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AEC RESEARCH AND
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FEBRUARY 1969

MASTER

**SOUTHWEST EXPERIMENTAL
FAST OXIDE REACTOR
DEVELOPMENT PROGRAM
NINETEENTH QUARTERLY REPORT
NOVEMBER 1968 – JANUARY 1969**

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

BREEDER REACTOR DEVELOPMENT OPERATION

GENERAL ELECTRIC

SUNNYVALE, CALIFORNIA

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GEAP-5754
AEC Research and
Development Report
February 1969

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM

Nineteenth Quarterly Report
November 1968—January 1969

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TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	-1-
SECTION I INTRODUCTION	-1-
SECTION II SUMMARY	-4-
2.1 Completed Tasks	-4-
2.2 Task 4—Special Instrumentation	-4-
2.3 Task 6—Reactivity Controls Required For Experimentation	-4-
2.4 Task 9—Fuel Fabrication	-4-
2.5 Task 10—Startup Testing	-4-
2.6 Task 11—Transient Tests	-4-
2.7 SEFOR Operations	-5-
SECTION III TASK 4—SPECIAL INSTRUMENTATION	-5-
3.1 Task 4.2—Core Flux Detector System	-5-
SECTION IV TASK 6—REACTIVITY CONTROLS REQUIRED FOR EXPERIMENTATION	-9-
4.1 Task 6.1—Fast Reactivity Excursion Device (FRED)	-9-
4.2 Oscillator Drive	-11-
SECTION V TASK 9—FUEL FABRICATION	-11-
5.1 Purpose	-11-
5.2 Past Accomplishments	-11-
5.3 Current Results	-11-
SECTION VI TASK 10—STARTUP TESTING	-35-
6.1 Purpose	-35-
6.2 Past Performance	-35-
6.3 Predicted SEFOR Minimum Critical Loading	-35-
6.4 Venturi Flow Meter Calibration	-41-
6.5 In-Hour Relations For SEFOR	-42-
6.6 References	-42-
SECTION VII TASK 11—TRANSIENT TESTS	-44-
7.1 Purpose	-44-
7.2 Past Performance	-44-
7.3 Current Results	-44-
SECTION VIII SEFOR OPERATIONS	-45-
8.1 Pre-Operation Tests	-45-
8.2 Fuel Inspection	-45-
8.3 Operator Training And License Examinations	-45-
8.4 Operations	-45-
8.5 Plant Engineering	-45-
8.6 Experimental Program	-46-
8.7 Radiation and Industrial Safety	-46-
SECTION IX SEFOR SODIUM STRAINER EXPERIENCE	-47-
9.1 Introduction	-47-
9.2 Strainers 1A and 1B	-47-
9.3 Strainer 2	-50-
9.4 Strainer No. 3	-56-

TABLE OF CONTENTS (Continued)

	<u>Page</u>
CONTRIBUTORS	-64-
DISTRIBUTION	-65-

LIST OF ILLUSTRATIONS

<u>Figure</u>	<u>Title</u>	<u>Page</u>
3-1	System Elementary Diagram	-6-
3-2	Amplifier Elementary Diagram	-7-
3-3	Calibration Unit Elementary Diagram	-8-
4-1	FRED Reactor Model Test Results (for 210 tests)	-10-
5-1	Barewire Fuel Thermocouple - Break	-13-
5-2	Fuel Thermocouple After Second Break	-14-
5-3	Typical Modified 1-1/8 inch Long Pellet	-16-
5-4	Fuel Thermocouples	-17-
5-5	Modified Fuel Ready for Loading	-18-
5-6	Upper Fuel Rod Segment with Spacer Inserted (seen through the Glove Box Window)	-19-
5-7	Completed Instrumented Fuel Rod	-20-
5-8	Instrumented Fuel Assembly Flow Meter (exploded view)	-21-
5-9	Instrumented Fuel Assembly Flow Meter (assembled view)	-22-
5-10	IFA Seal Block Braze Operation (chamber head removed)	-23-
5-11	IFA Seal Block Braze Operation (through the viewing port)	-24-
5-12	Coolant Thermocouple Braze	-25-
5-13	Bulkhead Braze Operation	-26-
5-14	Spider Before Assembly to Flow Meter	-28-
5-15	Flow Meter and Spider Subassembly	-29-
5-16	Tightener Rod and Sleeve (exploded view)	-30-
5-17	Lower Part of Instrumented Fuel Assembly	-31-
5-18	IFA in the Assembly Fixture	-32-
5-19	IFA Ready for Shipment	-33-
5-20	IFA Shipping Container on Truck	-34-
6-1	k_{eff} versus Percent of Full SEFOR Core Load (Control Reflectors Up—Temperature at 350° F)	-39-
6-2	Asymptotic Reactor Period for SEFOR Minimum Critical Core	-43-
9-1	Sodium Strainers No. 1A and No. 1B	-48-
9-2	Sodium Strainer No. 2	-51-
9-3	Strainer No. 2 After Removal From Sodium Loop	-53-
9-4	Pressure Loss versus Sodium Flow Rate - Strainer No. 2	-55-
9-5	Sodium Strainer No. 3	-57-
9-6	Water Test Facility Piping Arrangement	-60-
9-7	Water Flow Test Results for Strainer No. 3	-62-

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, 1968 through January 31, 1969. The work was performed under contract AT(04-3)-540 with the United States Atomic Energy Commission.

SECTION I

INTRODUCTION

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

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

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

Phase I—Preoperational research and development,

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

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

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

1. "Southwest Experimental Fast Oxide Reactor Development Program, First Quarterly Report, April - June 1964," GEAP-4594 (July 1964).
2. "Southwest Experimental Fast Oxide Reactor Development Program, Second Quarterly Report, July - September 1964," GEAP-4742 (October 1964).
3. "Southwest Experimental Fast Oxide Reactor Development Program, Third Quarterly Report, October 1964 - January 1965," GEAP-4799 (February 1965).
4. "Southwest Experimental Fast Oxide Reactor Development Program, Fourth Quarterly Report, February - April 1965," GEAP-4864 (May 1965).

5. "Southwest Experimental Fast Oxide Reactor Development Program, Fifth Quarterly Report, May - July 1965," GEAP-4934 (August 1965).
 6. "Southwest Experimental Fast Oxide Reactor Development Program, Sixth Quarterly Report, August - October 1965," GEAP-4994 (November 1965).
 7. "Southwest Experimental Fast Oxide Reactor Development Program, Seventh Quarterly Report, November 1965 - January 1966," GEAP-5108 (February 1966).
 8. "Southwest Experimental Fast Oxide Reactor Development Program, Eighth Quarterly Report, February - April 1966," GEAP-5160 (May 1966).
 9. "Southwest Experimental Fast Oxide Reactor Development Program, Ninth Quarterly Report, May - July 1966," GEAP-5208 (August 1966).
 10. "Southwest Experimental Fast Oxide Reactor Development Program, Tenth Quarterly Report, August - October 1966," GEAP-5301 (November 1967).
 11. "Southwest Experimental Fast Oxide Reactor Development Program, Eleventh Quarterly Report, November 1966 - January 1967," GEAP-5442 (February 1967).
 12. "Southwest Experimental Fast Oxide Reactor Development Program, Twelfth Quarterly Report, February - April 1967," GEAP-5498 (May 1967).
 13. "Southwest Experimental Fast Oxide Reactor Development Program, Thirteenth Quarterly Report, May - July 1967," GEAP-5533 (August 1967).
 14. "Southwest Experimental Fast Oxide Reactor Development Program, Fourteenth Quarterly Report, August - October 1967," GEAP-5561 (November 1967).
 15. "Southwest Experimental Fast Oxide Reactor Development Program, Fifteenth Quarterly Report, November 1967 - January 1968," GEAP-5595 (February 1968).
 16. "Southwest Experimental Fast Oxide Reactor Development Program, Sixteenth Quarterly Report, February - April 1968," GEAP-5629 (May 1968).
 17. "Southwest Experimental Fast Oxide Reactor Development Program, Seventeenth Quarterly Report, May - July 1968," GEAP-5673 (August 1968).
 18. "Southwest Experimental Fast Oxide Reactor Development Program, Eighteenth Quarterly Report, August - October 1968," GEAP-5705 (November 1968).
- In addition, the following topical reports have been issued:
1. Novak, P. E., and Asamoto, R. R., "A Survey for a High Temperature Sensor for SEFOR," GEAP-4903 (July 1965).
 2. Quinn, E. P., "Vibration of SEFOR Fuel Rods in Parallel Flow," GEAP-4966 (September 1965).
 3. Johnson, M. L., "A Description of Fuel and Other Core Materials Procured for SEFOR Critical Experiments in ZPR-III," GEAP-5133 (December 1965).
 4. Protsik, R., "Epithermal Capture and Fission Distribution in a Fuel Rod - The Computer Code REAX," GEAP-4973 (April 1966).

5. Novak, P. E., and Asamoto, R. R., "An Out-of-Pile Evaluation of W-Re Thermocouple Systems for Use to 4700°F in Mixed Oxide," GEAP-5166 (May 1966).
6. Pflasterer, G. R., "SEFOR Experimental Program Planning, Vols. I and II," GEAP-5092 (August 1966).
7. Morrell, R. P., "SEFOR β/l Measurements by Noise Analysis Techniques," IDO-17210 (August 1966).
8. Williams, R. F., and Polomik, E. E., "Pressure Drop and Dye Tests with Water on the SEFOR Prototype Fuel Bundle," GEAP-5129 (February 1966).
9. Craig, C. N., Ryer, C. M., and Thompson, M. L., "Plutonium-Uranium Mixed Oxide Fuel Pellet Fabrication Development for the Southwest Experimental Fast Oxide Reactor," GEAP-5285 (October 1966).
10. Polomik, E. E., and Williams, R. F., "Coolant Surface Turbulence Study, SEFOR Reactor Vessel," GEAP-5099 (December 1966).
11. Wilkinson, L., "Use of Neutron Filters for Fast Reactor Fuel Irradiations in a Thermal Core," GEAP-5035 (April 1967).
12. Wilkinson, L., et al., "Devices for Cyclic and Transient Power Control in a Steady-State Neutron Flux Gradient," GEAP-5036 (May 1967).
13. Reynolds, A. B., and Stewart, S. L., "Analysis of the SEFOR Mockup Critical Experiments in XPR-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-R3 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).

SECTION II

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:

<u>TASK</u>	<u>TITLE</u>
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 4—SPECIAL INSTRUMENTATION

All equipment for the transient reactor power detection system has been completed, checked out, and shipped to the SEFOR site.

2.3 TASK 6—REACTIVITY CONTROLS REQUIRED FOR EXPERIMENTATION

The testing of the FRED reactor model is complete. The drywell, positioner, and other miscellaneous equipment have been shipped to the SEFOR site. The control equipment for the balanced oscillator tests was completed and shipped to SEFOR.

The only remaining equipment to be shipped are the FRED device, the oscillator mechanism, and the light-beam oscillograph.

2.4 TASK 9—FUEL FABRICATION

The first instrumented fuel assembly was delivered to the SEFOR site. A representative of Nuclear Fuel Services (NFS) corrected minor deficiencies (concerning fuel rod end-plug straightness) on 20 fuel rods at the SEFOR site. NFS is scheduled to submit their proposal to GE for repair of the 28 leaking fuel rods (see the 18th Quarterly Report) in March 1969.

2.5 TASK 10—STARTUP TESTING

A recalculation of the total SEFOR reflector control strength (using transport theory) showed the predicted worth to be in good agreement with previous diffusion theory calculations (11.2\$).

Recalculation of the initial minimum critical mass yielded 285 kg fissile Pu (519 fuel rods), which is also in good agreement with the previous prediction (512 fuel rods).

The asymptotic reactor period versus excess reactivity relationship (in-hour curve) was calculated for the SEFOR minimum critical core.

2.6 TASK 11—TRANSIENT TESTS

The AEC cask, B of E 1233, rented from the AEC for shipping radioactive fuel,

was cleaned and repainted on the outside. The inside was decontaminated and the cask is scheduled for shipment to SEFOR during the next quarter. The license application was completed for submittal to the AEC and DOT.

2.7 SEFOR OPERATIONS

The writing of Pre-operation Test Procedures was completed and the performance of the pre-operation tests was nearly

completed. All sodium systems (except the irradiated fuel storage tank) were filled, cold trapped, and circulated. Pre-irradiation dimensional inspection and gamma scan of fuel rods was completed. SEFOR personnel received seven AEC Senior Operator licenses and five Operator licenses. Emphasis in Operations is directed toward the preparation for initial startup and operation of the facility. Approximately half of the Task 10.3 Test Procedures (Zero Power Tests) were issued.

SECTION III

TASK 4—SPECIAL INSTRUMENTATION

3.1 TASK 4.2—CORE FLUX DETECTOR SYSTEM

3.1.1 Purpose

The objective of Task 4.2 has been to develop a suitable detector system for recording the transient reactor power during the SEFOR experimental program. The data will be used to determine the power, energy, and net reactivity as functions of time during the transient.

3.1.2 Past Accomplishments

Because of the special response requirements specified for this equipment, it was necessary to develop both the detectors and associated electronic amplifiers from a conceptual stage. Detector development included the building and testing of various assemblies in an attempt to determine optimum characteristics. Because of the lack of test facilities at very high power levels, this development program included the building of gamma detectors and fast neutron detectors which were sensitive to thermal neutrons in order to obtain current response at high levels for the final selected mechanical configurations.

The electronic equipment development consisted mostly of the development of an

amplifier which would have a fast response with a long (50-feet)-length of cable on the input. Prototypes of complete channels (detectors plus amplifiers) were built, and static and dynamic tests were conducted on the equipment.

3.1.3 Current Accomplishments

All equipment was completed during the quarter and shipped to the SEFOR site. Final electrical circuits are shown in Figures 3-1, 3-2, and 3-3.

The calibration unit was built so it could be used as a manual device or programmed by the data acquisition system. Each amplifier was statically calibrated on the linear and logarithmic scales.

The static calibration data was used for a least-squares fit to determine how closely the amplifiers matched the desired characteristic. For the logarithmic scale, the desired characteristic is:

$$V = A \log_{10} I + B$$

where:

V = output volts

I = input current

A and B are constants

FIGURE 3-1. SYSTEM ELEMENTARY DIAGRAM

FIGURE 3-2. AMPLIFIER ELEMENTARY DIAGRAM

The amplifiers were adjusted until the actual and desired characteristics (voltage output for a given current input) agreed within 1% over the required operating range (10^{-6} amps to 10^{-2} amps).

Rough drafts on the instruction manuals for this equipment were completed, and are being edited prior to publication.

SECTION IV

TASK 6—REACTIVITY CONTROLS REQUIRED FOR EXPERIMENTATION

4.1 TASK 6.1—FAST REACTIVITY EXCURSION DEVICE (FRED)

4.1.1 Purpose

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

4.1.2 Past Accomplishments

The FRED Model I Test Program was completed and the fabrication and assembly of the FRED reactor model was completed. The detail design and fabrication of the positioner was completed.

4.1.3 Current Results

4.1.3.1 FRED

The final phase of testing on the FRED reactor model consisting of both room temperature and heated tests, is complete. A total of 210 tests were performed in this final phase of testing. One series of tests was carried out varying the accumulator pressure over a wide range to determine the characteristics of the device.

After inspection of the data from the first series of tests on the reactor model, two pressures were established for a second series of tests with the transfer

chamber to simulate reactor operations. These accumulator pressures are 1500 and 600 psi which require that the transfer chamber be charged to 2000 psi and 800 psi, respectively. The results of these tests indicate the following performance at these two points:

<u>Pressure</u>	<u>Mean Travel Time (msec)</u>	<u>Range with $\pm 3\sigma$ deviation</u>
600 psi	144	133.5 to 154.5
1500 psi	99	94.5 to 103.5

These travel times were measured using the lift-off switch and the proximity switch since these are the primary position indication methods that are to be used during reactor testing. A photo-electric system and a lift-off analog signal are also used to provide additional information for the experimental program.

The results of these tests are shown in Figure 4-1 for 20-inch exit time versus the accumulator pressure. In all of the tests, the device exhibited the self-limiting characteristics that were present in the first test model.

4.1.3.2 Drywell

Fabrication of the drywell has been completed in accordance with the requirements of section III of the ASME Boiler and Pressure Vessel Code. The drywell has been shipped to the SEFOR Site.

FIGURE 4-1. FRED REACTOR MODEL TEST RESULTS (for 210 tests)

4.2 OSCILLATOR DRIVE

4.2.1 Purpose

The purpose of this task is to develop a device for introducing oscillating increments of reactivity into the core. The results of these tests will give information relative to the design requirements, including plant dynamic response and full power reactivity (temperature) coefficient.

4.2.2 Past Accomplishments

The testing of the oscillator mechanism was completed which included a series of heated tests to simulate reactor head temperatures. The fabrication of the balanced oscillator system was started.

4.2.3 Current Results

4.2.3.1 Oscillator Mechanism

The oscillator mechanism tests were completed during the previous quarter, and the mechanism is scheduled to be shipped to the SEFOR Site in February 1969. The positioner mechanism (used for both the FRED and oscillator devices was shipped to SEFOR).

4.2.3.2 Balanced Oscillator System

The fabrication of the balanced oscillator system was completed and all components shipped to the SEFOR Site.

4.2.3.3 Oscillator Poison Rod

The fabrication of the oscillator poison rod has been completed and it was shipped to the SEFOR Site.

SECTION V

TASK 9 - FUEL FABRICATION

5.1 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.

5.2 PAST ACCOMPLISHMENTS

The complete fuel rod order (756 standard and 18 25-percent plutonium enrichment rods) was received and inspected at SEFOR. Fifty fuel rods were rejected because of helium leaks and dimensional deficiencies. Negotiations were begun concerning repair by Nuclear Fuel Services (NFS). All core hardware (tightener sleeves, tightener rods, and extension rods) were received at SEFOR. The first two instrumented fuel rods were completed for the first instrumented fuel assembly.

5.3 CURRENT RESULTS

5.3.1 Fuel Rods

A manufacturing engineer from Nuclear Fuel Services (NFS) visited the SEFOR site and repaired 20 of the rejected rods by straightening the end plugs (on 17 rods) and filing the slot in the lower end plugs (on 4 rods) to allow proper orientation of the rods during remote handling and core loading operations. GE performed bend tests on fuel-rod end-plug weld samples to determine whether to accept the end plugs which had been straightened by the NFS engineer. The test consisted of two parts: (a) cold bending an end-plug sample 50 times through the largest angle (approximately 0.3 degree) actually found on the 17 straightened fuel rods, (b) cold bending an end plug specimen through an angle 100 times larger (30 degrees) than that in (a) above. After bending, the weld

samples were checked for helium leaks, and no leaks were observed. This test, in conjunction with calculations on the amount of cold work induced by the straightening operation and extensive experience at GE-APED on fuel rod straightening, has led to the conclusion that the straightened SEFOR rods are acceptable.

Further helium leak checks were made on the 28 leaking fuel rods to determine which end plugs were leaking. This information was requested by NFS to assist them in formulating their proposal for repairing these rods.

5.3.2 Fuel Pellet Gas Content

As discussed in the previous quarterly report (GEAP-5705), GE and ORNL obtained 60 $\mu\text{l/gm}$ to 100 $\mu\text{l/gm}$ (specification limit is 50 $\mu\text{l/gm}$) gas release values on two sample fuel pellets taken from one production outgassed rod (Rod 289, outgassed in furnace load 10) while measurements performed by GE and NFS on pellet samples taken from 15 pre-production and production outgassed rods showed the pellet gas content to be within specification. During the present reporting period GE performed additional gas release measurements on pellet samples (3 grams taken from a 60 gram pellet) taken from four fuel rods which had been outgassed in furnace loads 8, 9, and 10. These tests, aimed at determining the extent of the apparent high-pellet gas content problem, resulted in 15 values in the range 12 $\mu\text{l/gm}$ to 49 $\mu\text{l/gm}$ and 8 values in the range 54 $\mu\text{l/gm}$ to 69 $\mu\text{l/gm}$. The gas constituents were found to be approximately 45% hydrogen, 35% nitrogen, and 20% carbon monoxide. The variation in the gas release results is not caused by the gas measurement equipment since a gas measurement standard deviation of $\pm 4\%$ has been demonstrated using National Bureau of Standards samples. Rather, the variation in small sample gas

release results is thought to be caused by surface contamination of the pellets which occurred during pellet fabrication. Fuel rod cladding stress calculations have shown that a 100 $\mu\text{l/gm}$ pellet gas content produces tube wall stresses below that allowed by the ASME Pressure Vessel Code Section III. The pellet gas content tests described above have shown no indication of gas content in excess of 100 $\mu\text{l/gm}$, the average test value being more nearly 50 $\mu\text{l/gm}$.

Because of the wide variation observed in individual pellet-sample gas-release measurements (a factor of two observed between high and low values obtained from seven samples taken from one pellet), work was begun to obtain gas release measurements on samples composed of several whole pellets. As the quarter closed, plans were made to measure the gas pressure resulting from heating a tube containing nine production outgassed fuel pellets to 1000° F.

5.3.3 Instrumented Fuel Assemblies

The first instrumented fuel assembly (IFA) was completed and shipped to SEFOR.

5.3.3.1 Instrumented Fuel Rods

Fabrication of fuel rods containing fuel thermocouples continued. The broken bare-wire thermocouple mentioned in the 18th quarterly report continued to delay completion of the second set of fuel rods (two rods per set). Figure 5-1 shows the broken thermocouple extending from the upper section of the partially fabricated fuel rod. A resistance welding technique was developed to repair the broken W/25 Re thermocouple wire. A splice was successfully made in the wire, but during the repair, a second failure occurred in the W/5 Re thermocouple wire at a position where it entered the double bore insulator. Figure 5-2 shows the second break.

BROKEN FUEL THERMOCOUPLE

BROKEN FUEL THERMOCOUPLE

FIGURE 5-1. BAREWIRE FUEL THERMOCOUPLE - BREAK

FUEL THERMOCOUPLE

FUEL THERMOCOUPLE

FIGURE 5-2. FUEL THERMOCOUPLE AFTER SECOND BREAK

Further repair attempts were abandoned, and the fabrication of a replacement fuel rod was started. At the end of this quarter, the replacement fuel rod was in the latter stage of fabrication.

The other fuel rod in the second set was delayed in fabrication when the quality control x-ray indicated some tube wall thinning (at the clad-to-spacer weld). Extensive examination of the x-ray film was conducted, and sample welds (intentionally thinned for calibration samples) were used for further evaluation and measurement of wall thinning. These studies showed conclusively that the tube wall thinning was less than 10% and was acceptable for reactor service.

Figure 5-3 shows a typical modified fuel pellet. Fuel modification for six IFA's (twelve rods) was completed. Also, hot vacuum outgassing of the pellets was completed.

Figures 5-4 through 5-7 illustrate components and steps in fabricating the first two fuel rods. As shown in Figure 5-4, the thermocouples are received at the Plutonium Laboratory already brazed into the top end spacer. Fuel pellets are arranged in "stacks" or segment groups after being modified as required for the thermocouples. Figure 5-5 shows a complete set of fuel pellets as required for a fuel rod. Figure 5-8 shows the upper segment loaded with fuel pellets and the spacer pressed into the cladding tube end. The mechanical press for inserting the spacer plug is also shown in this figure. The thermocouple junction extending out of the upper section measures the fuel temperature in the middle segment.

Fabrication of the fuel rod begins with the thermocouple end spacer subassembly (Figure 5-4). The top, middle, and bottom fuel segments are loaded (in that order) with springs, reflectors, insulator

pellets, and fuel pellets. The final weld is a plug weld at the tip of the bottom end plug which closes the gas vent hole needed to prevent "blow-out" of the final weld.

Figure 5-7 shows the completed instrumented fuel rod.

5.3.3.2 Flowmeters

The flowmeters for the first six IFA's and two test flowmeters were completed. Figure 5-8 shows a large flowmeter just prior to final assembly. The electrical lead is shown entering the upper part of the housing and penetrating through to the interior prior to attachment to the frame sprayed inner tube. The ring magnets are shown above the flowmeter assembly in Figure 5-8. Figure 5-9 shows a completed flowmeter as viewed from the upper end. The small tube stub is a pinch-off tube used for Helium leak checking and back filling of the flowmeter.

5.3.3.3 Weld and Braze Development

Weld and braze development was completed. The entire IFA, completed to the seal block braze above the fuel rods, was contained in a pressure vessel for atmospheric control during the braze operation. The development of equipment and technique consumed a large amount of time and talent. Figures 5-10, 5-11 show the seal block braze. An example of a coolant thermocouple braze is shown in Figure 5-12. The upper bulkhead brazes were made in a vacuum chamber using an induction coil for heating as shown in Figure 5-13.

5.3.3.4 Assembly

A sequence of pictures taken during the assembly operation are included in this report.

1-1/8 inch MODIFIED PELLET

1-1/8 inch MODIFIED PELLET

FIGURE 5-3. TYPICAL MODIFIED 1-1/8 inch LONG PELLET

FUEL THERMOCOUPLES

FIGURE 5-4. FUEL THERMOCOUPLES

UPPER AND MIDDLE SEGMENT FUEL

MIDDLE SEGMENT FUEL

LOWER SEGMENT FUEL

FIGURE 5-5. MODIFIED FUEL READY FOR LOADING

FIGURE 5-6. UPPER FUEL ROD SEGMENT WITH SPACER INSERTED (Seen Through The Glove Box Window)

GEAP-5754

TOP SEGMENT

MIDDLE SEGMENT

BOTTOM SEGMENT

BOTTOM END PLUG

FIGURE 5-7. COMPLETED INSTRUMENTED FUEL ROD

FIGURE 5-8. INSTRUMENTED FUEL ASSEMBLY FLOW METER (EXPLODED VIEW)

FIGURE 5-9. INSTRUMENTED FUEL ASSEMBLY FLOW METER (Assembled View)

FIGURE 5-10. IFA SEAL BLOCK BRAZE OPERATION
(Chamber Head Removed)

FIGURE 5-11. IFA SEAL BLOCK BRAZE OPERATION (Through The Viewing Port)

FIGURE 5-12. COOLANT THERMOCOUPLE BRAZE

FIGURE 5-13. BULKHEAD BRAZE OPERATION

Figure 5-14 shows the spider viewed from the lower end, that portion which is welded to the flowmeter via two rings. The lower end plug of the tightener rod is welded to the backside of the spider. Note the T/C inlet coolant thermocouple junction protruding through the wall of the end plug.

Figure 5-15 shows the spider and flowmeter welded together. Flowmeter lead and the inlet coolant thermocouple junction are also shown in this picture.

After the tightener rod sections are welded together, the tightener sleeve is placed over the rod. Figure 5-16 depicts the tightener rod-and-sleeve assembly just prior to being welded to the flowmeter-and-spider sub-assembly. The BeO pellets are shown in the foreground. The BeO pellets, nickel reflector, and spring will be placed inside the tightener tube.

Upon completion of the flowmeter-tightener rod welding, the fuel rods are inserted into the spider. Note the inlet

coolant thermocouple protruding through the tightener rod and sleeve in the direction of the fuel rod as shown in Figure 5-17. Figure 5-18 shows the completed IFA in the final assembly fixture. Dimensional and alignment quality control measurements were taken from the surface plates supporting the assembly fixture.

5.3.3.5 Shipping

The IFA clamped into the transport fixture, which was in turn clamped into the inner shipping container, is shown in Figure 5-19. Special padding material was used between the IFA and the transport fixture clamps to prevent abrasion of the IFA during shipment. The entire shipping container arrangement was designed to limit IFA shock loading to less than two g's. The IFA was shipped on the truck that is used to transport UF_6 from Oak Ridge to San Jose. Figure 5-20 shows the structural steel frame supporting the IFA shipping container over the UF_6 containers.

FIGURE 5-14. SPIDER BEFORE ASSEMBLY TO FLOWMETER

FIGURE 5-15. FLOWMETER AND SPIDER SUBASSEMBLY

FIGURE 5-16. TIGHTENER ROD AND SLEEVE (Exploded View)

FIGURE 5-17. LOWER PART OF INSTRUMENTED FUEL ASSEMBLY

FIGURE 5-18. IFA IN THE ASSEMBLY FIXTURE

FIGURE 5-19. IFA READY FOR SHIPMENT

FIGURE 5-20. IFA SHIPPING CONTAINER ON TRUCK

SECTION VI

TASK 10—STARTUP TESTING

6.1 PURPOSE

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

6.2 PAST PERFORMANCE

Computer programs were written and debugged for analysis of the oscillator tests. Modifications to the SEFOR data acquisition computer programs in order to provide more flexibility in data handling at the SEFOR site were completed by a software vendor. Plutonium-239 foils for fission distribution measurements were delivered to the SEFOR site. The experimental uncertainty in the SEFOR Pu-239 alpha measurement was estimated, and an expanded experimental program for reducing this uncertainty from 15% to approximately 5% was studied. A recalculation of important SEFOR nuclear parameters using more recent nuclear cross sections and more sophisticated computer codes was initiated.

6.3 PREDICTED SEFOR MINIMUM CRITICAL LOADING

6.3.1 New Calculations

Calculations were done prior to SEFOR fuel loading to check the earlier predictions made of the size of the minimum critical configuration in the SEFOR fuel loading sequence⁽¹⁾ and of the excess reactivity that would be available in a SEFOR core fully loaded with fuel.⁽²⁾

Several changes have been made in the data used in GE-BRDO cross section files, as well as developments in codes

and analytical methods, since completion of the earlier work. The new calculations are intended to assess the effect of these data and methods changes and of small changes in the as-built core and fuel from the earlier design model on the loading predictions.

Analysis based on the new calculations done to date and their comparison with the earlier ones yields a predicted minimum critical mass of 285 kg Pu (239 + 241) at 350°F, corresponding to 80% of the full core (519) fuel rods). The expected range, which may be roughly interpreted as a standard deviation, is 265 kg to 307 kg (482 to 559 rods) corresponding to a ± 0.017 reactivity range. Calculated reactivity of the fully loaded core [355.7 kg Pu (239 + 241) or 648 rods] with all reflector segments up (most reactive) is $k_{\text{eff}} = 1.053$.

The predicted minimum critical loading of 519 rods is essentially the same as the 512 rods predicted from the earlier calculations.⁽¹⁾ The excess reactivity available in the fully loaded core (0.053) is consistent with the earlier design criterion which allowed an 0.035 reactivity uncertainty margin at 20 MWt, with about 0.018 required for operation between 350°F shutdown and full power plus 50% excess for experiments.*

Computational methods used in the present analysis, and not in the earlier design work, included an extension of 2-dimensional multigroup diffusion theory calculations from 4 to 13 energy groups,

* The 0.035 reactivity margin of the full core at 20 MWt is compensated by replacing some fuel rods (of the 648 in a full core) with boron carbide rods.

comparisons between 2-dimensional 4-group transport theory and diffusion theory calculations to improve estimates of neutron leakage effects, and one-dimensional transport theory (S_{16} with 24 energy groups) cell calculations to improve estimates of the heterogeneity effects in the SEFOR rod lattice and in the ZPR-III critical assembly mockup plate lattices. The transport theory leakage and heterogeneity corrections added about 0.01 to the calculated k_{eff} values of both SEFOR and the ZPR-III critical assembly mockup of SEFOR. Since the k_{eff} values calculated for SEFOR were normalized to those computed for the critical assembly, the transport and heterogeneity effects did not significantly alter the earlier critical mass prediction. Several small differences in composition between the as-built core and the earlier

computational model produced no significant overall calculated reactivity effect.

Most of the major differences between the cross sections used in this study and the ENDF/B⁽³⁾ file were summarized in the Eighteenth Quarterly Report.⁽⁴⁾ In addition to the changes noted there, the U238 capture cross section was lowered ~4% in the 200 keV range and was lowered from 5% to 10% in the range between 4 keV and 20 keV.

Calculations of k_{eff} were made for four SEFOR core sizes, covering a range from about 70% to 98% of the fully loaded reactor and for the ZPR-III critical assembly mockups of the SEFOR one-segment and two-segment core designs. Results are shown in Table 6-1 both before and

TABLE 6-1
CALCULATED k_{eff} OF SEFOR AND ZPR-III ASSEMBLY 47 MOCKUPS

Reactor Model**	Calculated k_{eff} *			Normalized k_{eff} ***
	2-Dimensional Diffusion		2-Dimensional Synthesis	
	4 Groups	13 Groups	13 Groups	
ZPR-III, 1-segment core	0.990	0.996	0.986	1.006
ZPR-III, 2-segment core	0.980	0.984	0.976	0.995
SEFOR, 71.9% of full core (350°F)	-	-	0.957	0.976
SEFOR, 78.34% of full core (350°F)	0.980	-	0.977	0.996
SEFOR, 80.0% of full core (350°F)	-	-	-	1.000****
SEFOR, 83.85% of full core (350°F)	-	-	0.992	1.011
SEFOR, 98.23% of full core (350°F)	-	1.039	1.028	1.048
SEFOR, 100% of full core (350°F)	-	-	-	1.053****

*"Calculated k_{eff} " includes correction factors listed in Table 2.

**SEFOR full core model contains 355.7 kg Pu (239 + 241), 648 fuel rods (108 fuel channels plus FRED channel in center), no boron rods. Core is assumed loaded from center outward in approximately cylindrical pattern.

***The normalization is based on a value of $k_{eff} = 1.0$ for an average of the ZPR-III 1-segment and 2-segment values.

****Value interpolated or extrapolated using curve of Figure 1.

after normalizing to the ZPR-III experiments. Normalized k_{eff} values are listed for a recommended normalization in which an average of the calculated k_{eff} values for the 1-segment and the 2-segment critical assemblies is made equal to unity.

Results given in Table 6-1 before normalization are based on two-dimensional multigroup diffusion theory in both 4 and 13 energy groups and also for a 13-group crude two-dimensional synthesis⁽⁵⁾ that combined one-dimensional flux and adjoint flux solutions performed along the orthogonal coordinates. Correction factors, including transport theory effects, which were applied to the diffusion results are shown in Table 6-2 and are discussed below.

The correction factors shown in Table 6-2 are for the following effects:

- a. Estimated over-prediction of core leakage by diffusion theory (based on comparisons between two-dimensional, 4-energy group S_4 transport calculations using the DOT⁽⁶⁾ code and two-dimensional, 4-group diffusion calculations using the DOT2DB⁽⁶⁾ code).
- b. High energy and low energy heterogeneity effects (over-and-above resonance self-shielding) of the ZPR-III plate lattice cells and the SEFOR rod lattice cell (based on one-dimensional, 24-group S_{16} transport cell calculations using the SN1D code⁽⁷⁾ and confirmed using an independent analytical model). Also included are estimated negative reactivity effects due to increased leakage associated with the lattice heterogeneity.

TABLE 6-2
CORRECTIONS TO DIFFUSION THEORY CALCULATIONS OF k_{eff}
BECAUSE OF TRANSPORT, HETEROGENEITY, GEOMETRY,
AND SELF-SHIELDING EFFECTS

Correction Factor	ZPR-III 1-Segment	ZPR-III 2-Segment	SEFOR ~80% of Full Core*	SEFOR Full Core
(a) Transport Core Leakage Effect	+0.0086	+0.0089	+0.0097	+0.0089
(b) Cell Heterogeneity				
High energy	+0.0067	+0.0067	+0.0042	+0.0042
Low energy	-0.0017	-0.0017	-0.0004	-0.0004
Leakage	-0.002	-0.002	-0.001	-0.001
(c) Excess Reactivity in ZPR-III Mockup Models	-0.0073	-0.0042	-	-
(d) Non-cylindrical Outer Periphery	-	-	-0.003	-0.003
(e) Doppler Effect (80°F to 350°F)	-	-	-0.003	-0.0033
(f) Overshielding Nickel Reflector	+0.001	+0.001	+0.001	+0.001
Total of Above Corrections	+0.0053	+0.0087	+0.0075	+0.0064

*These correction factors were applied to the four SEFOR cores in Table 6-1 having less than 85% of the full core loading. The "full core" correction factors were applied to the 98.23% and 100% cores.

- c. Reactivity reduction caused by partially withdrawn control drawer or nickel reflector segment in the ZPR-III critical configurations. (1) Such withdrawn control elements were not taken into account in the calculational model.
- d. Estimated reactivity reduction caused by non-cylindrical outer periphery of SEFOR full core and in the less predictable loading patterns approaching the minimum critical configuration. The calculational models were based on cylindrical approximations to the actual loadings.
- e. SEFOR Doppler reactivity effect from 300°K to 350°F. The Doppler broadened fuel material cross sections were computed at a temperature of 300°K because of limitations in the cross-section generation code.
- f. Estimated over-shielding of nickel reflector caused by neglect of surface effect.

The curve of Figure 6-1 is a plot of the calculated k_{eff} values (normalized to the ZPR-III assemblies versus percent of full core loaded. * The projected minimum critical configuration of 80% of the fully loaded core is the point where the curve crosses $k_{\text{eff}} = 1.0$. The basis for the expected ranges shown in Figure 6-1 is discussed below.

The predicted minimum critical configuration of 286 kg Pu (239 + 241) (see Figure 6-1) has an expected range corresponding to ± 0.017 in reactivity, which is

*The curve of Figure 6-1 lies close enough to that calculated earlier (Figure 3-5 of Reference 1) that the shape of the latter curve was used to extend that of Figure 6-1 to smaller core sizes than those included in Table 6-1.

roughly considered to be a standard deviation in the present state of the calculation. The ± 0.017 reactivity range is deduced from the following estimated components:

- a. Geometry defects in SEFOR ± 0.005
computational model. (Treatment of outer core periphery departure from cylindrical surface appears to be the largest defect.)
- b. Uncertainty in effect of BeO ± 0.003
reduction. (BeO side rods were replaced with steel rods, and this replacement was not represented in the ZPR-III mockup.)
- c. Uncertainty in fissile weight ± 0.003
of fabricated fuel.
- d. Uncertainty in effect of central FRED channel ± 0.003
(not present in ZPR-III mockup) and in spacing between fuel channels.
- e. Uncertainty in lattice heterogeneity effect differences ± 0.003
between the SEFOR rod system and the ZPR-III plate system.
- f. Uncertainties in differential neutron leakage (and other gap effects) between SEFOR and the ZPR-III 2-segment mockup, including effect of the enlarged region between core and reflector in the minimum critical core. (This uncertainty is accentuated by the persistent over-prediction of the negative reactivity of the fuel gap in going from the one-segment to the two-segment critical assembly, even with 2-dimensional transport theory calculations.)

FIGURE 6-1. k_{eff} VERSUS PERCENT OF FULL SEFOR CORE LOAD (Control Reflectors Up—
Temperature at 350°F)

- g. Sensitivity of the difference ± 0.005 in k_{eff} between the SEFOR cores and the ZPR-III critical assemblies to the specific multigroup calculation scheme used.

Correlating the three effects above, dependent mainly on neutron leakage, items a, d, and f, leads to propagated uncertainty of ± 0.017 .

Item f above is one of the largest uncertainties, and (as noted above) its magnitude is influenced by the apparent over-prediction of the negative reactivity effect of the fuel gap in the ZPR-III mockup. It is possible that the 2-dimensional, 4-group transport calculations used did not adequately reduce the difference in k_{eff} between the one-segment and the two-segment critical assemblies because of the inadequacy of a 4-energy group approximation for accurate treatment of the high energy leakage effects; or gap heterogeneity effects not properly taken into account in the calculational model may be responsible for a large part of the persistent over-prediction of the negative reactivity because of the fuel gap. (A 50% reduction in the U-238 capture in the gap would account for only 0.003 of the 0.011 reactivity effect.) Nevertheless, the over-prediction remains substantial and at present is not accounted for. For this reason, it appears that complete reliance for the prediction of k_{eff} in SEFOR should not be placed on normalization to only the ZPR-III, 2-segment experiment, which would assume that the calculation is entirely in error in its apparent over-prediction of the negative reactivity produced by the fuel gap.

Rather it appears highly possible that some other unidentified factors in the 1-segment or 2-segment mockups, or both, may not be properly accounted for in the computational model, and, hence, the

normalization of SEFOR k_{eff} values was made to both the 1-segment and the 2-segment critical assemblies.

6.3.2 Predicted Reflector Control Worth for SEFOR Minimum Critical Core

Total reflector control worth for the predicted minimum critical core was calculated using 4 energy-group, 2-dimensional, S_4 transport theory. These calculations were done prior to fuel loading to check the earlier prediction of the control worth. (1) The earlier prediction was based on diffusion theory calculations using modified boundary conditions to treat the voided region left when the reflectors are lowered. It also relied on extrapolation of experimental results obtained from the single-rod and the quadrant-rod measurements done in the ZPR-III critical assembly mockup of the full core. (2) With transport theory, the voided region with reflectors down is incorporated directly into the multigroup computational model, thereby providing a check of the earlier diffusion theory calculations by use of an appreciably different method.

The transport calculations gave a total reflector reactivity control worth of 0.036 for the predicted minimum critical core. This is in very good agreement with the earlier predicted value of 0.035 (11%).

Table 6-1 lists several values of total reflector control worth calculated earlier for the full core using different diffusion theory approximations and extrapolation from the ZPR-III critical assembly measurements. (2) Also listed are the total reflector control worths for the predicted minimum critical core calculated using modified-boundary diffusion theory (earlier calculation) and transport theory (present calculation). The calculated worths are all nearly the same. The calculations also indicate that the control

worth does not change very much in going from the full core condition to the minimum critical (i.e., the higher leakage of the minimum critical core and, hence, its greater sensitivity to reflector effectiveness relative to that of the full core appears to compensate the decrease in the reflector control worth caused by the increased distance between core and reflector.

The predicted total reflector reactivity control worth, based on the values listed in Table 6-3 and a consideration of the uncertainties in the calculations, is 0.036 ± 0.006 (11.2\$ $\pm$ 1.8\$). The ± 0.006 is an estimated standard deviation arising from possible consistent errors in the different calculations indicated in Table 6-3. These include uncertainties in the method used for condensation from the 60-group to the 4-group cross-section differences between the idealized cylindrical geometry of the computational model and that of the real reactor (and also some small composition differences), uncertainties in the treatment of nickel self-shielding (including the surface effect), and possible nuclear data

errors (due to data uncertainties) that may be sensitive to the difference in spectrum between the reflector-in and the reflector-out conditions.

The full core reflector worth is also predicted to be 0.036 ± 0.006 . The 0.036 value is about twice the amount of calculated control required for operation.

6.4 VENTURI FLOW METER CALIBRATION

In preparation for the main coolant flow measurements (using a water-calibrated venturi), the venturimeter calibration data were reviewed to assess the uncertainty in the sodium flow measurement caused by uncertainties in the calibration data and "as-built" venturi dimensions. Calibration curves and the associated errors for sodium flow in the 400°F to 1000°F temperature range were calculated from the water-flow data. For a given pressure drop, the calculated sodium flow error (assuming random, uncorrelated flow calibration and venturi dimension errors) was 0.15% of the measured flow over the 2000 gpm to 6000 gpm range.

TABLE 6-3
REFLECTOR CONTROL REACTIVITY WORTHS

<u>Case</u>	<u>Reactor Model</u>	<u>Method of Calculation</u>	<u>Total Control Reactivity Worth**</u>
1	SEFOR full	Extrapolation of ZPR-III experimental results on removal of nickel segments, using 4-group, 2-dimensional diffusion theory*	0.036 (11.2\$)
2	SEFOR full core	4-group, one-dimensional diffusion theory with modified boundary condition to account for voided region with reflectors down. (Gap homogenized with fuel regions.)*	0.037 (11.6\$)
3	SEFOR minimum critical	Same as Case 2.	0.035 (10.9\$)
4	SEFOR minimum critical	4-group, 2-dimensional, S_4 transport theory	0.036 (11.2\$)

*See Sections 6.2 and 6.3 for details and Reference 2.

**Conversion from Δk to \$ is based on a β_{eff} value of 0.0032.

6.5 IN-HOUR RELATION FOR SEFOR

Figure 6-2 shows the excess reactivity versus asymptotic period (in-hour curve) calculated for the SEFOR minimum critical core. Similar calculations are planned for the full size core. These calculations represent the current best estimate of the in-hour curve for SEFOR. However, it has been reported that Los Alamos has new data on the delayed neutron yields which, if the new data represents a significant difference from our present data, will be incorporated in the calculations.

The asymptotic reactor period (T) was calculated from the in-hour equation, taking into account 24 delayed neutron groups, 6 each for Pu-239, Pu-240, U-238, U-235:

$$\rho(T) = \frac{\Delta k}{k\beta_{\text{eff}}} = \sum_{q=1}^4 \sum_{i=1}^6 \frac{\beta_{\text{ieff}}^q / \beta_{\text{eff}}}{1 + \lambda_{iq} T} \quad (1)$$

$$\beta_{\text{eff}}^q = \frac{\beta_i^q \int_V \sum_k x_{D,i} \phi_k^+(r)}{\sum_{q=1}^4 \int_V \sum_k x_k \phi_k^+(r)} \quad (2)$$

$$\frac{\sum_j \nu_j^q \sigma_{f,j}^q N_j^q \phi_j(r) dV}{\sum_j \nu_j^q \sigma_{f,j}^q N_j^q \phi_j(r) dV} \quad (2)$$

$$\beta_{\text{eff}} = \sum_{q=1}^4 \sum_{i=1}^6 \beta_{\text{ieff}}^q \quad (3)$$

where:

$x_{D,i}$ = Energy spectrum of the i-th delayed neutron group.

x_k = Energy spectrum of the prompt neutrons

N = Atomic density

i = Index for the delayed neutron group

q = Index for the isotope

The asymptotic period reactivity relationships obtained from the above expressions are shown in Figure 6-2.

β_{ieff} was calculated by Equation (2) with the BISYN code. λ_{iq} , β_i^q and $x_{Di}(E)$ are taken from Keepin, "Physics of Nuclear Kinetics," Addison-Wesley Publishing Company, 1965. For β_i^q , the values for fast fission were used.

If the isotopes U-235 and Pu-240 were neglected, the approximate error would be $\Delta\rho/\rho = -3.5\%$ in the whole reactivity range from 1¢ to 35¢.

6.6 REFERENCES

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FIGURE 6-2. ASYMPTOTIC REACTOR PERIOD FOR SEFOR MINIMUM CRITICAL CORE

SECTION VII

TASK 11—TRANSIENT TESTS

7.1 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\$ 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.

7.2 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 was received from Atomics International and visually examined. Preparations were initiated to correct all deficiencies in the cask.

The design of all required equipment was completed. The required equipment includes:

- A shipping container licensed in accordance with the USAC Rules and Regulations Title 10, Part 71, "Packaging of Radioactive Materials for Transport," and Department of Transportation (DOT) Regulations (49 CFR Parts 170 - 179).
- A transfer tube to extend between the refueling cell valve and the cask, sealing to each, for maintenance of an inert atmosphere in the cask, refueling cell, and tube during the transfer of the fuel rods into the cask.

- Radiation shielding around the transfer tube.
- A vehicle for transportation of the cask from the loading dock into position beneath the vertical transfer port in the reactor building.
A cart is available at SEFOR and, with the addition of a top plate to distribute the cask load, will be used to transport the cask.
- Fuel rod shipping canisters.

7.3 CURRENT RESULTS

The cask was placed in the GE Radioactive Materials Laboratory for decontamination. A carbon steel canister (inadvertently left in the cask by Atomics International) was removed from the cask. The cask lids and inside were cleaned in preparation for shipment to SEFOR. The outside of the cask and the skid were cleaned and repainted. The fire shield insulation was replaced and the cask is ready for shipment to SEFOR.

The license application for AEC and DOT authorization to use cask B of E 1233 for shipment for inactivated SEFOR fuel rods has been completed.

The planned work for the coming period includes review and release of the license application and design drawings.

SECTION VIII

SEFOR OPERATIONS

8.1 PRE-OPERATION TESTS

The writing of Pre-operation Test Procedures was completed and all procedures were issued. The performance of all pre-operation tests has been completed except for No. 55/56/57, Sodium Hydraulics, and portions of No. 54, Sodium Flush; No. 22/23, Nitrogen Cooling; No. 20, Freon System; and No. 38, Special Startup Instrumentation.

Pre-operation Test No. 54, Sodium Flush, was interrupted because of difficulties with the strainers in the main primary loop. A discussion of these difficulties is given in Section Nine. Pre-operation Tests No. 22/23, Nitrogen Cooling, and No. 20, Freon Systems, will be completed when the nitrogen cells are closed and a sufficient heat load can be generated. Pre-operation Tests No. 55/56/57 will be completed after the successful performance of No. 54. With the correction of minor malfunctions and modifications, the tests have demonstrated the satisfactory performance of the equipment and the readiness for operation. All sodium systems were filled (except the irradiated fuel storage tank). Cold trapping and plugging measurements were made. Flows of 5000 gpm in the main secondary and 4600 gpm in the main primary and design flows in the service systems were attained. The argon supply and recirculation purifiers were filled with NaK.

8.2 FUEL INSPECTION

Pre-irradiation dimensional inspection and gamma transmission scanning of 66 standard and 17 guinea-pig rods have been completed in the Fuel Inspection Station within the Refueling Cell. This reference information will be used for comparison with data acquired from post-irradiation inspection throughout the program. The first Instrumented Fuel Assembly (IFA) was received at the site.

8.3 OPERATOR TRAINING AND LICENSE EXAMINATIONS

Oral examinations were conducted by the AEC Operator License Branch in November. SEFOR personnel were issued seven Senior Operator Licenses and five Operator Licenses. A continuing instruction and study program was initiated to prepare other Operations personnel for examinations.

8.4 OPERATIONS

As the performance of Pre-Operation Tests neared completion, greater emphasis has been directed to the preparation for operation of the facility and the performance of the test program. Procedures to perform the surveillance tests required by the Technical Specifications were developed, valves and process line labeling was initiated, and a Jumper Control Procedure was prepared and implemented. Start tests were performed for the Emergency Diesel Generator. The continuing review and revisions of Standard Operating Procedures was initiated.

8.5 PLANT ENGINEERING

Repair of equipment malfunctions, equipment modifications, and general site maintenance activities were accomplished, including the following: adjustment of sodium-pipe snubbers with sodium temperature of 700°F, stroking of sodium valves, calibration of the differential pressure sensor across the venturi in the main primary sodium loop, installation of isolation transformers on sodium level probe power supplies, repair and realignment of argon compressor, modifications to valves in the nitrogen cooling system, modifications to the controls for the Freon system, elimination of leaks in the various gas systems, modifications to the main air blast cooler doors, gamma compensation measurements

on the neutron chambers, checkout of the He^3 neutron chambers (to be used in center channel for core loading), fabrication of fuel transfer cans (to transport fuel rods from the new fuel storage vault to the refueling cell), preparation of Preventive Maintenance Procedures, instrument repairs, and ordering of spare parts. The strainer (100 mesh) initially installed in the main primary sodium loop was replaced by a second strainer (80 mesh) which has been removed to be replaced by a third strainer (50 mesh). The strainer initially installed in the auxiliary primary system was removed from the pipe.

8.6 EXPERIMENTAL PROGRAM

The checkout and correction of computer codes (prepared by SEL) continued. The writing of Test Procedures (Program) continued with the following procedures issued:

- I-A. 1 (Rev. II) Initial Core Loading
- I-A. 2 Inherent Source Measurement
- I-A. 3 Reflector Worth Measurements

I-A. 4 Material Worth Measurements

I-B. 1 Full Size Core Loading

I-B. 2 Reflector Worth Measurements

I-B. 3 Neutron Instrumentation Calibration

I-B. 4 Isothermal Temperature Coefficient

All of the remaining (approximately 10) procedures in Group I (Zero Power) have been drafted and are being reviewed. A Test Performance Plan has been prepared and issued. A preliminary check-out of the Reactivity Meter was made using an analog computer.

8.7 RADIATION AND INDUSTRIAL SAFETY

A whole body reference base count was obtained on each of the site personnel who will have access to potential contaminated areas. A series of First Aid classes was initiated. A draft of the Radiation and Industrial Safety Manual is being reviewed.

SECTION IX

SEFOR SODIUM STRAINER EXPERIENCE

9.1 INTRODUCTION

A temporary full-flow screen type strainer was installed in each of the SEFOR primary sodium loops with the objective to collect any construction debris that inadvertently may have found its way into the coolant system during installation. It was planned that prior to fuel loading, the sodium would be circulated up to full flow for a minimum of eight hours after which the system would be drained of sodium and the strainers with any collected debris removed from the system.

Strainer location in each loop was made to coincide with the coolant system final closure welds where essentially zero torque was required for pipe positioning prior to welding. The main loop strainer was located in the horizontal 10-inch pipe near the discharge of the 5000 gpm pump. The auxiliary loop strainer was located in the horizontal 4-inch pipe near the inlet side of the auxiliary pump.

On the basis of specifications established for the sodium coolant systems, the following requirements were contained in the specification for the strainer assembly:

Material:	Type 304 stainless steel
Screen Size:	100 mesh
Flow capability:	Up to 100% loop flow
System design pressure:	100 psig
System design temperature:	700°F
Maximum ΔP (maximum pump discharge pressure):	50 psi

Selection of 100 mesh screen size was based on earlier experience with screens used for sodium system flushing. Previous experience with large sodium loops using 80 mesh screen has resulted in successful operation and screens of 100 mesh size had been satisfactorily used in small scale test loops. The principal design objective of the screen was to remove all foreign particles from the system that might be large enough to cause possible flow restriction. This design objective requires the maximum screen opening to be smaller than the minimum coolant passage restriction. This condition would be satisfied by screens with much larger hole sizes (8 to 16 mesh). The selection of the 100 mesh screen size was therefore a very conservative choice motivated by past successful experience in a similar application.

For convenience in identifying the various filter screens which have been used or are planned for the pre-operational test program, the following designations are used in the report:

- 1A - Initial strainer in main primary coolant loop.
- 1B - Auxiliary primary coolant loop strainer.
- 2 - Second strainer in main primary coolant loop.
- 3 - Third strainer in main primary coolant loop.

9.2 STRAINERS 1A AND 1B

9.2.1 Description

Strainers 1A and 1B were of a truncated cone configuration as shown in Figure 9-1. In both cases, the screen

(1A)

(1B)

FIGURE 9-1. SODIUM STRAINERS NO. 1A AND NO. 1B

assembly was constructed of an inner 100 mesh screen backed by a 4 mesh screen with the inlet end welded to a ring which in turn was welded to the pipe inner diameter. The 100 mesh screen was constructed of 4-1/2 mil diameter wire with a 5-1/2 mil spacing between adjacent wires. The 4 mesh screen was constructed of 80 mil diameter wire with 0.170 inch spacing between adjacent wires. Screen 1A was positioned near the discharge end, on the outside surface, by means of a cylindrical ring which was attached to the inner pipe surface through three equally spaced supports. All the strainers (1A, 1B, 2 and 3) were constructed of type 304 stainless steel.

Independent mechanical strength calculations showed that the backup 4 mesh screen used for strainer 1A could withstand 57 psi across the element without excessive stress, satisfying the design requirement that the screen could withstand the 50 psi (maximum) pump discharge head.

Vendor catalog information for the 100 mesh screen indicated a clean pressure drop of 1 psi for 17 gpm/in.² of screen area. On this basis, a clean pressure drop of about 2 psi was predicted for strainer 1A at 5000 gpm flow.

The initial strainers were fabricated and welded in short pipe sections (Figure 9-1) by Dravo Corporation, the shop fabricator for the stainless steel piping used in the SEFOR sodium systems.

9.2.2 Operation Summary

After initial filling of the SEFOR Primary Sodium System and before any significant flow was established, the sodium temperature was increased to 550°F to "wet" the system, cold trapping was initiated and maintained until a plugging temperature of 270°F was obtained.

Over a two day interval, the main primary flow was established at various flow rates up to 4000 gpm with fluctuations and drifting experienced; flow in the auxiliary coolant loop was maintained at about 180 gpm. The main loop flow decreased from 3200 gpm to 1100 gpm for the same voltage (~315V) and current (395 amps) settings on the EM pump, indicating that the strainer was becoming clogged. Because the initial installation did not permit pressure measurements at the strainer inlet, the flow was reduced to approximately 500 gpm while a determination was made to verify that the cause of the flow behavior was high pressure drop across the strainer.

A check of the voltage and current values at the main primary pump and comparison with the main secondary pump (which is identical to the primary and which showed good agreement with predicted pump performance curves) verified that the pump was operating satisfactorily and that a high pressure drop (approximately 20 psi) existed across the strainer. Following this determination, the temperature was decreased, the primary coolant systems drained, and the pipe section containing strainer 1A was cut out. The excess sodium at the strainer was removed mechanically and the residue reacted with alcohol. A sample of debris (~2 ounces volume) was removed from the strainer and submitted for analysis. The 100 mesh screen was loose from the retaining ring at the inlet for about one-third of the circumference but appeared to have remained in place, pressed against the outer screen.

9.2.3 Discussion of Results

The decrease of main loop flow with constant applied voltage to the EM pump during operation supported the hypothesis of buildup of particulate at the strainer.

After the strainer was removed and cleaned of sodium, 50 to 100 grams of insoluble residue were obtained. The size and number of particles contained in this residue provided strong support for the hypothesis that the flow behavior observed was caused by gradual partial plugging of the screen.

The residue from Strainer 1A, after subsequent reaction with alcohol and water, consisted principally of a mixture of small metallic particles ranging from approximately 1 mil diameter to approximately 1/16-inch long and some very fine particulate of varying colors of gray, green, brown, and black. The residue was examined by X-ray diffraction and emission spectography.

The analysis showed approximately 85% of the residue to be metallic, consistent with the austenitic stainless steel composition. The remaining 15% of the residue was determined to be refractory material, principally consisting of aluminum oxide.

The results of the analysis of the residue are consistent with the hypothesis that fine particulates were introduced into the system from the grinding of the stainless steel nozzle at the main IHX in the preparation of a new weld end.

The initial sodium flush with strainer 1A did not permit attainment of full flow as specified in the pre-operational test. It was considered desirable to attain full flow with a strainer in the system to provide assurance that particles which might still be in the system and capable of being transported with higher sodium flow rates are removed from the system. Therefore, it was concluded that a new strainer should be installed in the main coolant loop and the pre-operational flush test carried to completion. In addition, since strainer 1B

appeared to be operating satisfactorily without evidence of excessive clogging, it was planned that this strainer would not be removed until full flow was achieved in the main loop.

9.3 STRAINER 2

9.3.1 Mechanical Description

The design objectives for strainer 2 were identical to those for 1A (see Section 9.1) but with the added requirement to increase the screen area in strainer 2 by about a factor of three to help assure adequate particulate removal capacity so as to avoid the excessive plugging experienced in strainer 1A. In order to reduce fabrication problems associated with meeting this additional requirement, the cartridge geometry was changed from a cone shape to a right circular cylinder. The final configuration chosen is shown in Figure 9-2.

The body of the strainer was made from 10-inch, schedule 20, type 304 stainless steel which conformed to the requirements for the SEFOR main primary loop piping. The ends were prepared in conformance with SEFOR welding specifications so the unit could be butt welded directly into the existing piping.

The strainer cartridge was fabricated by SEFOR site craft personnel as two concentric cylinders approximately 23-1/2 inches long and 7 inches in diameter. The outer (backup) cylinder was 4 mesh, 304 stainless steel screen with 0.170-inch square opening and 0.080-inch diameter wire.

The inner cylinder was fabricated from 80 mesh, 304 stainless steel screen with 0.007-inch square opening and 0.0055-wire diameter. The 80 mesh screen was selected (rather than the 100

FIGURE 9-2. SODIUM STRAINER NO. 2

mesh screen used in strainer 1A) to increase the strength of the cartridge while maintaining adequate flow area through the cartridge.

The two screens were connected to support rings located at both the upstream and downstream end using tungsten inert gas welding techniques. The same welding process was used in fastening the longitudinal seam joints for both screens. The upstream support ring was then welded to the 10-inch pipe body. Near the downstream end, a support ring with spider attachments was welded to the 10-inch pipe for additional cartridge support.

To aid in evaluating strainer performance during operation, a differential pressure transmitter with 10 psi full scale readout, available from GE's San Jose sodium test facility, was installed across the strainer at previously existing pressure taps.

As stated previously, the design objective for Strainer 2 was to increase the open hole area by a factor of 3 above that of strainer 1A to obtain increased particulate retention capability. For the as-fabricated configuration described above, the open hole area was increased by a factor of approximately 2.8.

9.3.2 Design Summary

The mechanical strength for the backup 4 mesh screen used with strainer 2 was evaluated to assure that the design requirements had been met. These calculations indicated that (based on ASME Code equations) the strainer cartridge could withstand a differential pressure of 81 psi. This value provided an adequate safety margin over the 50 psi ΔP design requirement.

Based upon vendor ΔP data for wire cloth, the pressure drop through the screen

was calculated to be less than 1 psi. Hydraulic calculations to include contraction losses indicated the overall pressure drop to be in the range of 5 to 10 psi. This value was considered acceptable since the preliminary hydraulic data from the operation with strainer 1 indicated that the pump would be capable of delivering full flow with an added system pressure drop of about 15 psi.

9.3.3 Operation Summary

The plugging temperature in the primary drain tank prior to filling the primary coolant system was 255°F. After filling the primary systems at a temperature of 350°F, the plugging temperature reached approximately 300°F. With the main loop at 1550 gpm, a ΔP of approximately 9 psi was indicated across strainer 2. The flow was reduced to about 950 gpm, cold trapping and system heat-up were initiated. Approximately nine hours later at about 400°F and 1500 gpm, the ΔP had decreased to 3 psi. Approximately 16 hours later, at a temperature of 550°F and 3000 gpm, a ΔP of 10 psi was measured indicating no further reduction in screen resistance. Temperature was increased to 765°F. A maximum flow of 4600 gpm (limited by pump voltage) was maintained for about nine hours without voltage change. Plugging temperature had been reduced to 240°F. The temperature and flow were decreased and the loop drained. An auxiliary drain line was installed to drain the residual sodium (approximately 20 gallons) from the strainer body and horizontal pipe into containers. The pipe section and strainer were removed from the loop.

9.3.3.1 Description of Failure

The outer (4 mesh) screen was intact but the inner (80 mesh) screen had torn into many pieces. The photograph in Figure 9-3 shows the intact 4 mesh screen

FIGURE 9-3. STRAINER No. 2 AFTER REMOVAL FROM SODIUM LOOP

with the fragments of fine mesh trapped at the discharge end. A large number of pieces of the failed screen were rectangular shaped and ranged up to about 1-inch wide and several inches long. The inlet end of the 80 mesh screen for a length of about 1-1/2 inches was firmly attached to the plate support and clamping ring. The outlet end was also firmly attached to the 4 mesh screen. Approximately 7 inches of the 80 mesh screen was still attached to the down stream end of the 4 mesh screen.

The strainer was washed in alcohol to react the residual sodium. Steps were initiated to collect and inventory all of the pieces of the torn screen section. The total weight of the 80 mesh screen was calculated to be 203.1 grams; 151.5 grams of this total have thus far been collected. Equipment is being procured and fabricated to melt and strain the 20 gallons of sodium which comprised the heel of sodium that remained after initial loop draining in an effort to recover any pieces of the screen that may have collected in the sodium heel during draining of the system.

9.3.3.2 Discussion of Data Obtained

A differential pressure gage was installed to measure the pressure drop across the second strainer. Data obtained during the tests are shown in Figure 9-4. These data show an initially high strainer pressure loss of about 9 psi at 1500 gpm. At the time of the test, this was attributed to oxide plugging of the screen, and cold trapping of the system was initiated. Flow was reduced to 1000 gpm and held there for about 6 hours. During this time, the strainer pressure loss was observed to decrease and the flow rate to increase at constant voltage. This was interpreted as clearing of the oxide plugging. Subsequent observation of the failed screen and analysis of the pressure drop data indicate that the screen failure may have been in

progress during this time. This conclusion is based on the observation that no measurable change in pressure loss characteristics occurred during the remainder of the flushing operation with strainer No. 2. This hypothesis cannot be proven conclusively, however, because analysis of the estimated pressure loss for this strainer indicates that the pressure drop due to the 80 mesh screen itself was only a small part of the total pressure drop. Thus, failure of the screen would not cause a significant reduction in pressure. These results support the assumption that oxide plugging contributes to the initially high pressure drop and leaves the actual time of screen failure in doubt.

A recheck of the pressure drop calculations after strainer No. 2 operation showed that the predicted clean ΔP could be as high as 19 psi using more conservative loss coefficients. In addition, a more comprehensive check of vendor data for filter screens indicated a wide variance in flow versus pressure drop data. For example, on the basis of data supplied by one vendor, the pressure drop was calculated to be 109 psi for the flow rate and screen geometry of strainer No. 2. Data from other vendors yielded much lower pressure drop values (in the order of 20 psi).

Table 9-1 summarizes the hydraulic parameters for strainer No. 2.

The estimated pressure loss of 19 psi at 5000 gpm is less than the measured values at low flows projected to 30 psi at 5000 gpm for data obtained at the end of the test and considerably less than the projected initial data of 99 psi at 5000 gpm. (Based on 8.9 psi at 1500 gpm.)

9.3.3.3 Analysis of Failure Mode

The screen failure was most probably caused by extreme turbulence on both

FIGURE 9-4. PRESSURE LOSS VERSUS SODIUM FLOW RATE - STRAINER NO. 2

TABLE 9-1
HYDRAULIC PARAMETERS FOR STRAINER NO. 2

<u>Location</u>	<u>Flow Area</u>	<u>Velocity at 5000 gpm</u>	<u>Velocity Head</u>	<u>Loss Coefficient, C</u>
10-in. Pipe	82.5 in. ²	19.5 ft/sec	2.2 psi	0.5
7-in. Screen entrance	38.5	41.9	10.0	1.0
Screen (open area)	178	9.1	0.47	1.0
Downstream Annular Passage	44.1	36.5	7.6	1.0

$$\Delta P = 0 \times V.H.$$

$$\text{Estimated } \Delta P = (0.5 \times 2.2) + (1.0 \times 10) + (1.0 \times .47) + (1.0 \times 7.6) = 19.2 \text{ psi}$$

sides of the screen. The calculated velocities of 41.9 ft/sec and 36.5 ft/sec (see Table 9-1) are very high, and the associated velocity heads could have caused a cyclic pressure load across the screen. This load would alternately tend to collapse and then expand the screen against the 4 mesh backup screen at a very high frequency. Screen failure would then occur because of fatigue. The rectangular shape and size of many pieces of the failed screen provide evidence of this mode of failure. The square weave construction of the 4 mesh screen provides point or line support for the fine screen, and therefore, results in many points of high stress.

9.4 STRAINER NO. 3

As a result of the failure of the 80 mesh screen strainer No. 2, a sodium flush at full flow as specified in the pre-operational tests which would satisfy the objectives stated in Section 9.1 was not achieved. It was concluded, therefore, that a third strainer should be installed in the main coolant so the pre-operational flush test could be completed. The design and construction of strainer No. 3 is described in the following section. Except for screen size, the design requirements for strainer No. 3 included all of those

specified for strainers No. 1A and No. 1B. In addition to these requirements, the observations and analyses described in Section 9.3.3.3 led to the additional design requirements for strainer No. 3 of low velocity in the strainer and perforated plate support for the fine screen.

9.4.1 Mechanical Description

This strainer consists primarily of a cylindrical screen with a diameter of 18 inches and a length of 42 inches. The screen is held in a cylindrical container, 24 inches in diameter (See Figure 9-5). The sodium enters the cylinder through a 10-inch pipe at one end, then flows radially outward to the annulus between the screen and the container to a 10-inch discharge nozzle on one side of the container. The screen is held in place by eccentric rings at each end of the container and is placed off center so velocities and pressure losses near the exit will be kept low.

The 50 mesh screen used for this strainer is supported by a 1/8-inch thick perforated plate on the outside and a 4 mesh screen on the inside. The fine mesh screen size was changed to 50 mesh for strainer No. 3 to take advantage of the higher strength of the 50 mesh screen

FIGURE 9-5. SODIUM STRAINER NO. 3

compared to that of the screen used in strainers No. 2 and No. 1A. The 50 mesh screen still meets the overall objective of the sodium flush tests, however, (as summarized in Section 9.1) which was to remove any loose material from the primary coolant system that (a) could be moved by the primary coolant flow at full flow conditions and (b) that has dimensions that are of the same magnitude as the minimum dimensions in the core. The 50 mesh wire has a minimum dimension of 0.011 inch which is about one-sixth the size of the minimum spacing (0.065-in.) in the core. Local reductions in cooling of the fuel as a result of flow perturbations caused by material that could get through the 50 mesh screen would be less than that caused by the spacer buttons, dimples, and tightener springs that are used to support the fuel rods.

The perforated plate was chosen to provide the required strength, and the 4 mesh screen was chosen to avoid any possibility of a failure in the 50 mesh screen because of turbulence or pressure fluctuations. Each of these three components is supported by a ring at either end of the cylinder.

The perforated plate was formed into a cylinder with the rounded edges of the die-punched holes on the inside. The sharp edges on the outside of the cylinder were de-burred to assure system cleanliness.

The fine mesh screen was rolled into a cylinder with an overlapping seam. The seam was welded, and the screen was then inserted into the perforated plate cylinder.

The coarse mesh screen was rolled into a cylinder and inserted inside the assembly before the seam was welded. This assured a tight fit of all three screens. Each wire of the coarse screen was welded

to form its seam. Extreme care was exercised during this procedure to avoid damage to the fine screen.

The screen diameter was set at 18 inches so that the upstream end of the screen would be sufficiently far from the entrance nozzle to avoid damage due to turbulent flow conditions. The large diameter, combined with the length of the screen, provide a large volume in which the entering velocity of about 20 feet per second (at 5000 gpm) can be dissipated without excessive pressure loss. The screen dimensions also provide sufficient open area to reduce the average velocity through the screen to about 6 ft/sec. Velocities of less than 10 ft/sec are generally recommended by strainer manufacturers.

Pressure losses for this strainer are estimated to be 3-1/2 psi at 5000 gpm in sodium, or 4 psi at 5000 gpm in water. This estimate is based on an entrance loss of 1 velocity head (20 ft/sec), a screen loss of 1 velocity head (6 ft/sec), and an exit loss of 1/2 velocity head (20 ft/sec). Data obtained in the water test facility indicate that these values are reasonable (see Section 9.4.3.3).

Following tests in the water facility, the outer shell of the casing was thoroughly cleaned, and a new screen assembly was installed. The completed unit was then pressure tested and coded for 100 psig service and shipped to the SEFOR site for installation in the main primary sodium loop.

9.4.2 Design Summary

Screen Length	42 in.
Screen Diameter	17-5/8 in.
Shell Length (inside)	44 in.
Shell Diameter	24 in.
Perforated Plate:	
Thickness	0.125 in.
Hole Size	1/4 in.

Hole Spacing	3/8 in. (staggered)
Open Area	40.3%
Fine Screen:	
Square Mesh	50
Width of Opening	0.011 in.
Wire Diameter	0.009 in.
Open Area	30.3%
Coarse Screen:	
Square Mesh	4
Width of Opening	0.170 in.
Wire Diameter	0.080 in.
Open Area	46.2%
Screen Assembly:	
Total Screen Area	2240 sq. in.
Open Area	274 sq. in. = 2240x0.303x 0.403
Fluid Velocity	5.8 ft/sec at 5000 gpm
Velocity Head	0.2 psi (Na at 600°F)
Strainer Assembly:	
10-in. Pipe Velocity	19.5 ft/sec at 5000 gpm
Velocity Head	2.2 psi
Estimated Pres- sure Loss	3.5 psi at 5000 gpm, Sodium
	4.0 psi at 5000 gpm, Water

9.4.3 Test Program Prior to Sodium Use

To verify the integrity and performance characteristics of strainer No. 3, several water tests were conducted prior to strainer use in the SEFOR primary system.

9.4.3.1 Description of Test Facility

The facility chosen for the water test work was the boiling water reactor (BWR) core spray test facility located at the Vallecitos Nuclear Center near Pleasanton, California. The primary purpose of this facility is to test the BWR emergency core cooling systems presently being developed for use on the large, commercial boiling

water reactors being built by the General Electric Atomic Power Equipment Department (APED). A schematic of the facility piping arrangement is shown in Figure 9-6.

Suction for the main pump is taken from an 8000-gallon storage tank over the top of which is located a mockup of the BWR core structure. Pump discharge is routed back to the storage tank after passing through a flow orifice, throttling valve, emergency cooling headering arrangement, and emergency cooling discharge nozzles. The main pump is rated at 4800 gpm and 103 psig discharge pressure. A 125 hp fan is also provided which discharges to the space between the bottom of the mocked-up core and tank water level. This air then passes up through the core and simulates rising steam which would exist under actual emergency cooling conditions.

In order to use this facility for strainer testing purposes, a new 10-inch line was routed from a point downstream of the throttling valve to the strainer and an additional 10-inch line was routed from the outlet of the strainer to the 8000-gallon storage tank. The line was routed inside the tank to provide good mixing. (Good mixing was considered necessary in order to dissipate pump energy more efficiently.)

Instrumentation taps were provided at the inlet and outlet of the strainer on the 10-inch piping. A 0-100 psig Heise gage was used for monitoring strainer inlet pressure, and a mercury manometer was used for monitoring outlet pressure. Normal loop instrumentation was used for measuring such parameters as strainer flow, pump discharge pressure, and loop temperature.

In addition, a small 100 gpm Cuno filter was installed on a 2-inch bypass line for water cleanup purposes. Pressure gages were installed on each side of the filter, and a rotometer was installed on

FIGURE 9-6. WATER TEST FACILITY PIPING ARRANGEMENT

the filter outlet line in order to monitor filter ΔP and flow rate. This cleanup system was run for several hours prior to the strainer test runs to insure a clean system.

9.4.3.2 Discussion of Tests Conducted

Three types of tests were run, including a hydraulic flow versus pressure drop test, a long-term endurance test, and a plugging test. The hydraulic flow versus strainer ΔP tests were run on an intermittent basis whenever the strainer was being brought on or off the line. This test consisted of simply measuring strainer inlet and outlet pressure, pump discharge pressure, and loop temperature for various flows from 1000 to 5000 gpm.

The long-term endurance test was run in order to insure strainer integrity over a reasonable time interval under high flow (> 4500 gpm) conditions. This test was run over a two-day period and the total accumulated time (at a flow in excess of 4500 gpm) was approximately 13 hours. Of this, approximately 10 hours was at or slightly above 4950 gpm.

The plugging test was conducted after foreign material (corrosion products, sand, and wood chips) was inadvertently introduced into the system during the long-term endurance test. (This foreign material was introduced during a water "bleed and feed" operation which was being done to hold down loop temperature.)

At the time the material was introduced, strainer inlet pressure was approximately 8.8 psig at 4950 gpm. Plugging was allowed to continue until the strainer inlet pressure reached 42 psig at a flow rate of 4530 gpm. The initial plugging rate was very steep, with the inlet pressure reaching 20 psig only about 10 minutes after introduction of the material. Plugging test duration (to 42 psig) was

about 3 hours, with about 2 hours and 50 minutes of this time at inlet pressures in excess of 20 psig.

Following the plugging test, the strainer cartridge was removed for inspection and cleaning purposes. Samples of the material were collected, and photographs of the cartridge were taken. No physical damage to any of the screens which make up the strainer cartridge was detected. Following this, the strainer cartridge was cleaned and reassembled for further endurance testing.

During the remaining portion of the long-term endurance run, no evidence of further plugging was detected. At the end of the endurance run, the strainer cartridge was removed a second time for inspection. Again, no damage of any kind was discernible.

9.4.3.3 Analysis of Results Obtained

Inspection of the screen after the plugging test showed the fouling to be evenly distributed over the entire screen area, indicating good flow distribution. The screen was cleaned with a compressed-air hose and with a water hose and was then reinstalled in the water loop. The observed pressure loss was back to the original value for the clean screen, and the test was completed with no further evidence of fouling.

Pressures were measured at the strainer inlet and at the strainer outlet. Figure 9-7 shows a plot of these data against flow rate. There are some discrepancies in the data, which were caused by the piping configuration required to fit the strainer into the existing test loop. In particular, the end of the discharge pipe could not be extended far enough below the pool surface to maintain the discharge pipe full of water at all times.

FIGURE 9-7. WATER FLOW TEST RESULTS FOR STRAINER NO. 3

Thus, the downstream pressure measurements are uncertain. The upstream pressure measurement is valid, however, and indicates that the overall pressure losses are within the design objective of 5 psi at 5000 gpm. A pressure loss of up to 10 psi at 5000 gpm would permit completion of the sodium flushing test at SEFOR.

The best estimate of the measured pressure loss is 4.6 psi at 5000 gpm. Pressure loss at lower flow rates may be less than predicted, because of reduced loss coefficients at lower velocities.

9.4.4 Tentative Results from Sodium Operation

Following installation of strainer No. 3 in the main primary loop, sodium

circulation was established over the period February 22 through February 24 with apparently good results. Sodium flow was maintained 5000 gpm or greater for 23 hours, including three hours at 5500 gpm, with no evidence of excessive pressure drop or change in pressure drop. Based upon full flow ΔP data with the sodium temperature in the range of 550°F to 600°F, the strainer ΔP was determined to be 4 psi which compares very favorably with the predicted value of 3.5 psi (see Section 9.4.1).

A final evaluation of strainer No. 3 performance will be completed after removal of the unit from the coolant loop and subsequent examination of the screen assembly.

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