THERMOCOUPLE HOT JUNCTION IS HEATED AND THE EMF OUTPUT IS RECORDED FOR ALL COMBINATIONS OF THERMOCOUPLE W RES.

Figure 7. Sensor Lead Check Out

THE COMPLETED UPPER SECTION OF THE IFA AS VIEWED IN THE FINAL ASSEMBLY JIG.

Figure 8. Completed Upper Part of IFA in the Assembly Jig.

DETAILS OF THE UPPER END OF THE IFA, SECURED IN THE TRANSPORT FIXTURE. THE HINGED LIFTING LUG OF THE TRANSPORT FIXTURE ALLOWS THE IFA TO MAKE A CONTROLLED TRANSIT FROM THE HORIZONTAL POSITION TO A VERTICAL POSITION.

Figure 9. IFA Supported in the Transport Fixture

CORE LOADING STATUS

Figure 10. Core Loading for Assembly I-E

Figure 11. Fine Reflector No. 3 Calibration at 650°F (Assembly I-E)

Figure 12. Fine Reflector No. 8 Calibration at 650°F (Assembly I-E)

Table 1
REFLECTOR WORTHS IN ASSEMBLY I-E

Reflector Number	Calibration at 350°F (reflector No. 5 down)		Calibration at 650°F	
	Worth (dollars) ^a (swapped with No. 3)	Position of Reflector No. 8 (cm) ^b	Worth (dollars) ^a (swapped with No. 3)	Position of Reflector No. 8 (cm) ^b
1	1.42	44	1.39	84
2	1.44	45	1.43	88
3	1.32 ^c	39	1.38 ^c	82
4	1.00	17	1.30	75
5	1.23 ^d	40	1.27	71
6	0.87	2	1.24	70
7	0.88	3	1.16	64
8	1.20 ^c	—	1.25 ^c	—
9	1.00	17	1.18	66
10	1.34	40	1.33	78

a. The estimated standard deviation in the difference between two values at the same temperature is ± 0.01 dollars.

b. Position of reflector No. 8 when the indicated reflector was lowered.

c. Worth obtained from period measurements.

d. Worth measured with reflector No. 10 lowered.

the position of its neighbor. In particular, the worth of a reflector is less when an adjacent reflector is lowered, or partially lowered, than when both adjacent reflectors are completely raised. For the measurements at 250°F, reflector No. 5 was lowered and No. 8 was partially or almost completely lowered. The worths of reflectors No. 6 and No. 7 are thus considerably less than the worths of the other reflectors, since they were determined with neighboring reflectors on both sides in the lowered position. The worths of No. 4 and No. 9 demonstrate the same "shadowing" effect, although to a lesser extent, since their worths were determined with only one completely lowered neighboring reflector.

A comparison of the calibrations at 650°F, which were performed with all reflectors raised, with calibrations at 350°F clearly demonstrates the influence of the lowered reflector No. 5 and partially lowered reflector No. 8 on other reflector worths. The worths of reflectors No. 1, No. 2, and No. 10 are essentially unchanged and No. 5 is only slightly changed, while the worths of No. 4, No. 6, No. 7, and No. 9 are from 18% to 43% larger at 650°F, where No. 5 is completely raised and No. 8 is raised more than at 350°F. The worths of No. 3 and of No. 8 are both about

4% larger at 650°F than at 350°F. This is probably due to the influence of reflector No. 5 on the calibration of No. 3 and the influence of reflector No. 10 on the calibration of No. 8 (see above).

The "shadowing effects" thus appear to be localized, and the worth of a reflector is nearly independent of the position of non-adjacent reflectors provided the immediately adjacent reflectors are up. As indicated above, however, the worth of a reflector may be influenced slightly (up to ~4%) by the position of a neighbor that is only two reflectors removed, and the worth is strongly dependent on the position of immediately adjacent reflectors.

4.4 SEFOR FLOW AND PRESSURE COEFFICIENTS OF REACTIVITY

The flow and pressure coefficients of reactivity in SEFOR are essentially zero. The reactivity effects that are associated with changes over the normal operating range of reactor vessel hydraulic pressure and up to 80% of maximum primary coolant flow are smaller than the standard deviation (0.5 cent) in reactivity difference measurements.¹ These results are in agreement with expectations.²

In order to enable the ultimate determination of the Doppler reactivity coefficient, a number of experiments are performed in SEFOR to determine the relationship, if any, between reactivity (reflector position) and plant variables such as fuel temperature, reflector temperature, core inlet temperature, etc. The experiments discussed here were performed to determine the relationship between reactivity and coolant flow, and between reactivity and reactor vessel hydraulic pressure. In these experiments, the flow and the vessel-cover gas pressure were separately and independently changed in increments between minimum and maximum values while the reflector position was adjusted to maintain a constant flux level. The variation in reactor temperature during these measurements was held to less than 5°F.

The data are tabulated in the sequential order of data collection in Tables 2 and 3, and the relative reactivity after corrections for the temperature variations are plotted versus the flow and pressure in Figures 13 and 14. Note that the tests were conducted for flow rates up to 4000 gal/min, whereas the maximum flow rate is 5000 gal/min. The limitation of 4000 gal/min was imposed by a low sodium level scram. When this system deficiency is remedied, the tests will be extended to 5000 gal/min flow rate.

The data indicates little or no correlation between flow and reactivity, or between pressure and reactivity. For example, the maximum difference between any of the reactivity values in Figure 13 (or Table 2) is 0.53 cent, and the maximum deviation from the mean of these values is only 0.38 cent. The maximum deviation between any two values in Table 2 is 1.15 cents, and the maximum deviation from the mean of these values is 0.82 cent. These deviations are all of the same order of magnitude as the estimated standard deviation¹ (± 0.5 cent) in the reactivity measurements.

The data thus indicates that if there are any reactivity effects which accompany changes in flow or pressure, they are no larger than the 0.5 cent estimated standard deviation in reactivity measurements. Any effects of flow and pressure on reactivity are, therefore, negligible and can be neglected.

4.5 TEMPERATURE COEFFICIENT MEASUREMENTS

The reactor was heated from 350°F to 760°F with the primary loop trace heating system, and the reactivity feedback effect was determined from the recorded temperatures and calibrated reflector positions. The temperatures and reflector positions required to maintain a constant flux

Table 2
FLOW-REACTIVITY MEASUREMENTS

Coolant Flow (gal/min)	Reflector No. 8		Temperature (°F)	Relative System Reactivity at 350°F ^b (cents)	Deviation from Mean Reactivity (cents)
	Position (cm)	Worth (cents) ^a			
500	39.54	48.35	350.5	48.01	0.20
1000	39.56	48.39	351.0	47.71	-0.10
1500	39.56	48.39	351.0	47.71	-0.10
2000	39.56	48.39	351.0	47.71	-0.10
2500	39.63	48.51	351.0	47.83	-0.02
3000	39.84	48.87	351.0	48.19	0.38
3500	40.13	49.37	352.0	48.01	0.20
4000	40.36	49.77	353.0	47.73	-0.08
370	38.66	46.83	348.75	47.68	-0.13
830	38.65	46.81	348.75	47.66	-0.15
1410	38.88	47.21	349.25	47.72	-0.09
2000	38.96	47.35	349.5	47.69	-0.12
2600	38.97	47.36	349.5	47.70	-0.11
3150	39.15	47.68	349.5	48.02	0.21
Mean = 47.81					

a. Relative to worth in the completely lowered position.

b. Using a temperature coefficient at 350°F of -0.68 cents/°F.

Table 3
PRESSURE-REACTIVITY MEASUREMENTS

Cover Gas Pressure (psig)	Reflector No. 8 Position (cm)	Worth ^a (cents)	Temperature (°F)	Relative System Reactivity (cents)	Deviation From Mean Reactivity (cents)
20	38.85	47.16	348.75	48.01	-0.17
15	38.85	47.16	348.75	48.01	-0.17
0	39.36	48.04	349.0	48.72	0.54
20	39.03	47.47	349.25	47.98	-0.20
15	39.14	47.66	349.5	48.00	-0.18
10	39.13	47.64	349.5	47.98	-0.20
5	39.13	47.64	349.5	47.98	-0.20
0	39.21	47.78	349.0	48.46	0.28
20	38.66	46.83	348.5	47.85	-0.33
0	39.23	47.81	348.25	49.00	0.82
20	38.67	46.84	348.25	48.03	-0.15
				Mean = 48.18	

a. Relative to worth in the completely lowered position.

b. Using a temperature coefficient at 350 °F of -0.68 cents/°F.

level were recorded every 15 minutes. After each 100°F rise in temperature, the temperature was held constant within about 2°F for a period of 2 to 3 hours to determine if there were any temperature-dependent feedback effects with very long time constants. Within the accuracy of the incremental temperature (~0.25°F) and reactivity (0.5 cent) measurements, no long term reactivity effects were observed, and the reactor appeared to maintain thermal-reactivity equilibrium at a heating rate of 15°F to 20°F per hour.

Representative values of temperature and reactivity obtained during the experiment are summarized in Table 4 and are illustrated in Figure 15, along with the temperature dependent feedback calculated using the predicted⁴ Doppler $T(dk/dT)$ of -0.0082 and the combined density expansion coefficient of -0.36 cent/°F. As the figure illustrates, there is excellent agreement between experiment and prediction. The measured and calculated⁴ temperature coefficients are compared in Table 5. Although the agreement between this experiment and prediction is quite good over the entire range of the experiment, the coefficient in the range between 350°F and 400°F is higher than the previously³ measured value of -0.59 cent/°F. The reason for this disparity may be determined when the reactor is heated to 760°F for the zero-power oscillator test.

4.6 ACTIVATION MEASUREMENTS OF THE FULL SIZE CORE

For the analysis of the fission rate profiles, diffusion calculations with 13 energy groups using a two-dimensional flux synthesis program and a true two-dimensional program have been performed. For these calculations the core region was divided into five radial core annuli and the 14B₄C rods were homogenized into two of these zones to approximate the actual geometry in Assembly I-D (see Figure 16).

For comparison of the results, the ratios of measured to calculated values from the two-dimensional calculation are listed in Table 6. Assuming only small calculational uncertainties at the point of relatively flat flux distribution (i.e., at the point of the maximum power) all these ratios were normalized to this point.

Generally, good agreement between the experimental and the theoretical data can be observed for all isotopes except near the boundaries between the core and the reflectors and in the gap. All radial profiles show an increasing ratio for locations close to the reflector. This might be due either to spectral or transport effects. Therefore, additional reaction rate calculations using transport theory and more energy groups are planned. Although the positions FF and FE are diametrically opposite in the core and are the same

Figure 13. Flow Reactivity Measurements

Figure 14. Pressure Reactivity Measurements

Figure 15. Temperature-Reactivity Feedback (Assembly I-E)

CORE LOADING STATUS

Figure 16. Core Loading for Assembly I-D

Table 4
TEMPERATURE-REACTIVITY MEASUREMENTS

Temperature (°F)	Reactivity Change* (cents)	Temperature (°F)	Reactivity Change* (cents)
356.5	4.5	562.75	139.4
370.0	13.1	577.0	147.7
384.25	23.4	588.0	154.0
398.5	32.6	606.25	165.0
409.25	39.8	618.75	172.2
415.5	43.9	631.5	181.1
426.75	51.2	639.75	185.4
437.5	59.0	650.0	192.0
451.75	68.4	659.5	198.9
461.0	74.2	663.5	201.4
469.25	79.2	669.0	204.4
483.0	87.5	677.5	207.9
492.5	94.9	690.0	215.2
507.0	105.5	700.0	221.4
511.5	105.5	712.25	227.2
519.75	114.3	724.5	234.1
524.5	114.9	736.75	241.0
535.0	123.0	750.0	247.5
550.0	132.0	759.0	253.2

* Relative to a temperature of 350°F.

Table 5
TEMPERATURE COEFFICIENT MEASUREMENTS

Temperature Range (°F)	Average Reactivity Coefficient (cents/°F)	
	Experiment	Calculation
350 - 450	-0.67	-0.66
450 - 550	-0.65	-0.63
550 - 650	-0.59	-0.61
650 - 760	-0.57	-0.58

distance from the center, position FE is near the partially lowered (approximately 50%) reflector No. 8, but all other reflectors were up. This fact may explain the lower ratios for rod FE relative to rod FF.

The axial profiles show the same trend. The generally higher ratios near the upper reflector (axial position No. 8)

relative to those near the lower reflector (axial position No. 1) are due to the smaller distance between foils No. 8 and the upper reflector compared to foils No. 1 and the lower reflector.

In the gap region, experimental and calculated values agree better for U-235 than for U-238. This indicates that the fast flux is relatively higher than calculated. A probable explanation for this effect is the inaccuracy of the diffusion approximation near boundaries, since diffusion theory underestimates the leakage of high energy neutrons.

There were no significant differences between the reaction rates obtained with the two-dimensional synthesis and with the true two-dimensional programs. Generally, however, the reaction rates from the synthesis calculations were higher near the reflector zones, thus increasing the disparity relative to the measured data by about 20%.

The higher than predicted PU-239 and U-238 fission rates near the core boundaries indicates a flatter flux distribution in the core than obtained from diffusion calculations. This implies a flatter power distribution, which is qualitatively in agreement with the results of the material worth measurements.

4.7 NOISE MEASUREMENTS

Reactor noise measurements have been performed in Assembly I-D. Measurements were made at four different power levels between approximately 0.2 and 6000 watts. The recording time for each measurement was 2.5 minutes.

To determine β/λ from the tape-recorded currents of the two He-3 chambers, the auto- and cross-correlation functions as well as the auto- and cross-spectral densities have been calculated. To obtain statistically independent results from the correlation functions, each measurement was divided into series of 20 sec recording times, and then each series was analyzed separately.

Corrections for the finite band width of the recording system, determined previously with "white noise" and a sine-wave generator were made. In several cases corrections were also made for induced background. Calculations showed that corrections due to delayed neutrons and gamma radiation are very small, and these were therefore neglected.

The He-3 chambers showed saturation effects at power levels greater than or equal to 1 kW, and the background level was relatively high for the measurements at the lowest power level of 0.2 watts. Because of these effects, only the results from measurement at the relatively low power of 22 watts were considered reliable. They are listed in Table 7 together with the theoretical values.

Table 6
RATIOS OF MEASURED TO CALCULATED FISSION RATES NORMALIZED TO THE POINT FB, No. 3
 (The calculated values are those obtained from a two-dimensional diffusion calculation)

		FB	FG	FC	FF	FE	FD
Radial Position Distance ^a		5.59	19.56	22.86	42.93	42.93	43.94
Axial Position No.	Distance ^b	Pu-239					
8	133.24	1.40	—	—	—	—	—
7	115.97	1.07	—	—	—	—	—
6	100.86	1.07	—	—	—	—	—
5 } gap	97.68	1.04	—	—	1.28	—	1.30
4	95.78	1.02	—	—	—	—	—
3	79.90	1.00	0.99	0.99	1.21	—	1.21
2	69.11	0.98	—	—	—	—	—
1	43.71	1.19	—	—	—	—	—
		U-238					
8	133.24	1.32	1.33	1.38	—	—	—
7	115.97	1.01	1.03	1.04	1.34	—	1.34
6	100.86	1.06	1.08	1.06	1.38	—	1.48
5 } gap	97.68	1.13	1.15	1.12	1.53	1.34	1.53
4	95.78	1.08	1.12	1.17	1.52	—	1.58
3	79.90	1.00	0.96	0.97	1.21	1.20	1.31
2	69.11	0.96	0.97	0.96	1.20	—	1.28
1	43.71	1.07	1.09	1.05	—	—	—
		U-235					
8	133.24	1.31	1.36	1.37	—	—	—
7	115.97	1.12	1.11	1.12	1.59	—	1.16
6	100.86	1.06	1.07	1.08	1.07	—	1.17
5 } gap	97.68	1.03	1.06	1.05	1.20	1.00	1.10
4	95.78	1.03	1.05	1.02	1.07	—	1.06
3	79.90	1.00	1.00	1.02	1.06	1.02	1.05
2	69.11	1.00	1.03	1.00	1.04	—	1.05
1	43.71	1.18	1.20	1.20	—	—	—

a. Distance in cm from core center line.

b. Distance in cm from bottom of reactor.

Table 7
COMPARISON BETWEEN CALCULATED AND MEASURED VALUES OF ℓ/β

Power Obtained From Noise Measurement	22.3 watts
Core Subcriticality	-1.20 ± 0.5 cents
Measured ℓ/β (sec)	$0.205 \times 10^{-3} \pm 1\%$
Calculated 2-Dimensional Synthesis ℓ/β (sec)	0.214×10^{-3}
Calculated 2-Dimensional Diffusion Program ℓ/β (sec)	0.184×10^{-3}

The subcriticality in Table 7 was obtained from the calibrated reflector readings. The corresponding power level was calculated from the parameters of the cross-correlation function without using the value of the He-3 detector efficiency. This power calibration is in good agreement with the calibration obtained from the subcritical measurements that use the theoretical value for the inherent neutron source.³

The value of β/ℓ was calculated with a two-dimensional synthesis and with a true two-dimensional diffusion program. In the first method relatively high uncertainties are expected due to the approximate methods of obtaining axial bucklings for the reflector regions. Therefore, the result of the two-dimensional calculation must be regarded as the best theoretical value. The comparison with the experimental result indicates an 11% higher neutron lifetime than calculated.

5. TASK 11—TRANSIENT TEST

5.1 PURPOSE

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

5.2 PAST PERFORMANCE

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

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

SEFOR fuel rods has been completed and is being reviewed. The license application contains a description of the cask, the procedures to be followed when handling the fuel, and sufficient analysis to show that the conditions described in 10CFR71 are met.

The design of the equipment required for the fuel transfer operation was completed, and has been submitted to vendors for bids. The equipment includes the transfer tube and seals between the refueling cell valve and transfer cask; the set of canisters used to ship a single failed fuel rod, and the canister used to ship three non-failed fuel rods.

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

5.3 CURRENT RESULTS

Fabrication of the following equipment has been completed and is being prepared for shipment to the SEFOR Site.

1. One set of canisters to be used for shipping a single failed fuel rod
2. One canister to be used for shipping three nonfailed fuel rods
3. The equipment for holding the canisters while the fuel is being loaded into them

4. The equipment required to connect the shipping cask to the refueling cell.

A modification was made to the design of the three-rod shipping container, providing additional protection to the fuel elements during a hypothetical accident involving dropping the cask. Cushioning material was included at both ends of the fuel rod to absorb the impact energy without stressing the clad above its yield.

Review of the license application has been completed, and the final draft is being prepared.

6. SEFOR SAFETY AUDIT

Three trips were made to the site during this quarter for safety audit purposes. The following areas were checked:

1. **Standard Operating Procedures.** Copies of completed procedures used for recent reactor startups were reviewed. All steps were filled in and signed off, as required. Over 300 startups were made during the first two quarters of plant operation. This number includes criticality checks made during the initial core loading process.
2. **License Test Procedures.** These tests are used to assure compliance with surveillance requirements of the technical specifications. Review of recently completed tests showed that these tests are being performed on time, and that the occasional discrepancies which were observed were corrected in a timely manner. The license test procedures were checked against the technical specifications to assure that all

requirements were met. Additional tests not required by the technical specifications are included in the license test procedures.

3. **Refueling Cell Access for IFA Installation.** Two instrumented fuel assemblies (IFA's) were installed in the reactor at the end of this quarter. This procedure required man access to the refueling cell while the reactor head was removed. The sodium level in the reactor vessel was lowered below the tops of the fuel channels, and a platform was installed in the reactor vessel to serve both as a vessel cover and a working platform. A safety harness was used by the man working on the platform and lanyards were attached to all tools used by him. Platform temperatures were checked (165°F) and found to be well below the safe limit for man access suit contact (250°F). The highest vessel wall temperature near the platform was 185°F. Radiation levels were also checked and found to be safe (1 mR/h).

7. SEFOR OPERATIONS

7.1 EXPERIMENTAL PROGRAM

On January 10, the reactor core was returned to the premaintenance (October 31, 1969) configuration by inserting the fuel, poison, and tighteners, which were removed for installation of the immersion heaters. A reference criticality check was made to provide reactivity accountability assurance. After temperature correction, a slight reactivity loss of -3.7 cents between criticality checks was noted. This small, but perceptible loss, is attributable (within the measurement uncertainty) to the decay of Pu-241 in the core for the extended down period (72 days). Similar reactivity loss has been measured and reported by Argonne National Laboratory.⁵ The core was rearranged for full power operation. This rearrangement provided a relatively symmetrical material and power distribution within the core and incorporated the best current intelligence about full power reactivity requirements. The rearrangement accomplished the following objectives:

- a. Obtained the most symmetrical distribution of B₄C and guinea pig rods possible in the core to rectify the apparent core flux asymmetry noted on Assembly I-E (the previous full power core loading).
- b. Reduced the amount of through-head reactivity adjustment to reflect the better knowledge of excess reactivity demands in going from 350°F to 760°F (core average temperature) and 20 MWt. This improved knowledge results from the temperature coefficient measurement to 760°F (TP I-D.4).

With the core in the revised full power configuration (I-F) and at a core temperature of approximately 350°F, the fine reflector No. 8 was differentially calibrated over its full length. The core excess reactivity was reduced to approximately \$1.50 to facilitate the calibration of fine reflector No. 8 in the 20 MWt configuration—all coarse and one fine reflector raised with one fine reflector down. This reduction in excess reactivity was accomplished by symmetrically removing the through-head fuel. Since the through-head fuel must be removed prior to installing the head, utilization of the fuel removal to establish reactivity requirements minimizes fuel and poison handling in-cell.

Further reactor operation was restricted by the malfunction of the inverter supplying electrical power to the control room recorders so installation of two instrumented fuel assemblies VNC-25 and VNC-29 was undertaken. These two instrumented assemblies were installed on January 25, 1970.

The following test procedures were completed during the reporting period:

Test Procedure	Status
Test Plan Procedure—Group III (TP III-1)	Approved
Natural Circulation Test (TP II-7)	Approved
Reflector Temperature Coefficient Measurement (TP II-8)	Approved
Heat Balance and Nuclear Instru- mentation Calibration (TP II-3)	Approved
Fuel Post-Irradiation Inspection Plan	Approved
Plant Heat Balance (TP II-6)	Approved

An "off-line" Fourier analysis program was completed and a test case run for comparative purposes. The results were compared with the Sunnyvale version of the "off-line" Fourier analysis for the same test case. The test case selected was the "simulated" oscillator magnetic tape which had been transmitted to Sunnyvale earlier to verify computer compatibilities. Some minor problems were encountered, but after corrections were made, comparable results from both analyses were obtained.

A Nyquist diagram code was written which takes output from the "off-line" Fourier analysis program and calculates Nyquist diagram data. Ultimate plans are to have this information plotted on the Cal-Comp plotter for rapid stability determinations at various power levels during the power ascension phase.

Preliminary analysis and reduction of data for the zero power tests continued during the period.

7.2 OPERATION AND MAINTENANCE

On November 1, at the completion of the zero power phase of the experimental program, a planned outage was initiated to perform maintenance and modifications prior to the ascension to power. The outage continued through the month of December and was terminated on January 4.

Prior to the outage, the reactor was operated to complete the isothermal temperature coefficients and the reflector worth measurements at 760°F. Following the outage, criticality checks were made after rearrangement of the

core for full power operation, and reflector No. 8 was recalibrated. A total of 15 reactor startups were made and six automatic scrams occurred; two from switching transients while changing the range selector switch on the wide range monitors, three from operator error in changing the range selector switch, and one from a spurious main primary low flow trip.

During the outage, all sodium systems were drained, insulation and heaters removed to permit re-radiograph of selected welds in all systems including the primary and secondary service systems. These radiographs, together with the construction radiographs, were reviewed to verify that all welds met the required quality standards. A few welds which did not meet the standards were repaired or replaced. The pressure sensors for the differential pressure measurement across the venturi in the primary system were removed, calibrations performed, and sensors replaced. The two cold traps in the primary system and the sodium level probe in the main primary IHX were replaced with new units. At the completion of this work, the sodium systems were refilled and cold traps operated until the plugging temperatures were below 250°F.

Operation checks of the oscillator and positioner were performed, and the positioner motor was mounted on the reactor vessel head. A total of 19 temporary operating procedures were prepared to attain plant conditions which would permit the performance of the modification and maintenance activities.

Fifty-four (54) modifications were completed; 13 electrical and 41 mechanical. A total of 158 malfunctions were corrected; 46 electrical, 48 mechanical, and 64 instrumentation.

Maintenance and modifications of major significance included:

- Replacement of the two primary cold traps.
- Replacement of one reactor vessel sodium level probe.
- Replacement of main IHX level probe.
- Installation of the reactor cover gas monitor.
- Rewinding of stator on nitrogen blower motor.
- Installation of emergency diesel enclosure.
- Installation of liquid argon supply system.

Other maintenance and modification activities included:

Electrical:

- Automatic Flow Control—Main Sodium Pumps.
- Sodium Pump Duct and Windrive Temperature Inducting Switches.
- Reactor Nozzle 22 Thermocouple Readout.
- Reactor Building Pressure Controller DPRC 9235 Power Source.

Auxiliary ABC Door Permissive.

IRM—VPLM Period Scram Suppressor Circuit.

Data Acquisition System Inputs (Main Sec Flow and Oscillator Position).

2.4 kV Undervoltage Relays Power Supply Circuit.

Dillon Load Cell Readout Circuit (Crane Control Console).

Diesel Generator Start Motor Current Indication.

Data Acquisition System Input from Venturi Delta P Transmitter.

Freon Back Pressure Regulating Valves 8120-1, -2, and -3 Solenoid B Power Supply.

Refueling Cell Crane Power Cables and Connectors.

Mechanical:

- Equipment Door Safety Support.
- Breathing Air Compressor Cooling Water.
- IFA Installation Equipment.
- Moisture Collection Tanks Sight Glass Marks.
- Diaphragm Change for Valves, Purification Loop.
- Disassembly Station.
- Moisture Trap on Instrument N₂ Dryer Inlet.
- Floor Well Sleeve.
- Refueling Cell Gas Sample Line.
- Shield Plug.
- Manual Valve Locking Devices.
- Valve 934-3 Extension Handle.
- Valve 394-10 Extension Handle.
- Temporary Drain Line, Main Primary and Secondary Systems.
- Hose Guard, Impact Wrench.
- Crane Repair Marine Hatch Test Panel.
- Sodium Distillation Station.
- Bolt Test Fixture.
- Auxiliary Diesel Fuel Shutoff.
- Argon Blower Shaft Seal.
- Man Access Suits.
- Pressure Switch Shock Mounting.
- Argon Pressure Switch.
- Sodium Pipe Tunnel Penetration Test Manifold.

Drywell.
 Drywell Floor Plug.
 Floor Well Adapter Spool.
 Non-Contaminated Washer.
 Safety Step for PSV 281-6.
 Fuel Transfer Equipment.
 Gauge Test.
 Marine Hatch Fuel Loading Ramp.
 Man Access Locking Ring.
 IFST Drain Line on Dip Leg.

Malfunctions Corrected:

Reactor Building Air Handling Unit Bearings.
 Refueling Cell Crane.
 Auxiliary Secondary Expansion Tank Low Level Switch LS 244-8.
 Argon Blower Seal.
 Containment Penetrations.
 Reflector Guide Structure Thermocouples.
 Pump Around Pump Duct Thermocouples.
 Distillation Station Purge Valve
 Emergency Diesel Automatic Start.
 IFA Electrical Connector.
 Reactor Core Temperature Indication.
 Wide Range Monitor No. 2.
 125 Vdc System.

At the completion of the maintenance and modification activities, the refueling cell was filled with argon and the purification loop operated to lower the moisture and oxygen contents to a few ppm. The nitrogen zones were purged until the oxygen content was approximately 1.5%.

A gas chromatograph was procured and calibrated for measurement of the constituents in the various gas environments.

Two instrumented fuel assemblies were installed in the reactor core; VNC 25 in position N 5.0 E 2.5, and VNC 29 in position S 1.0 E 0.5. Check-out of signals from these assemblies was initiated.

Whole body counting was performed for 14 of the personnel to provide reference data.

A license for operation at steady state up to 20 MWt and in the pulsed mode was received on January 2, 1970. Numerous license test procedures and preventive maintenance procedures were accomplished to demonstrate that equipment was operable.

7.3 ENGINEERING SUPPORT

Additional engineering was performed on several of the above mentioned maintenance and modification items in support of SEFOR operation and maintenance.

A comprehensive audit of the sodium coolant system weld radiograph was completed during this period.

A review meeting was held with the Division of Compliance and it was determined that a portion of the construction radiographs displayed some deficiencies relative to code requirements. As a result, a complete review of the coolant system radiography was accomplished by General Electric. In addition, a separate independent audit of the field welds was provided by an outside, independent, consultant.

Based upon the evaluation of the original radiographs, it was determined that the majority of the welds were satisfactory without need for further investigation. A portion of the construction radiographs required re-radiography to meet the intent of the code, and where conditions indicated non-interpretable evidence of weld quality.

As a result of the radiographic audit, several weld deficiencies were discovered. Evaluation of these conditions led to the repair of nine of the coolant system welds to bring the quality of the coolant system piping up to the intended level.

With the completion of the radiographic and weld quality review program, and the resulting acceptable weld repairs, the coolant systems were determined to be adequate for the intended reactor service.

During initial phases of plant operation difficulty was experienced with the reactor sodium level instrumentation. Evidence shows that the major difficulties with the level probes have been due to oxide contamination of the external surfaces. Design of replacement probes (Type II) was initiated which provides coolant flow through the assembly. Fabrication of one of the Type II level probes was completed and the probe was installed in the reactor vessel.

As an additional measure, a tip probe (Type I) was designed and installed within the reactor vessel head. This probe contains the reactor low level trip functions. This probe is located in an area where coolant flow is readily available and provides redundant low level trip function to the tip probes located in the assemblies at the vessel perimeter.

During operation of the sodium coolant systems, it has been found that the sodium level in the reactor vessel is flow dependent. As flow is increased, the reactor level is lowered correspondingly, a small amount. Increasing the pump-around-pump flow has been determined to be sufficient to maintain reactor level under the range of primary sodium system flow conditions. Through-head viewing equipment is being obtained to be used in observing the sodium flow patterns over the reactor during the full range of flow conditions as an aid in determining the required steps for ultimate resolution.

Difficulty was experienced with maintaining flow through the primary-sodium cold traps following extended

operation of the coolant systems. Procurement of traps for replacement was completed, and the two cold traps were shipped to the site, and installed. Initial operation of the replacement traps indicates completely satisfactory operation.

Erection of the diesel generator enclosure was completed. The enclosure is insulated and heated to maintain 40°F during the cold weather months and will ensure reliable operation of the system.

The liquid argon supply system was installed for the reactor systems. The liquid supply reduces labor costs involved with bottle changes, provides a supply at a lower unit cost and is delivered with a higher purity.

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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, 1969 through January 31, 1970. The work was performed under contract AT(04-3)-540 with the United States Atomic Energy Commission.

1. INTRODUCTION

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

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

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

Phase I — Preoperational research and development,

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

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

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

1. "Southwest Experimental Fast Oxide Reactor Development Program, First Quarterly Report, April—June 1964," GEAP-4594 (July 1964).
2. "Southwest Experimental Fast Oxide Reactor Development Program, Second Quarterly Report, July—September 1964," GEAP-4742 (October 1964).
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1. Novak, P. E., and Asamoto, R. R., "A Survey for a High Temperature Sensor for SEFOR," GEAP-4903 (July 1965).
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