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SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM

**THIRTIETH QUARTERLY REPORT
AUGUST-OCTOBER 1971**

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

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SOUTHWEST EXPERIMENTAL FAST OXIDE
REACTOR DEVELOPMENT PROGRAM

Thirtieth Quarterly Report
August-October 1971

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ABSTRACT

This report summarizes the work performed on the Research and Development Program for the Southwest Experimental Fast Oxide Reactor during the period from August 1, 1971, through October 31, 1971. The work was performed under contract AT(04-3)-540 with the United States Atomic Energy Commission.

1. INTRODUCTION

Southwest Experimental Fast Oxide Reactor (SEFOR) is a 20 MW, sodium-cooled fast reactor with characteristics similar to the large, soft spectrum fast breeder reactors fueled with mixed PuO_2 - U-238 O_2 , for which studies indicate the potential for producing low cost power. SEFOR will be used to obtain physics and engineering data at fuel compositions, temperatures, and crystalline states characteristic of power reactor operating conditions. SEFOR is particularly designed for the systematic determination of the Doppler coefficient of reactivity at temperatures up to the vicinity of fuel melting.

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

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

Phase I—Preoperational research and development,

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

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

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

1. "Southwest Experimental Fast Oxide Reactor Development Program, First Quarterly Report, April-June 1964," GEAP-4594 (July 1964).
2. "Southwest Experimental Fast Oxide Reactor Development Program, Second Quarterly Report, July-September 1964," GEAP-4742 (October 1964).
3. "Southwest Experimental Fast Oxide Reactor Development Program, Third Quarterly Report, October 1964-January 1965," GEAP-4799 (February 1965).
4. "Southwest Experimental Fast Oxide Reactor Development Program, Fourth Quarterly Report, February-April 1965," GEAP-4864 (May 1965).
5. "Southwest Experimental Fast Oxide Reactor Development Program, Fifth Quarterly Report, May-July 1965," GEAP-4934 (August 1965).
6. "Southwest Experimental Fast Oxide Reactor Development Program, Sixth Quarterly Report, August-October 1965," GEAP-4994 (November 1965).
7. "Southwest Experimental Fast Oxide Reactor Development Program, Seventh Quarterly Report, November 1965-January 1966," GEAP-5108 (February 1966).
8. "Southwest Experimental Fast Oxide Reactor Development Program, Eighth Quarterly Report, February-April 1966," GEAP 5160 (May 1966).

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9. "Southwest Experimental Fast Oxide Reactor Development Program, Ninth Quarterly Report, May-July 1966," GEAP-5208 (August 1966).
10. "Southwest Experimental Fast Oxide Reactor Development Program, Tenth Quarterly Report, August-October 1966," GEAP-5301 (November 1967).
11. "Southwest Experimental Fast Oxide Reactor Development Program, Eleventh Quarterly Report, November 1966-January 1967," GEAP-5442 (February 1967).
12. "Southwest Experimental Fast Oxide Reactor Development Program Twelfth Quarterly Report, February-April 1967," GEAP-5498 (May 1967).
13. "Southwest Experimental Fast Oxide Reactor Development Program, Thirteenth Quarterly Report, May-July 1967," GEAP-5533 (August 1967).
14. "Southwest Experimental Fast Oxide Reactor Development Program, Fourteenth Quarterly Report, August-October 1967," GEAP-5561 (November 1967).
15. "Southwest Experimental Fast Oxide Reactor Development Program, Fifteenth Quarterly Report, November 1967-January 1968," GEAP-5595 (February 1968).
16. "Southwest Experimental Fast Oxide Reactor Development Program, Sixteenth Quarterly Report, February-April 1968," GEAP-5629 (May 1968).
17. "Southwest Experimental Fast Oxide Reactor Development Program, Seventeenth Quarterly Report, May-July 1968," GEAP-5673 (August 1968).
18. "Southwest Experimental Fast Oxide Reactor Development Program, Eighteenth Quarterly Report, August-October 1968," GEAP-5705 (November 1968).
19. "Southwest Experimental Fast Oxide Reactor Development Program, Nineteenth Quarterly Report, November 1968-January 1969," GEAP-5754.
20. "Southwest Experimental Fast Oxide Reactor Development Program, Twentieth Quarterly Report, February-April 1969," GEAP-10010 (May 1969).
21. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-First Quarterly Report, May-July 1969," GEAP-10010-21 (August 1969).
22. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Second Quarterly Report, August-October 1969," GEAP-10010-22 (November 1969).
23. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Third Quarterly Report, November 1969-January 1970," GEAP-10010-23 (February 1970).
24. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Fourth Quarterly Report, February-April 1970," GEAP-10010-24 (May 1970).
25. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Fifth Quarterly Report, May-July 1970," GEAP-10010-25 (August 1970).
26. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Sixth Quarterly Report, August-October 1970," GEAP-10010-26 (November 1970).

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27. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Seventh Quarterly Report, November 1970–January 1971," GEAP-10010-27 (February 1971).
28. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Eighth Quarterly Report, February–April 1971," GEAP-10010-28 (May 1971).
29. "Southwest Experimental Fast Oxide Reactor Development Program, Twenty-Ninth Quarterly Report, May–July 1971," GEAP-10010-29 (August 1971).

In addition, the following topical reports have been issued:

1. Novak, P. E., and Asamoto, R. R., "A Survey for a High Temperature Sensor for SEFOR," GEAP-4903 (July 1965).
2. Quinn, E. P., "Vibration of SEFOR Fuel Rods in Parallel Flow," GEAP-4966 (September 1965).
3. Johnson, M. L., "A Description of Fuel and Other Core Materials Procured for SEFOR Critical Experiments in ZPR-III," GEAP-5133 (December 1965).
4. Protsik, R., "Epithermal Capture and Fission Distribution in a Fuel Rod—The Computer Code REAX," GEAP-4973 (April 1966).
5. Novak, P. E., and Asamoto, R. R., "An Out-of-Pile Evaluation of W-Re Thermocouple Systems for Use to 4700°F in Mixed Oxide," GEAP-5166 (May 1966).
6. 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. Carig, C. N., Ryer, C. M., and Thompson, M. L., "Plutonium-Uranium Mixed Oxide Fuel Pellet Fabrication Development for the Southwest Experimental Fast Oxide Reactor," GEAP-5285 (October 1966).
10. Polomik, E. E., and Williams, R. F., "Coolant Surface Turbulence Study, SEFOR Reactor Vessel," GEAP-5099 (December 1966).
11. Wilkinson, L., "Use of Neutron Filters for Fast Reactor Fuel Irradiations in a Thermal Core," GEAP-5035 (April 1967).
12. Wilkinson, L., et al., "Devices for Cyclic and Transient Power Control in a Steady-State Neutron Flux Gradient," GEAP-5036 (May 1967).
13. Reynolds, A. B., and Stewart, S. L., "Analysis of the SEFOR Mockup Critical Experiments in ZPR-3," GEAP-5294 (March 1967).
14. Wilkinson, D., "Weighted Doppler Analysis Code—WEDOP," GEAP-5513 (October 1967).
15. Craig, E. R., Fulton, R. H., Damon, J. J., "Design and Testing of Solenoid and Instrument Lead Connector for Use in SEFOR Reactor Vessel," GEAP-5307 (March 1967).
16. Asamoto, R. R., Perry, K. J., and Zobroski, 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. Cockran, J. T., "SEFOR Fuel Rod Sodium Thermal Shock Tests," GEAP-5467 August 1968).
26. Craig, E. R., "SEFOR Instrumented Fuel Assembly Design and Development," GEAP-5615 (April 1968).
27. Steamer, A. G., Brynsvold, G. V., and Franckx, L. C., "Fuel Handling and Other Tests in the SEFOR Refueling Cell Mockup," GEAP-5701 (March 1969).
28. Johnson, M. L., Becker, R. A., Haar, J. M., "SEFOR Fuel Handling," GEAP-13565 (December 1969).
29. Noble, L. D., et al., "Results of SEFOR Zero Power Experiments," GEAP-13588 (March 1970).
30. Ikeuye, D. T., and Young, F. E., "Fabrication of Non-Fuel Core Components for SEFOR," GEAP-13595 (February 1970).
31. Meyer, R. A., Stewart, S. L., Craig, E. R., Reynolds, A. B., and Johnson, M. L., "Design and Analysis of SEFOR Core I," GEAP-13598 (April 1970).
32. McKeehan, E. R., "Design and Testing of the SEFOR Fast Reactivity Excursion Device (FRED)," GEAP-13649 (October 1970).
33. Noble, L. D., et al., "SEFOR Core I Test Results to 20 MW," GEAP-13702 (April 1971).

2. SUMMARY

2.1 COMPLETED TASKS

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

TASK	TITLE
------	-------

1.1	Specifications for the Reactor and Experimental Program
-----	---

1.2	Final Specifications of the Experimental Program
-----	--

2.0	Experimental Program in ZPR-III
-----	---------------------------------

3.0	Fuel Development
-----	------------------

4.0	Special Instrumentation
-----	-------------------------

5.0	Steady-State and Transient Characteristics of the Core Assembly
-----	---

7.0	Fuel Handling
-----	---------------

8.0	Core Design
-----	-------------

10.0	Core I Startup Testing
------	------------------------

11.0	Core I Transient Tests
------	------------------------

12.0	Core I Prompt Excursion Tests
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2.2 TASK 6 – REACTIVITY CONTROLS REQUIRED FOR EXPERIMENTATION

Fabrication of the Phase IV (125 cents) neutron absorber rod for the Core II super-prompt transient tests has been started.

2.3 TASK 9 – FUEL FABRICATION

Four Core II Instrumented Fuel Rods were partially completed using platinum/molybdenum thermocouples. A compatibility test of platinum/molybdenum and (Pu, U) O₂ fuel was completed successfully. The Seal Plate, system instrumentation and handling equipment were being completed on schedule.

2.4 TASK 12 – CORE I SUPER-PROMPT TRANSIENTS

Analyses of the SEFOR Core I super-prompt critical transients have shown good agreement between the Doppler effect determined from the transient tests, and the effects calculated with a Doppler $T dk/dT = -0.0081$. These results are also in excellent agreement with the Doppler effect (-0.0081) determined from the Core I power ascension and sub-prompt critical transient tests.

Fuel temperature measurements with the four newly installed Instrumented Fuel Assemblies (IFA's) have agreed well with predictions at steady state power levels up to 8 MW (temperatures up to $\sim 2100^{\circ}\text{F}$). Preliminary analyses of fuel temperature measurements during the super-prompt critical transients have indicated that there is agreement between calculated and measured time dependent response of the thermocouples.

2.5 TASK 13 – CORE II MEASUREMENTS

Measurements indicate the reactivity loss associated with the conversion from SEFOR Core I to Core II (replacement of 102 BeO tightener rods with stainless steel tightener rods) is $-2.52\% \pm 0.10\%$. This compares with the

expected loss, based on earlier worth measurements in Core I, of -2.7\$. When combined with earlier measurements in Core I, these results indicate that the minimum critical size for Core II would be 536 ± 5 standard fuel rods (for fuel having zero burnup and as-specified fuel) and 89 stainless steel tightener rods.

The reactivity worths of reflector segments have been measured in Core II. Compared to similar measurements in Core I, the reflectors have increased (because of the harder spectrum) approximately 8% in worth.

The reactivity worths of fuel, B_4C , UO_2 and guinea pig rods (1/3 more Pu than a standard rod) were measured at different locations in the initial Core II configuration. The measurements indicate that the reactivity worth of Pu is of the order of 14% larger in Core II than in Core I, while other material reactivity worths are unchanged. Preliminary calculations of fuel reactivity worth in Core II have indicated that the trend and order of magnitude of the change are reasonable. More detailed calculations for all Core II material worths have been initiated, and they will be compared with the experiments at a later date.

The delayed neutron parameters for SEFOR Core II have been calculated. These parameters are nearly the same as those calculated for Core I, and there is essentially no difference in the relation (the inhour equation) between period and reactivity for the two cores. The calculated value of the total effective delayed neutron fraction for Core II is 0.00310.

2.6 SEFOR SAFETY AUDIT AND LICENSING

Two requests for changes to the Technical Specifications were submitted to the DRL while approval of three changes was received. Three safety evaluations were completed, two safety audit trips were made to the site and a response to the Safety Review Committee's seventh report was issued.

2.7 SEFOR SUPPORT

A stress report was prepared concerning the analysis of the reactor vessel wall to flange transition region. Performance tests were conducted for the nitrogen cooling and air blast coolers.

2.8 SEFOR FACILITY

Four new instrumented fuel assemblies were installed in the reactor the first week of August. Super-prompt transient tests began on August 14, and were completed August 30. The reactor was then operated at 10 MW for 90 MW-days to irradiate a rod designed to measure the fission-to-capture ratio in Pu-239, U-238 and U-235. At the same time, special tests were performed to measure Air Blast Cooler performance and nitrogen cooling systems performance, as related to the Follow-On-Program, and Oak Ridge National Laboratory (ORNL) personnel performed experiments to measure reactor noise at power. Following the irradiation, a pump performance test was performed to determine follow-on capabilities and two guinea-pig fuel rods irradiated during Core I. The fission-to-capture rod was shipped to VNC for destructive examination and zero power noise measurements were performed by ORNL on Core I.

The change over from Core I to Core II was completed on October 1, and Core II zero power measurements began. Zero power measurements completed include reflector calibration, material worth measurements, noise measurements, and power distribution and fission ratio measurements. Zero power measurements remaining to be completed are flow coefficient and temperature coefficient.

3. TASK 6 – REACTIVITY CONTROLS REQUIRED FOR EXPERIMENTATION

3.1 TASK 6.1 – FAST REACTIVITY EXCURSION DEVICE (FRED)

3.1.1 Purpose

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

3.1.2 Past Accomplishments

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

The three different worth neutron absorber rods required for the FRED tests in the Core I experimental program were fabricated and then used to complete the Core I testing.

3.1.3 Current Results

FRED Rod Fabrication for Core II

The Core II experimental program requires neutron absorber rods with two different reactivity worths. The Phase III absorber rod from Core I will be adjusted and used for the sub-prompt Core II FRED tests. Another absorber rod is being fabricated for the Core II super-prompt FRED tests. It is designed to enable reactivity worth adjustment and remote assembly and disassembly from the extension shaft.

FRED Rod Analysis

Even though the FRED Rod cladding temperature is limited (by controlling reactor power levels) to less than 1600°F, an analysis has been completed which indicates the FRED Rods would be safe from failure even for a cladding temperature as high as 1800°F. By C-stainless steel material compatibility tests run at 1850°F for 100 hours resulted in a 0.015 inch depth of interaction in the stainless steel, and further demonstrated the capability of the FRED rod to operate at this higher temperature.

4. TASK 9 – FUEL FABRICATION

4.1 PURPOSE

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

4.2 PAST ACCOMPLISHMENTS

The complete core component and fuel rod order has been received at SEFOR in previous quarters. The stainless steel tightener rods for Core II were received. Six instrumented fuel assemblies were shipped to the SEFOR site. Fabrication was completed on 102 additional fuel rods. Design of SEFOR Core II instrumented fuel rods was completed. The design includes three rods located at three separate through-head ports. Each rod contains one fuel thermocouple.

4.3 CURRENT RESULTS

4.3.1 Project Planning for Core II Instrumented Fuel Rods

A CPM network was prepared and used to coordinate the various activities required to produce the IFRs for reactor site delivery on December 7, 1971. The work plan continues to be updated in an effort to provide the IFRs at the site as much time prior to the scheduled date as possible. Expediting effort was applied to the critical path, which includes the fuel thermocouple procurement, initial assembly fuel loading, and final assembly.

4.3.2 Fuel-Thermocouple Compatibility Test

The first compatibility test of Pt/Mo T/C materials and $(\text{Pu,U})\text{O}_2$ fuel material was completed (70 hours at 2300°F in helium). The test specimens were clean and ductile after the test. Initial visual and mechanical bend tests of the sheath and Pt/Mo wires indicated good material compatibility with the SEFOR fuel which leads to an optimistic appraisal that this material is fully acceptable for use in the 2300°F (Maximum Core II transient test temperatures) $(\text{Pu,U})\text{O}_2$ and helium environment of the SEFOR instrumented fuel rod.

The second compatibility test of W/Re T/C materials and $(\text{Pu,U})\text{O}_2$ fuel material was also completed (70 hours at 3200°F in helium). It was observed that the welded junction that was located on the stranded W/Re thermoelements could not be located upon disassembly. On later T/C fabrications care will be taken to prevent degradation of these weld joints.

The W/Re wires and sheath were observed to be brittle after the test in the area of the 3200°F temperature and is expected to be less ductile after heating to 3000°F . The upper end of the T/C materials was at 1800°F during the test. Recrystallization occurs around 3000°F depending on tungsten alloy content. Depending on junction solution, the W/Re combination could be used as the IFR fuel T/C up to 3000°F and probably above. A rerun of the W/Re compatibility test at 4000°F has been planned.

4.3.3 Parts Fabrication and Assembly

Initial IFR assembly operations has begun. This work consists of building up of the assembly with vendor-supplied thermocouple and incorporates all work prior to fuel loading. Four initial assemblies have already been completed using platinum molybdenum thermocouples.

A work guide and travel card have been prepared for the planned fuel loading activity. Welder qualifications were begun and fuel pellet selection was made. Outgassing and O/M adjustment reduced from 1.99 to 1.97 operations on the fuel pellets was started.

Drawings were updated to incorporate the non-fueled mini-rod in place of the fueled mini-rod. This change was possible after measurements at SEFOR showed that the fuel was not needed for reactivity compensation. A mini-stainless steel rod (no fuel) will be used in the IFR Assembly. During fabrication of the fuel thermocouples, a BRD Quality Control engineer and a Design engineer were assigned to oversee progress of the fabrication.

The complete order of 8 platinum/molybdenum thermocouples was delivered, and inspection has been completed on 4 units.

4.3.4 Seal Plate

The Seal Plate is an ASME coded adapter piece between the vessel nozzle and the nozzle cover. It provides an exit through the reactor vessel for the fuel thermocouple leads.

Following development testing of a gland seal made from code-acceptable material, the fabrication was 80% completed on the IFR Seal Plates.

4.3.5 System Instrumentation

Design was completed and equipment purchase orders were initiated for the required leadout system and Constant Temperature Junction Box for the IFRs.

4.3.6 Handling and Installation

The Installation Specification was issued for review.

The IFR handling container drawings for SEFOR that were done in detail design. This container will be used to transport IFRs into the refueling cell and will then be used in a cell dry well for post irradiation storage.

A stronger container was designed and built for transporting partially completed IFRs between fabrication shops.

5. TASK 12 – CORE I SUPER-PROMPT TRANSIENTS

5.1 TEST DESCRIPTION

There are eight super-prompt critical transients in the Core I experimental program. The transients are initiated by rapid insertion of a positive reactivity (produced by ejecting a B_4C slug from a central channel in a time interval of ~ 0.1 sec) in excess of $\$1.0$. The tests are terminated by introduction of large negative reactivities (reactor scram) within ~ 0.3 sec of transient initiation. These transients result in peak power levels that are more than 3 orders of magnitude larger than initial values, and the prompt behavior of the inherent Doppler feedback reactivity is demonstrated by the prompt turn-around of the transient power.

5.2 DATA ANALYSIS

The data from one of the detector channels is illustrated for each of the super-prompt critical transients in Figures 5-1 through 5-8, and the time dependent excess energy release for transient No. 8 is illustrated in Figure 5-9. The small secondary peak in the power for the 1.28\$ transients as illustrated in Figures 5-10 and 5-11 was expected, and occurs because the positive reactivity used to initiate the transients is still being inserted for a short time following the power turn-around. Initial and peak power levels, as well as the energy release at 0.3 sec (the nominal scram time) that were measured during the transients are compared with predicted values in Table 5-1. The power changes observed during super-prompt critical transients No. 2 and No. 8 are compared with the predicted curves in Figures 5-10 and 5-11. The predictions in Figures 5-10 and 5-11, and also the predicted values in Table 5-1 were obtained using a Doppler $T dk/dT = -0.0081$, and non-Doppler coefficients which contribute $\sim 12\%$ of the total feedback, with the FORE (5-1) computer program. The good agreement between measured and calculated values illustrated by Figures 5-10 and 5-11 and in Table 5-1 is indicative of the agreement discussed below between the expected and measured Doppler coefficient of -0.0081 .

The special detectors used during the super-prompt critical transients were located near the core mid-plane at the radial locations illustrated in Figure 5-12. During the first two super-prompt transients, one data channel used a U-238 detector over the lower flux levels and switched to a gamma chamber for the higher flux levels to avoid any deleterious effects due to U-238 detector/amplifier saturation. Comparison of this data with that from another channel, where a U-238 detector was used throughout the transient, demonstrated that saturation did not occur. Therefore, U-238 chambers only were used to cover the transient flux in tests 3 through 8. Two SEFOR Energy Probes (SEP's), were located, one above the other, at the location indicated in Figure 5-12. These calorimetric devices measure the energy during the transient rather than the power itself. Since the transient energy can also be obtained from the time integral of the power (flux), the energy computed from the flux monitors is compared with that from the SEP's in Table 5-2. The agreement is generally good. However it appears that the normalization factor (obtained by comparison to the flux integral during the sub-prompt transients) for SEP No. 2 may, on the average, be approximately 10% too large.

During the super-prompt transients a logarithmic amplifier is used to convert the detector current into a voltage signal that is proportional to the logarithm of the current. This data is recorded and reconverted, by exponentiation, using a measured current-voltage characteristic into a signal proportional to power. The procedures for smoothing this flux data, calculating reactivity, energy, etc. are described in detail in reference 5-2. As expressed there, the net reactivity, $K_n(t)$, is determined from the recorded power trace, $P(t)$, using the following expression

$$K_n(t) = 1 + \left(\frac{\ell}{\beta}\right) \frac{1}{P(t)} \frac{dP(t)}{dt} + \frac{P(t) - P(o)}{P(t)} \frac{1}{P(t)} \int_0^t [P(u) - P(o)] D(t-u) du$$

where ℓ/β is the ratio of the prompt neutron lifetime to the effective delayed neutron fraction, and where $D(t)$ is the delayed neutron kernel (defined in reference 5-2). Since ℓ/β is of the order of 2×10^{-4} sec, and $P(t)$ is much larger than $P(o)$ during all but the initial portions of the super-prompt critical transients (see Figures 5-1 to 5-8), the term containing ℓ/β contributes to the reactivity only when the rate of change of the power, dP/dt , is very large; specifically for a few millisec just prior to, and just after, the peak power point.

The feedback reactivity, K_{fb} , is the difference between the net reactivity, K_n , and the input reactivity, K_F , generated by the Fast Reactivity Excursion Device (FRED), i.e.,

Figure 5-1. Time Dependent Power in Super – Prompt Transient No. 1

Figure 5-2. Time Dependent Power for Super – Prompt Transient No. 2

Figure 5-3. Time Dependent Power for Super – Prompt Transient No. 3

Figure 5-4. Time Dependent Power for Super – Prompt Transient No. 4

Figure 5-5. Time Dependent Power for Super – Prompt Transient No. 5

Figure 5-6. Time Dependent Power for Super – Prompt Transient No. 6.

Figure 5-7. Time Dependent Power for Super – Prompt Transient No. 7

Figure 5-8. Time Dependent Power for Super – Prompt Transient No. 8

Figure 5-9. Time Dependent Energy Release for Super — Promit Transient No. 8

Figure 5-10. Comparison of Calculated and Measured Power for Super — Prompt Transient 2

Figure 5-11 Comparison of Measured and Calculated Power for Super-Prompt Transient No. 8

Table 5-1
MEASURED AND PREDICTED POWER AND ENERGY

Test	Initial Power (MW)	Slug Worth (\$)	Peak Power Level		Energy Release to 0.3 sec	
			Measured ^(a) (MW)	Predicted (MW)	Measured (MW-sec)	Predicted (MW-sec)
1	2	1.18	5300	5400	82	80
2	2	1.28	9000	8800	108	105
3	2	1.28	8800-9300	8800	106	105
4	5	1.18	4900-5100	5300	93	95
5	5	1.28	8200-8700	8600	120	125
6	8	1.18	5100-5500	5300	108	110
7	8	1.28	8300-8700	8500	135	140
8	8	1.28	8400-8800	8500	136	140

(a) Preliminary results from Channel A (range indicates variation using different amplifier calibrations).

Table 5-2
COMPARISON OF MEASURED ENERGY RELEASE
FOR SUPERPROMPT TRANSIENTS

Test	ENERGY RELEASE ^(a) (MW-sec)			
	SEFOR Energy Probe Number 1	SEFOR Energy Probe Number 2	Fission Detector A	Fission Detector B
1	43	47	46	43
2	62	65	62	64
3	58	65	60	59
4	50	56	49	48
5	69	77	66	64
6	62	64	56	56
7	75	84	73	70
8	73	78	74	70

(a) At 120 msec after lift-off.

Figure 5-12. Transient Detector Locations

$$K_{fb}(t) = K_n(t) - K_F(t)$$

In the time interval around the power peak the input reactivity is time dependent, and the apparent net reactivity determined from the power trace is dependent on the value of λ/β . Thus, small percentage uncertainties in K_F or uncertainties in λ/β , and thus in K_n , may cause relatively large uncertainties in the feedback reactivity $K_{fb}(t)$. The sensitivity of results to variations in λ/β and in amplifier voltage-current characteristics are currently being investigated.

To avoid the above uncertainties, the feedback and energy data were (as in reference 5-2) analyzed during the time interval following the complete ejection of the FRED from reactor regions of reactivity influence, but prior to reactor scram or re-entry of the poison slug into the core. During this time interval (between ~ 140 msec and ~ 270 msec after transient initiation), the input reactivity K_F is a constant and any changes in the net reactivity are a result of changes in the feedback. In this interval, since $K_F = K_0$, a constant, the net reactivity can be expressed as

$$K_n(t) = K_0 + \sum_{j=1}^J a_j E^j(t)$$

A least square fit of the above expression to the data during the stated time interval indicated that for the limited energy range under consideration an essentially linear relationship exists between energy release and reactivity. The slopes of these linear relations, or the energy coefficients, as well as the approximate energy intervals over which the analyses were performed, are shown in Table 5-3. Note that the coefficients tend to decrease slightly in magnitude with increasing energy release from the same initial power level, and they also decrease in magnitude with increasing initial power level. These trends are, as illustrated in more detail below, a result of the $1/T$ dependence of the Doppler coefficient.

Table 5-3
SUPERPROMPT TRANSIENT ENERGY COEFFICIENTS

Test	Initial Power (MW)	Slug Worth (%)	Energy Range (MW-sec)	Measured Energy Coefficient (¢/MW-sec)
1	2	1.18	50-75	-0.47
2	2	1.28	70-110	-0.45
3	2	1.28	70-110	-0.45
4	5	1.18	60-90	-0.42
5	5	1.28	75-115	-0.42
6	8	1.18	70-100	-0.39
7	8	1.28	90-125	-0.39
8	8	1.28	90-125	-0.38

The coefficients in Table 5-3 were corrected for a calculated⁽⁵⁻²⁾ $\sim 12\%$ non-Doppler (i.e., the fuel and cladding axial expansion, sodium and structure effects, etc.) reactivity effect, and they were adjusted by a factor⁽⁵⁻³⁾ of 0.88 to account for the difference between the energy produced during a transient, and the energy absorbed in the fuel during a transient. The resultant averaged Doppler energy coefficients, in units of cents per MW-sec of energy absorbed in the fuel, are compared in Table 5-4 with the results obtained⁽⁵⁻²⁾ from the earlier sub-prompt transients, and with calculated values appropriate to the initial power levels and energy ranges of analysis.

Table 5-4
TRANSIENT TEST RESULTS

Initial Power (MW)	Nominal Slug Worth (\$)	Energy Range of Analysis (MW-sec)	Doppler Energy Coefficient	
			Measured (¢/MW-sec)	Calculated ^(a) (¢/MW-sec)
2	0.97	<15	-0.56	-0.54
2	1.18	50-75	-0.47	-0.49
2	1.28	70-110	-0.45	-0.46
5	0.97	<25	-0.45	-0.45
5	1.18	60-90	-0.42	-0.42
5	1.28	75-115	-0.42	-0.40
8	1.18	70-100	-0.39	-0.37
8	1.28	90-125	-0.38	-0.36
10	0.97	<40	-0.35	-0.37

(a) Calculated with $T \frac{dk}{dT} = -0.0081$

The calculations were performed with the WEDOP⁽⁵⁻⁴⁾ program using a value of -0.0081 for the Doppler $T dk/dT$, fuel thermal conductivity and fuel to cladding heat transfer coefficients discussed in reference 5-5, and heat capacities from reference 5-6. The computer program⁽⁵⁻⁴⁾ calculates an initial steady state temperature distribution for fuel rods throughout the core, and performs a power squared weighting of the logarithm of the local temperatures to obtain the Doppler feedback. For transients, new temperature distributions, and reactivity changes, are calculated under the assumption that the transient is sufficiently rapid to prevent any appreciable heat diffusion from the rod during the time interval (~ 0.3 sec) under consideration. For the SEFOR one-inch fuel rods (having a time constant of ~ 30 sec), this assumption is valid.

It is seen in Table 5-4, that both the measured and calculated Doppler coefficient decrease with increasing energy release at a given power level and with increasing initial power level. The good agreement between the magnitudes and trends of the calculated and measured transient values substantiates the previously measured⁽⁵⁻⁵⁾ results obtained during the power ascension, and illustrates that the SEFOR Doppler feedback is well represented by a constant value of -0.0081 for the product $T dk/dT$.

5.3 IN-CORE INSTRUMENTATION

In-core instrumentation for SEFOR presently consists of six installed Instrumented Fuel Assemblies (IFA's) at three radial core locations. Each IFA has four fuel thermocouples, one inlet and two outlet coolant thermocouples and a magnetic coolant flowmeter. The stable reactor thermal data prior to the transient tests are given in the following tables. Table 5-5 summarizes the reactor heat balance data for each test. The IFA data is presented in Table 5-6 and 5-7 that show the fuel temperature rise above the coolant and the channel coolant temperature and flow. The in-core instrumentation that was installed prior to transient testing (IFA 2,3,4,6) continues to have the good operating status as shown in Tables 5-6 and 5-4.

Table 5-5
HEAT BALANCE DATA

Superprompt Transient Tests	Date (1971)	Thermal Power (MW)	Average Coolant Temperature	Main Primary System Temperatures			Auxiliary Primary System Temperatures		
				Inlet (°F)	Outlet (°F)	Flow (gpm)	Inlet (°F)	Outlet (°F)	Flow (gpm)
1	Aug 14	1.99	760.0	754.7	765.2	4858	761.6	765.5	227
1B ^(a)	Aug 14	1.22	756.0	752.7	759.2	4908	760.1	761.0	225
1C ^(a)	Aug 14	1.19	758.9	756.2	761.6	4818	746.5	760.6	236
2	Aug 17	1.96	762.6	757.5	767.6	4800	753.9	765.9	230
3	Aug 19	2.12	763.7	758.3	769.0	4869	754.3	768.1	236
4	Aug 21	5.24	764.7	751.3	778.0	4916	751.5	776.5	239
5	Aug 23	5.17	763.9	750.3	777.4	4813	755.7	776.2	230
6	Aug 26	8.05	761.3	740.4	782.1	4864	745.6	777.7	238
7	Aug 28	8.08	765.0	744.0	786.0	4818	744.5	781.9	234
8	Aug 30	7.99	764.1	743.2	784.9	4097	744.3	780.2	266

(a) Subprompt

The stable fuel temperature data obtained during the testing program compared closely with expected results. The fuel thermocouple description and local power density are given in Tables 5-8 and 5-9. With representative data from Table 5-6, the measured fuel temperature rise is compared with the calculation in Figure 5-13 showing the good correspondence between calculation and measurement. The basis of this comparison is the local power density modified by the influence factor for the presence of the thermocouple hole.

The transient response for the fuel thermocouples is given in Table 5-10. The IFA-1 and IFA-5 data have been omitted as these thermocouples have shown decalibration and respond to temperature erratically throughout the transient tests. Figure 5-14 shows the time response of a fuel thermocouple to a super-prompt transient observed during the tests. Measured and calculated transient fuel temperature rises are currently being compared, and preliminary results indicate good correspondence similar to the static test results.

Table 5-6
FUEL TEMPERATURE RISE ABOVE COOLANT TEMPERATURE (~760°F)

Test Number	Date (1971)	Thermal Power (MW)	IFA-1				IFA-2			
			Inner Rod		Outer Rod		Inner Rod		Outer Rod	
			Upper Number 1 (°F)	Lower Number 2 (°F)	Upper Number 3 (°F)	Lower Number 4 (°F)	Upper Number 5 (°F)	Lower Number 6 (°F)	Upper Number 7 (°F)	Lower Number 8 (°F)
1	Aug 14	1.99	212	395	R	R	211	339	177	660
1B ^(a)	Aug 14	1.22	123	231	106	264	125	198	101	785
1C ^(a)	Aug 14	1.19	107	217	R	R	120	180	102	68
2	Aug 17	1.96	193	372	R	399	191	312	164	289
3	Aug 19	2.12	205	386	R	R	192	314	170	405
4	Aug 21	5.24	484	881	R	978	513	854	446	716
5	Aug 23	5.17	489	885	R	984	501	910	447	367
6	Aug 26	8.05	NR	1283	184	NR	804	1371	754	1350
7	Aug 28	8.08	NR	1279	845	NR	833	1362	754	1212
8	Aug 30	7.99	NR	1279	592	1410	819	1334	746	1335

(a) Subprompt

NR = No emf response

R = Emf response less than coolant temperature

Table 5-6 (Continued)
FUEL TEMPERATURE RISE ABOVE COOLANT TEMPERATURE (~760°F)

Test Number	Date (1971)	Thermal Power (MW)	IFA-3				IFA-4			
			Inner Rod		Outer Rod		Inner Rod		Outer	
			Upper Number 9 (°F)	Lower Number 10 (°F)	Upper Number 11 (°F)	Lower Number 12 (°F)	Upper Number 13 (°F)	Lower Number 14 (°F)	Upper Number 15 (°F)	Lower Number 16 (°F)
1	Aug 14	1.99	154	286	170	275	189	301	169	288
1B ^(a)	Aug 14	1.22	86	157	96	154	109	169	94	163
1C ^(a)	Aug 14	1.19	80	149	93	148	101	159	86	151
2	Aug 17	1.96	139	260	156	263	168	278	154	264
3	Aug 19	2.12	143	269	157	267	183	287	156	277
4	Aug 21	5.24	375	729	413	683	468	756	417	719
5	Aug 23	5.17	378	735	416	695	468	723	414	713
6	Aug 26	8.05	591	160	636	1057	754	1157	668	1090
7	Aug 28	8.08	584	78	623	1026	742	1001	645	1052
8	Aug 30	7.99	581	79	620	1003	706	908	647	1024

(a) Subprompt

Table 5-€ (Continued)
FUEL TEMPERATURE RISE ABOVE COOLANT TEMPERATURE (~760°F)

Test Number	Date (1971)	Thermal Power (MW)	IFA-5				IFA-6			
			Inner Rod		Outer Rod		Inner Rod		Outer Rod	
			Upper Number 17 (°F)	Lower Number 18 (°F)	Upper Number 19 (°F)	Lower Number 20 (°F)	Upper Number 21 (°F)	Lower Number 22 (°F)	Upper Number 23 (°F)	Lower Number 24 (°F)
1	Aug 14	1.99	46	R	NR	380	112	166	86	NR
1B ^(a)	Aug 14	1.22	28	R	NR	235	66	99	51	NR
1C ^(a)	Aug 14	1.19	29	R	NR	218	66	97	55	NR
2	Aug 17	1.96	35	R	NR	353	100	153	74	NR
3	Aug 19	2.12	41	R	NR	369	98	158	80	NR
4	Aug 21	5.24	88	R	NR	882	256	410	214	NR
5	Aug 23	5.17	88	11	NR	890	254	416	196	NR
6	Aug 26	8.05	144	285	NR	1365	444	742	403	NR
7	Aug 28	8.08	151	269	NR	1347	446	738	422	NR
8	Aug 30	7.99	131	296	NR	1259	441	736	412	NR

(a) Subprompt

NR = No emf response

R = Emf response less than coolant temperature

Table 5-7
IFA COOLANT TEMPERATURES AND FLOW MEASUREMENTS

Test Number	Date (1971)	Thermal Power (MW)	IFA-1			IFA-2		
			Inlet Number 32 (°F)	Outlet Number 31 (°F)	Flowmeter Number 51 (gpm)	Inlet Number 34 (°F)	Outlet Number 33 (°F)	Flowmeter Number 52 (gpm)
1	Aug 14	1.99	777.8	769.0	NR	759.9	769.5	37.92
1B ^(a)	Aug 14	1.22	780.5	763.0	NR	757.0	762.7	37.24
1C ^(a)	Aug 14	1.19	773.0	765.8	NR	766.0	771.4	41.29
2	Aug 17	1.96	763.2	766.8	NR	757.8	767.2	41.33
3	Aug 19	2.12	773.2	766.3	NR	757.2	766.8	42.00
4	Aug 21	5.24	811.4	566.4	NR	754.4	778.2	42.54
5	Aug 23	5.17	785.9	774.9	NR	751.7	776.6	41.68
6	Aug 26	8.05	774.3	782.9	NR	745.6	783.6	42.35
7	Aug 28	8.08	788.2	783.3	NR	747.2	785.3	41.48
8	Aug 30	7.99	779.6	782.9	NR	745.7	784.2	41.57

(a) Subprompt series
 NR = No emf response (≈ 0.00)

Table 5-7 (Continued)
IFA COOLANT TEMPERATURES AND FLOW MEASUREMENTS

Test Number	Date (1971)	Thermal Power (MW)	IFA-3			IFA-4		
			Inlet Number 36 (°F)	Outlet Number 35 (°F)	Flowmeter Number 53 (gpm)	Inlet Number 38 (°F)	Outlet Number 37 (°F)	Flowmeter Number 54 (gpm)
1	Aug 14	1.99	771.1	780.8	49.78	143.6	771.0	49.57
1B ^(a)	Aug 14	1.22	757.0	763.7	49.77	775.0	763.3	49.76
1C ^(a)	Aug 14	1.19	762.4	768.3	49.80	139.0	767.2	49.74
2	Aug 17	1.96	758.1	768.7	49.80	158.9	769.0	49.52
3	Aug 19	2.12	757.1	767.8	49.82	124.1	768.6	49.39
4	Aug 21	5.24	754.3	781.4	49.76	133.9	782.3	49.45
5	Aug 23	5.17	751.8	780.1	49.76	152.3	780.7	49.34
6	Aug 26	8.05	745.2	787.9	49.66	NR	790.7	49.24
7	Aug 28	8.08	747.3	790.5	49.69	NR	791.6	49.44
8	Aug 30	7.99	746.6	785.5	49.68	NR	790.9	49.07

(a) Subprompt series
NR = No emf response (≅123.5)

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Table 5-7 (Continued)
IFA COOLANT TEMPERATURES AND FLOW MEASUREMENTS

Test Number	Date (1971)	Thermal Power (MW)	IFA-5			IFA-6		
			Inlet Number 40 (°F)	Outlet Number 39 (°F)	Flowmeter Number 55 (gpm)	Inlet Number 42 (°F)	Outlet Number 41 (°F)	Flowmeter Number 56 (gpm)
1	Aug 14	1.99	768.3	768.4	NR	758.1	769.1	30.31
1B ^(a)	Aug 14	1.22	761.8	761.9	NR	154.6 ^(b)	760.3	30.30
1C ^(a)	Aug 14	1.19	767.5	767.3	NR	738.6	771.3	30.31
2	Aug 17	1.96	768.2	768.3	NR	776.9	767.0	30.31
3	Aug 19	2.12	768.5	768.6	NR	127.0 ^(b)	766.6	30.33
4	Aug 21	5.24	778.3	779.0	NR	773.9	778.1	30.29
5	Aug 23	5.17	777.4	776.6	NR	751.9	775.8	30.22
6	Aug 26	8.05	785.4	783.8	NR	746.2	786.2	30.23
7	Aug 28	8.08	787.0	785.1	NR	764.9	787.2	30.25
8	Aug 30	7.99	785.9	784.7	NR	747.5	787.3	30.24

(a) Subprompt series

(b) No emf response (≈ 123.5)

NR = No emf response (≈ 0.00)

Table 5-8
IFA DESCRIPTION

		DAS Channel Number	Thermocouple Description (GEAP-5615)			
IFA-1 (E0.5-S1.0) Center Radial Power — Local P/A = 1.38 (VNC-29)						
Inner Rod	Upper Thermocouple	1	0.064 in. o.d.	BeO Insulation	0.005 in. wire	
	Lower Thermocouple	2	0.064 in. o.d.	BeO	0.005 in. wire	
Outer Rod	Upper Thermocouple	3	0.064 in. o.d.	BeO	0.005 in. wire	
	Lower Thermocouple	4	0.064 in. o.d.	ThO ₂	0.005 in. wire	
IFA-2 (W0.5-N1.0) Center Radial Power — Local P/A = 1.38 (VNC-27)						
Inner Rod	Upper Thermocouple	5	0.140 in. o.d.	BeO Insulation	0.010 in. wire	
	Lower Thermocouple	6	0.140 in. o.d.	BeO	0.010 in. wire	
Outer Rod	Upper Thermocouple	7	Bare Wire	ThO ₂	0.010 in. wire	
	Lower Thermocouple	8	0.140 in. o.d.	BeO	0.010 in. wire	
IFA-3 (E3.0-S0.0) Midradius Radial Power — Local P/A = 1.13 (VNC-26)						
Inner Rod	Upper Thermocouple	9	0.140 in. o.d.	BeO Insulation	0.010 in. wire	
	Lower Thermocouple	10	Bare Wire	ThO ₂	0.010 in. wire	
Outer Rod	Upper Thermocouple	11	Bare Wire	ThO ₂	0.010 in. wire	
	Lower Thermocouple	12	0.140 in. o.d.	BeO	0.010 in. wire	
IFA-4 (W1.5-S3.0) Midradius Radial Power — Local P/A = 1.13 (VNC-28)						
Inner Rod	Upper Thermocouple	13	0.064 in. o.d.	BeO Insulation	0.005 in. wire	
	Lower Thermocouple	14	0.064 in. o.d.	BeO	0.005 in. wire	
Outer Rod	Upper Thermocouple	15	0.064 in. o.d.	BeO	0.005 in. wire	
	Lower Thermocouple	16	0.064 in. o.d.	BeO	0.005 in. wire	
IFA-5 (E2.5-N5.0) Peripheral Radial Power — Local P/A = 0.80 (VNC-25)						
Inner Rod	Upper Thermocouple	17	0.064 in. o.d.	BeO Insulation	0.005 in. wire	
	Lower Thermocouple	18	0.064 in. o.d.	BeO	0.005 in. wire	
Outer Rod	Upper Thermocouple	19 Dummy	0.140 in. o.d.	BeO	0.010 in. wire	
	Lower Thermocouple	20	0.064 in. o.d.	BeO	0.005 in. wire	
IFA-6 (W5.0-S0.0) Peripheral Radial Power — Local P/A = 0.80 (VNC-24)						
Inner Rod	Upper Thermocouple	21	0.140 in. o.d.	BeO Insulation	0.010 in. wire	
	Lower Thermocouple	22	Bare Wire	ThO ₂	0.010 in. wire	
Outer Rod	Upper Thermocouple	23	0.140 in. o.d.	BeO	0.010 in. wire	
	Lower Thermocouple	24 Dummy	0.140 in. o.d.	BeO	0.010 in. wire	

Table 5-9
LOCAL POWER DENSITY FOR
IFA FUEL THERMOCOUPLE LOCATIONS

T/C Description	DAS	Local Power Density (W/cc/MW)	Geometric Correction for T/C Hole ^(a)
IFA-1	1	3.62	0.963
IFA-1	2	6.04	0.963
IFA-1	3	3.62	0.963
IFA-1	4	6.16	0.963
IFA-1	5	4.14	0.878
IFA-2	6	6.92	0.878
IFA-2	7	4.07	0.878
IFA-2	8	6.79	0.878
IFA-3	9	3.53	0.878
IFA-3	10	5.90	0.878
IFA-3	11	3.26	0.878
IFA-3	12	5.44	0.878
IFA-4	13	3.53	0.963
IFA-4	14	5.90	0.963
IFA-4	15	3.26	0.963
IFA-4	16	5.44	0.963
IFA-5	17	2.47	0.963
IFA-5	18	4.35	0.963
IFA-5	19	2.17	Dummy
IFA-5	20	3.80	0.963
IFA-6	21	2.54	0.878
IFA-6	22	4.25	0.878
IFA-6	23	2.22	0.878
IFA-6	24	3.71	Dummy

(a) The fuel thermocouple temperature rise above the coolant is related to the local power density by

$$\int_{\text{Surface T}}^{\text{center T}} k dT = \frac{QR^2}{4} f_{T/C} ;$$

Q — local power density
R — fuel rod radius
 $f_{T/C}$ — geometric correction
for T/C hole

Table 5-10

**FUEL THERMOCOUPLE RESPONSE
SUPERPROMPT TRANSIENT FUEL TEMPERATURE**

		Superprompt Tests							
		Number 1		Number 2		Number 3		Number 4	
T/C Description	DAS	Initial (°F)	Maximum (°F)	Initial (°F)	Maximum (°F)	Initial (°F)	Maximum (°F)	Initial (°F)	Maximum (°F)
IFA-2	5	986	1163	988	1251	988	1260	1286	1502
IFA-2	6	1094	1398	1183	1530	1094	1537	1568	1921
IFA-2	7	959	1157	965	1256	975	1272	1235	1499
IFA-2	8	1879	E	1281	E	1174	E	1247	E
IFA-3	9	939	1086	941	1150	947	1157	1164	1340
IFA-3	10	1064	1346	1051	1457	1060	1480	1488	E
IFA-3	11	972	1101	955	1166	963	1183	1194	1397
IFA-3	12	1060	1286	1047	1384	1060	1400	1454	1718
IFA-4	13	963	1135	963	1212	972	1227	1247	1466
IFA-4	14	1061	1352	1060	1482	1071	1499	1494	1839
IFA-4	15	948	1112	947	1174	953	1183	1201	1399
IFA-4	16	1054	1312	1049	1412	1060	1440	1476	1798
IFA-6	21	903	1002	911	1051	923	1067	1077	1198
IFA-6	22	966	1124	971	1205	987	1217	1247	1448
IFA-6	23	886	981	894	1032	905	1045	1047	1170
IFA-6	24	Dummy Thermocouple — No emf response during transient testing.							

E = Erratic transient response

Table 5-10 (Continued)

FUEL THERMOCOUPLE RESPONSE
SUPERPROMPT TRANSIENT FUEL TEMPERATURE

		Superprompt Tests							
T/C Description	DAS	Number 5		Number 6		Number 7		Number 8	
		Initial (°F)	Maximum (°F)	Initial (°F)	Maximum (°F)	Initial (°F)	Maximum (°F)	Initial (°F)	Maximum (°F)
IFA-2	5	1277	1558	1594	1831	1614	1921	1583	1879
IFA-2	6	1522	1990	2044	2406	2044	2517	2003	2456
IFA-2	7	1235	1578	1512	1800	1534	1911	1516	1887
IFA-2	8	944	E	1816	E	1559	E	1404	E
IFA-3	9	1163	1390	1386	1578	1404	1655	1386	1633
IFA-3	10	1476	E	934	E	882	E	871	E
IFA-3	11	1191	1438	1430	1650	1440	1730	1425	1706
IFA-3	12	1447	1812	1837	2124	1839	2234	1816	2181
IFA-4	13	1243	1530	1508	1744	1511	1809	1464	1748
IFA-4	14	1448	1883	1770	2119	1699	2199	1613	2060
IFA-4	15	1205	1454	1438	1655	1440	1718	1425	1697
IFA-4	16	1448	1850	1830	2169	1809	2239	1762	2171
IFA-6	21	1077	1235	1237	1363	1251	1425	1242	1412
IFA-6	22	1243	1508	1511	1737	1522	1819	1502	1792
IFA-6	23	1030	1217	1205	1355	1232	1447	1216	1430
IFA-6	24	Dummy Thermocouple — No emf response during transient testing.							

E = Erratic transient response

Figure 5-13. Fuel Centerline Temperature Rise Above Coolant Temperature

Figure 5-14. Time Dependence of a Fuel Thermocouple During a Transient

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- 5-3. L. D. Noble, and C. D. Wilkinson, "Final Specifications for the SEFOR Experimental Program", GEAP-5576, January 1968.
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- 5-5. L. D. Noble et al., "SEFOR Core I Test Results to 20 MW", GEAP-13702, April 1971.
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6. TASK 13 – CORE II TEST PROGRAM

6.1 CONVERSION TO CORE II

6.1.1 Description of the Experiment

Core II of the Southwest Experimental Fast Oxide Reactor is obtained from Core I by replacing BeO tightener rods with stainless steel tightener rods. This exchange is made to produce a harder neutron spectrum, and a smaller Doppler coefficient for Core II.

The conversion from Core I to Core II was performed in steps, with the reactivity losses resulting from tightener rod replacement being compensated by replacement of B_4C rods with fuel rods. At the end of each step, and also after each B_4C to fuel exchange, the reactor was brought to critical by raising the reflector segments. Changes in reactivity between critical configurations were then determined from the observed changes in reflector positions.

The core loading just prior to initiation of the conversion from Core I to Core II is illustrated in Figure 6-1. This loading was similar to those used for the Core I static, oscillator, and transient measurements, but all rods (15 total, including 5 tightener rods) accessible under the through-head ports had been removed to allow removal of the vessel head. They remained out of the reactor during the core conversion because their extra long extension rods would have protruded well above the sodium level and hindered the tightener rod replacement.

The locations of the BeO to steel exchanges during the sequential steps in the core conversion are illustrated in Figure 6-2. Also shown in Figure 6-2 are the locations of the 3 B_4C rods which were replaced by fuel rods. The final configuration, which was the initial Core II loading (Assembly II-A), is shown in Figure 6-3. This core contains five vacant tightener rod locations which were later filled with stainless steel rods when the vessel head was replaced.

6.1.2 Experimental Data

The data obtained during the core conversion are tabulated in Table 6-1. Two values are listed for the reflector worths at the measured positions. One value was obtained using the reflector calibrations in Core I (specifically those in Assembly I-I at 750°F, as the reflector configurations for these calibrations were similar to the configuration in I-Q), and the other using the calibrations in Core II (Assembly II-A). Since the data were obtained during the transition from Core I to Core II, the effective reactivity change in a given conversion step (Table 6-2), or for a particular B_4C to fuel exchange (Table 6-3), was obtained from a weighted average of the effects determined from the two calibrations. The weights varied linearly with the number of exchanged tightener rods. The reactivity loss for replacement of the final 5 tightener rods was not measured directly, but its value, shown as step 6 in Table 6-2, was estimated using the average worth per rod obtained from the Core II reflector calibrations.

The estimated standard deviations shown in Tables 6-2 and 6-3 include the previously⁽⁶⁻¹⁾ estimated $\pm 4\%$ uncertainty resulting from use of reflector calibrations for reactivity measurements in configurations other than that used for the calibration, and they include an estimated uncertainty⁽⁶⁻¹⁾ of $\pm 1.5\%$ in an individual reactivity measurement. The combined uncertainty is calculated as the square root of the sum of the squares of the individual components.

6.1.3 Estimated Core II Minimum Critical Size

The minimum critical size for Core II was estimated by postulating a prescribed sequence of changes to the Core I minimum critical configuration and by using measured reactivity effects as follows:

Assume that the Core II minimum critical is obtained by adding to the Core I periphery just that number of fuel and BeO rods (at a 6 to 1 ratio) which will provide additional reactivity that exactly offsets the reactivity loss when all (including those added) BeO tightener rods are replaced by stainless steel rods. Then, based on measurements⁽⁶⁻¹⁾ in Core I of 12¢ per peripheral fuel rod and 4¢ per peripheral BeO rod, the added reactivity for X fuel rods, and X/6 BeO rods would be 12.67 X cents. Using the average of the measurements discussed above, replacement of all (518+X)/6 BeO rods with stainless steel would result in a reactivity loss in cents of

$$\left(\frac{518+X}{6} \right) \left(\frac{252.1}{102} \right)$$

Figure 6-1. Core I Loading Assembly I-Q

Figure 6-2. Core I to Core II Loading Changes

Figure 6-3. Core II Loading Assembly II-A

Table 6-1
SEQUENTIAL CRITICAL MEASUREMENTS

Sequential Reactor Configurations	Reflector ^(a) 3			Reflector ^(a) 8			Average Temperature (°F)
	Position (cm)	Worth ^(b)		Position (cm)	Worth ^(b)		
		Core I Calibration (¢)	Core II Calibration (¢)		Core I Calibration (¢)	Core II Calibration (¢)	
Ccre I – Thru Head Rods Removed	40.00	56.91	60.51	40.82	57.83	61.76	350
BeO to Steel – Step 1 (16 rods)	40.00	56.91	60.51	63.14	99.38	106.30	354
B ₂ C to fuel @ E1.5N1.0B	45.00	66.53	71.11	0.2	-0.6	0.04	360
BeO to Steel – Step 2 (13 rods)	40.00	56.91	60.51	31.82	40.66	43.51	354
B ₄ C to fuel @ W1.5N1.0E	8.52	6.55	7.28	0.2	-0.6	0.04	350
BeO to Steel – Step 3 (21 rods)	40.01	56.93	60.53	2.47	1.10	1.95	354
BeO to Steel – Step 4 (28 rods)	40.01	56.93	60.53	46.24	68.53	72.98	350
B ₄ C to fuel @ E0.0 S2.0D	26.56	31.80	33.36	0.2	-0.6	0.04	350
BeO to Steel – Step 5 (19 rods)	0.3	-0.5	0.0	47.12	70.20	74.76	350

(a) All other reflectors were completely raised.

(b) The calibration curves were obtained using calculated parameters in the inhour equations that convert asymptotic period data into reactivities.
(see reference 6-1 and Section 6.3 of this report)

Table 6-2
REACTIVITY CHANGES DURING BeO TO STAINLESS STEEL EXCHANGES

	Incremental ^(a) Reactivity Change (ϕ)	Cumulative Number of Rods	Cumulative ^(a) Reactivity Change (ϕ)
Step 1 16 rods	-39.4	16	-39.4 $\pm$ 2.2
Step 2 13 rods	-35.6	29	-75.0 $\pm$ 3.7
Step 3 21 rods	-51.0	50	-126.0 $\pm$ 5.7
Step 4 28 rods	-72.3	78	-198.3 $\pm$ 8.5
Step 5 19 rods	-41.1	97	-239.4 $\pm$ 10.1
Step 6 5 rods	-12.7 ^(b)	102	-252.1 ^(b) $\pm$ 10.2

(a) Reactivity corrections for temperature changes were made using a coefficient of $-0.6\phi/^{\circ}\text{F}$.

(b) Estimated using average worth per rod from Core II reflector calibrations

Table 6-3
REACTIVITY WORTHS OF B₄C TO FUEL ROD EXCHANGES

Location of Exchange	Reactivity Worth ^(a) (¢)
E1.5 N1.0B	94.9±4.3
W1.5 N1.0E	90.9±4.2
E0.0 S2.0D	99.1±4.5

(a) Reactivity corrections for temperature changes were made using a coefficient of $-0.6\phi/^{\circ}\text{F}$.

Equating the reactivity loss to the reactivity gain yields a value for X of 17.4. Adding 18 rods to the Core I equivalent critical loading⁽⁶⁻¹⁾ of 518 rods would give a minimum critical mass for Core II of 536 standard fuel rods and 89 stainless steel tightener rods.

In determining the Core II minimum critical size, it is assumed that the value for the average worth per rod for BeO to steel exchanges is the same for both the partially loaded (minimum critical) core and the fully loaded core. This assumption appears reasonable for several reasons.

The measured reactivity loss during the core transition was within 8% of the expected value. This indicates that the worth does not change significantly because of the Core I to Core II spectral changes. In addition, the worth of the BeO to steel exchange is relatively constant across the core (Figure 6-4). This implies that the reactivity loss during the exchange is dependent on the number of rods exchanged and not on the particular size of the core.

Material reactivity worth measurements performed on a fully loaded core (Assembly I-D) can be used to estimate the worth of removing a BeO rod in a partially loaded core (Assembly I-B). This implies that the worth does not change as a function of core loadings within the range of interest. This estimate is illustrated in Figure 6-4 where the fully loaded core reactivity measurements for insertion of stainless steel rods are combined with the measurements for substitution of BeO by stainless steel tightener rods to predict the worth of BeO tightener rod removal. The predicted value at the core edge is -3.8ϕ which is in good agreement with the measurement in a partially loaded core (Assembly I-B) of -4.0ϕ . The reactivity worths for the stainless steel rods shown in Figure 6-4 are 18% of the actually measured values. This adjustment is necessary because the tightener rods are smaller than the stainless steel rods (fuel rod size) used for the steel insertion reactivity measurements.

The uncertainty in the estimated Core II minimum critical loading is ± 5 rods. This includes the uncertainty of 2 rods in the measured BeO to steel interchange, an uncertainty of 2 rods (10%) in the postulated addition of 18 rods and an uncertainty of 1 rod in the Core I critical mass.

The above value of 536 standard fuel rods is for a core containing no Instrumented Fuel Assemblies (IFA's), and uniform standard fuel rods without burnup or Pu-241 decay from the as-fabricated content. If the Core II critical measurements were actually performed today, the estimated minimum critical size would be 554 ± 6 fuel rods and 92 stainless steel tightener rods. The additional 18 fuel rods would be required to offset the reactivity losses of the centrally located lower Pu fuel rods (8 rods⁽⁶⁻¹⁾), Pu-241 decay over the 29 month period since May 1969 (1.8 rods*), a burnup of 865 MW-days (2.6 rods*), and installation of the six IFA's (5 rods*).

*Using, as discussed above, a reactivity worth of 12.7ϕ per rod.

Figure 6-4. Reactivity Worth per Rod for BeO and Stainless Steel Rods

6.2 CORE II MATERIAL WORTH MEASUREMENTS

The reactivity worths of fuel B_4C and UO_2 rods were determined by observing the reactor temperature and reflector configurations for different core loadings. The critical conditions were recorded for the initial core loading, a loading with a fuel rod [removed from a selected location (liquid sodium thus filling the location), a loading with a reference fuel rod] inserted in the selected location, a loading with B_4C inserted, a loading with a UO_2 rod inserted, and for another measurement of the initial core loading after the original fuel rod was reinserted. The reactivity worths of the guinea pig rods containing one-third more Pu than standard rods were measured by observing the critical conditions after rod insertion into vacant (sodium filled) locations of Assembly II-A (Figure 6-3).

The positions within the core at which measurements were made are illustrated in Figure 6-5. Data obtained during the measurements are tabulated in Tables 6-4 through 6-10. The reactivity values are summarized in Table 6-11, and they are compared with corresponding measurements for Core I in Table 6-12. It is evident, from Table 6-12, that fuel and guinea pig rod worths for Core II are larger than in Core I while other worths are essentially unchanged.

The measured reactivity worth data are plotted as a function of core radius in Figures 6-6 through 6-11. The data in Figure 6-8 indicate an apparently anomalous value for the worth of a UO_2 rod replacing sodium. A careful inspection of Figures 6-6 and 6-7, combined with the smooth trends for the UO_2 substitutions illustrated in Figures 6-9 and 6-10, indicates that it may be the sodium measurement at a radius of 26.6 cm (Row 1 in Table 6-7) which is anomalous.* If this were so, the values at 26.6 cm in Figures 6-9 through 6-11 would be unchanged, but the corresponding values in Table 6-11, and in Figures 6-6 through 6-8 would be less.

A detailed comparison of measurements and calculations is being prepared and will be reported later.

6.3 CORE II DELAYED NEUTRON FRACTIONS

The in-hour equation is used to convert the measured reflector position versus period relationships into a relationship between position and reactivity. The relation between asymptotic period (T) and reactivity (K) for SEFOR is illustrated in Figure 6-12. The relationship was calculated from the in-hour equation taking into account 24-delayed neutron groups, 6 each for Pu-239, Pu-240, U-238, and U-235. The resulting expression may be written as follows:

$$K = \left(\frac{1}{T} \right) \left(\frac{\ell}{\beta} \right) + \sum_{i=1}^6 \sum_{j=1}^4 \frac{\beta_{ij}/\beta}{1 + \lambda_{ij}T}$$

with (6-2, 6-3) the effective delayed neutron fractions β_{ij} computed from the actual fractions $\bar{\beta}_{ij}$ according to the following expression:

$$\beta_{ij} = \frac{\bar{\beta}_{ij} \int d^3r \int dE' f_{ij}(E') \psi(E', r_{\infty}) \nu_j(E) \sum f_j(E, r_{\infty}) \phi(E, r_{\infty})}{\sum_{j=1}^4 \int d^3r \int dE' \chi(E') \psi(E', r_{\infty}) \nu_j(E) \sum f_j(E, r_{\infty}) \phi(E, r_{\infty})}$$

and with the effective delayed neutron fraction, β , defined by

$$\beta = \sum_{i=1}^6 \sum_{j=1}^4 \beta_{ij}$$

*If the indicated position of reflector 8 were actually 20.50 or 20.25, instead of 22.50, the worths of all materials at E 1.5 N3.0F would be decreased, relative to sodium, by $\sim 3.4\%$.

Figure 6-5. Locations for Core II Material Reactivity Worth Measurements (Assembly II-A)

Table 6-4
CRITICAL CONDITIONS FOR ASSEMBLY II-A

Reflector 3		Reflector 8		Average Temperature (°F)	Relative Reactivity ^(a) at 350°F
Position (cm)	Worth (β)	Position (cm)	Worth (β)		
47.91	77.30	0.16	0.01	353.5	75.21
49.60	80.89	0.16	0.01	359.5	75.20
47.20	75.78	0.18	0.02	350.5	75.50
48.31	78.15	0.18	0.02	353.5	76.07
47.33	76.06	0.16	0.01	350.5	75.77
47.64	76.72	0.16	0.01	351.0	76.13
					<hr/> 75.65

(a) Reactivity corrections for temperature changes were made using a coefficient of $-0.6\beta/^\circ\text{F}$.

Table 6-5
CORE II MATERIAL WORTH MEASUREMENTS AT W2.5 S5.0B

Material in Core Location	Reflector 3		Reflector 8		Temperature (°F)	Relative ^(a) Reactivity at 350°F (β)	Reactivity Change (β)
	Position (cm)	Worth (β)	Position (cm)	Worth (β)			
Sodium	47.90	77.28	13.80	13.59	358.5	85.77	0.0
Fuel Rod 781	49.49	80.66	0.16	0.01	358.5	75.57	10.2
B ₄ C Rod X21	47.90	77.28	23.02	27.16	359.0	99.04	-13.3
UO ₂ Rod DE	47.77	77.00	10.00	9.19	351.5	85.29	0.5
Fuel Rod [*] 121	Base (Table 6-4)				350.0	75.65	10.1

(a) Reactivity corrections for temperature changes were made using a coefficient of $-0.6\beta/^\circ\text{F}$.

Table 6-6
CORE MATERIAL WORTH MEASUREMENTS AT W2.0 S4.0B

Material In Core Location	Reflector 3		Reflector 8		Temperature (°F)	Relative ^(a) Reactivity at 350°F (¢)	Reactivity Change (¢)
	Position (cm)	Worth (¢)	Position (cm)	Worth (¢)			
Sodium	49.98	81.69	15.00	15.12	358.0	92.01	0.0
Fuel Rod 781	48.01	77.51	0.16	0.01	353.5	75.42	16.6
B ₄ C Rod X21	47.90	77.28	28.39	36.80	350.5	113.78	-21.8
UO ₂ Rod DE	47.90	77.28	16.08	16.57	349.5	94.15	-2.1
Fuel Rod 635	Base (Table 6-4)				350.0	75.65	16.4

(a) Reactivity corrections for temperature changes were made using a coefficient of $-0.6¢/°F$.

Table 6-7
CORE II MATERIAL WORTH MEASUREMENTS AT E1.5 N3.0F

Material In Core Location	Reflector 3		Reflector 8		Temperature (°F)	Relative ^(a) Reactivity at 350°F (¢)	Reactivity Change (¢)
	Position (cm)	Worth (¢)	Position (cm)	Worth (¢)			
Sodium	47.90	77.28	22.50	26.30	351.5	102.68	0.0
Fuel Rod 781	47.90	77.28	0.16	0.01	351.5	76.39	26.3
B ₄ C Rod X21	41.49	63.67	46.50	73.50	349.0	137.77	-35.1
UO ₂ Rod DE	46.10	73.45	25.50	31.45	348.0	106.10	-3.4
Fuel Rod 692	Base (Table 6-4)				350.0	75.65	27.0

(a) Reactivity corrections for temperature changes were made using a coefficient of $-0.6¢/°F$.

Table 6-8
CORE II MATERIAL WORTH MEASUREMENTS AT W1.0 S2.0B

Material In Core Location	Reflector 3		Reflector 8		Temperature (°F)	Relative ^(a) Reactivity at 350°F (¢)	Reactivity Change (¢)
	Position (cm)	Worth (¢)	Position (cm)	Worth (¢)			
Sodium	48.09	77.68	24.50	29.68	349.0	107.96	0.0
Fuel Rod 781	46.67	74.66	0.18	0.02	350.0	74.68	33.3
B ₄ C Rod X21	48.22	77.96	52.00	84.84	351.0	162.20	-54.2
UO ₂ Rod DE	49.08	79.79	30.00	39.91	352.0	118.50	-10.5
Fuel Rod 783	Base (Table 6-4)				350.0	75.65	32.3

(a) Reactivity corrections for temperature changes were made using a coefficient of $-0.6¢/°F$.

Table 6-9
CORE II MATERIAL WORTH MEASUREMENTS AT W0.5 S1.0F

Material In Core Location	Reflector 3		Reflector 8		Temperature (°F)	Relative ^(a) Reactivity at 350°F (¢)	Reactivity Change (¢)
	Position (cm)	Worth (¢)	Position (cm)	Worth (¢)			
Sodium	47.90	77.28	30.18	40.26	357.0	113.34	0.0
Fuel Rod 781	47.69	76.83	0.18	0.02	355.0	73.85	39.5
B ₄ C Rod X21	47.90	77.28	64.62	108.91	353.5	184.09	-70.8
UO ₂ Rod DE	47.90	77.28	35.56	51.05	352.0	127.13	-13.8
Fuel Rod 842	Base (Table 6-4)				350.0	75.65	37.7

(a) Reactivity corrections for temperature changes were made using a coefficient of $-0.6¢/°F$.

Table 6-10
CORE II GUINEA PIG MATERIAL WORTH MEASUREMENTS

Core Location	Material In Core Location	Reflector 3		Reflector 8		Temperature (°F)	Relative ^(a) Reactivity at 350°F	Reactivity Change
		Position (cm)	Worth (¢)	Position (cm)	Worth (¢)		(¢)	
E0.0 N4.0A	Sodium	Base (Table 6-4)				350.0	75.65	0
E0.0 N4.0A	Guinea Pig A04	28.55	37.03	0.16	0.01	350.0	37.04	38.6
E2.0 N2.0C	Sodium	Base (Table 6-4)				350.0	75.65	0
E2.0 N2.0C	Guinea Pig A04	22.58	26.46	0.16	0.01	351.0	25.87	49.8

(a) Reactivity corrections for temperature changes were made using a coefficient of $-0.6¢/°F$.

Table 6-11
CORE II MATERIAL REACTIVITY WORTH MEASUREMENTS
(Assembly II-A)

Core Location	Type of Rod	Worth Relative to Sodium Measured ^(a) (d)
W2.5 S5.0B (Radius = 42.7 cm)	Fuel (121)	+10.2
	Fuel (781)	+10.1
	UO ₂ (DE)	+ 0.5
	B ₄ C (X21)	-13.3
W2.0 S4.0B (Radius = 34.6 cm)	Fuel (635)	+16.6
	Fuel (781)	+16.4
	UO ₂ (DE)	-2.1
	B ₄ C (X21)	-21.8
E1.5 N3.0F (Radius = 26.6 cm)	Fuel (692)	+27.0
	Fuel (781)	+26.3
	UO ₂ (DE)	-3.4
	B ₄ C (X21)	-35.1
E0.0 N4.0A (Radius = 26.6 cm)	Guinea Pig (A04)	+38.6
W1.0 S2.0B (Radius = 18.6 cm)	Fuel (783)	+32.3
	Fuel (781)	+33.3
	UO ₂ (DE)	-10.5
	B ₄ C (X21)	-54.2
E2.0 N2.0C (Radius = 18.6 cm)	Guinea Pig (A04)	+49.8
W0.5 S1.0F (Radius = 5.8 cm)	Fuel (842)	+39.5
	Fuel (781)	+37.7
	UO ₂ (DE)	-13.8
	B ₄ C (X21)	-70.8

(a) The estimated standard deviation in a measurement is $\pm 1.5\%$ or $\pm 4\%$, whichever is larger.

Table 6-12
COMPARISON OF CORE I AND CORE II
REACTIVITY WORTH MEASUREMENTS

Core Location	Type of Rod	Worth Relative to Sodium		Ratio of Core II to Core I Worths
		Core II (¢)	Core I ^(a) (¢)	
W2.5 S5.0B (Radius = 42.7 cm)	Fuel (121)	+10.2	+ 8.7	1.13 ^(b)
	Fuel (781)	+10.1	+ 9.2	
	UO ₂	+0.5	0.0	—
	B ₄ C	-13.3	-13.0	1.02
W2.0 S4.0B (Radius = 34.6 cm)	Fuel (635)	+16.6	+14.6	1.10 ^(b)
	Fuel (781)	+16.4	+15.4	
	UO ₂	-2.1	-2.7	—
	B ₄ C	-21.8	-22.1	0.99
E0.0 N4.0A (Radius = 26.6 cm)	Guinea Pig (A04)	+38.6	+33.5	1.15
W1.0 S2.0B (Radius = 18.6 cm)	Fuel (783)	+32.3	+28.4	1.15 ^(b)
	Fuel (781)	+33.3	+28.8	
	UO ₂	-10.5	-9.8	—
	B ₄ C	-54.2	-52.3	1.04
E2.0 N2.0C (Radius = 18.6 cm)	Guinea Pig (A04)	+49.8	—	1.19
W2.5 N1.0A (Radius = 18.6 cm)	Guinea Pig (A04)	—	+41.8	
W0.5 S1.0F (Radius = 5.8 cm)	Fuel (842)	+39.5	—	1.12 ^(b)
	Fuel (781)	+37.7	+34.5	
	UO ₂	-13.8	-13.0	—
	B ₄ C	-70.8	-70.9	1.00

(a) Taken from Tables 4-2 and 4-3 of reference⁶⁻¹.

(b) Averaged value.

Figure 6-6. Reactivity Worths of Inserting Fuel Rods into Core II

Figure 6-7. Reactivity Worths of Inserting B_4C Rods into Core II

Figure 6-8. Reactivity Worths of Inserting UO_2 Rods into Core II

Figure 6-9. Reactivity Worths of Replacing B_4C Rods by Fuel Rods in Core II

Figure 6-10. Reactivity Worths of Replacing UO_2 Rods by Fuel Rods in Core II

Figure 6-11. Reactivity Worths of Replacing UO_2 Rods by B_4C Rods in Core II

Figure 6-12. Relationship Between Excess Reactivity and Reactor Period for SEFOR Core II

where:

T = period in seconds
 k = reactivity in dollars
 ℓ = neutron lifetime
 $f_{ij}(E)$ = energy spectrum for delay group i and isotope j
 $\chi(E)$ = energy spectrum of prompt neutrons
 $\nu_j(E)$ = prompt neutrons per fission for isotope j
 $\sum f_j(E, \lambda_j)$ = fission cross section for isotope j
 i = index for delayed neutron group
 j = index for isotope
 $\phi(E, \lambda_j) dE d^3r$ = neutron flux
 $\psi(E, \lambda_j) dE d^3r$ = adjoint flux
 $\bar{\beta}_{ij}$ = delayed neutron fraction for delay group i and isotope j
 β_{ij} = the effective delayed neutron fraction for delay group i and isotope j
 λ_{ij} = decay constant for delay group i and isotope j

The effective delayed neutron fractions by group and by isotope, β_{ij} , were calculated from the above expression with the SYN⁽⁶⁻⁴⁾ program. The values for λ_{ij} , $\bar{\beta}_{ij}$, and $f_{ij}(E)$ were taken from Keepin⁽⁶⁻⁵⁾. The calculated β_{ij} as well as the λ_{ij} taken from Keepin⁽⁶⁻⁵⁾, are shown in Table 6-13.

For period measurements of the order of 100 sec, (the range of periods used in reflector calibrations), the first term, $(1/T)(\ell/\beta)$, in the in-hour equation contributes $\sim 0.02\%$ to the value of K in SEFOR and it may, for all practical purposes, be ignored.

The calculated relations between period and reactivity for Core II are compared with the calculated* relations⁽⁶⁻¹⁾ for Core I in Table 6-14. As the values indicate, the reactivities corresponding to a given period are essentially the same in both cases, the Core II reactivities being only $\sim 0.1\%$ larger than the Core I values.

6.4 CORE II REFLECTOR CALIBRATIONS

The magnitudes of most reactivity changes between different steady state reactor conditions in SEFOR are determined from observed changes in the positions of the two calibrated fine reflectors 3, and 8. These reflectors were calibrated using the asymptotic period technique described below. Core I calibrations are described in references 5-5 and 6-1.

The reflector calibrations were performed in the following manner: The reactor was brought to critical at low power (~ 600 watts) with the fine reflector to be calibrated in the completely lowered position, and the other fine reflector in the completely raised position. 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 were maintained in the completely raised position). The reflector to be calibrated was raised a few centimeters to produce an asymptotic period of the order of 100 seconds; the period was measured, and the other fine reflector was lowered to bring the reactor back to critical at the original flux level. This process was repeated until the entire length of the fine reflector had been calibrated. The accuracy in the measurement periods is estimated⁽⁶⁻¹⁾ to be $\sim 1\%$.

Earlier⁽⁶⁻¹⁾ reflector worth measurements demonstrated that the worth of a fine reflector is slightly dependent on the configuration of other reflectors even when the reflectors adjacent to a fine reflector are always maintained in the completely raised position. To provide calibration curves for the different coarse reflector configurations expected

*The calculated relation was modified as a result of the zero power oscillator tests⁽⁵⁻⁵⁾, but later transient tests⁽⁵⁻²⁾ were more in agreement with the originally calculated values shown here.

Table 6-13
DELAYED NEUTRON PARAMETERS FOR CORE II

U-238		Pu-239		Pu-240		U-235	
$\beta_{ij} \times 10^5$	λ_{ij} (sec ⁻¹)	$\beta_{ij} \times 10^5$	λ_{ij} (sec ⁻¹)	$\beta_{ij} \times 10^5$	λ_{ij} (sec ⁻¹)	$\beta_{ij} \times 10^5$	λ_{ij} (sec ⁻¹)
1.746	0.0132	6.119	0.0129	0.131	0.0129	0.167	0.0127
18.920	0.0321	45.948	0.0311	1.298	0.0313	0.958	0.0317
22.105	0.139	35.067	0.134	0.902	0.135	0.835	0.115
53.472	0.358	53.741	0.331	1.661	0.333	1.827	0.311
30.973	1.41	16.849	1.26	0.606	1.36	0.574	1.40
<u>10.324</u>	4.02	<u>5.697</u>	3.21	<u>0.137</u>	4.04	<u>0.117</u>	3.87
137.540		163.421		4.735		4.478	

$$\beta = \sum_i \sum_j \beta_{ij} = 0.00310$$

Table 6-14

COMPARISON OF CORE I AND CORE II REACTIVITY CONVERSION FORMULAS

Period (sec)	Reactivity (β)	
	Core I	Core II
10	36.104	36.138
15	29.786	29.816
20	25.646	25.673
30	20.353	20.375
40	17.012	17.030
50	14.669	14.685
60	12.920	12.934
70	11.558	11.571
80	10.464	10.476
90	9.5648	9.5751
100	8.8110	8.8204
130	7.1349	7.1424
150	6.3357	6.3424
175	5.5598	5.5657
200	4.9546	4.9597
300	3.4552	3.4587
400	2.6541	2.6569
500	2.1551	2.1573
600	1.8142	1.8161
700	1.5665	1.5681
800	1.3784	1.3798
900	1.2306	1.2319
1000	1.1115	1.1126
3600	0.3161	0.3164

during operation, the fine reflectors were calibrated with the configurations that resulted for two core loadings, Assembly II-A and Assembly II-C, at 350°F. Assembly II-A (shown in Figure 6-3) was the initial Core II configuration, and Assembly II-C as shown in Figure 6-13 is the loading that will be used for temperature coefficient measurements at zero power. Additional calibrations will be performed on Assembly II-C at elevated temperatures.

The calibrations are listed in Tables 6-15 through 6-18, where the conversion from measured doubling time (which is the period multiplied by $\ln 2$) to reactivity was made using the in-hour equation of the previous section. The total worths of the fine reflectors decrease between Assemblies II-A and II-C. The decrease in worths is expected, since more reflectors are in the lowered position during the calibrations in II-C than in II-A. This effect of reflector configuration has been previously (5-5, 6-1) observed. A comparison of values also shows that the apparent worth of reflector 3 decreased more during the loading change from Assembly II-A to Assembly II-C than did that of reflector 8. Inspection of Figures 6-3 and 6-13 shows that B₄C was loaded on the reflector 3 side of the core in converting Assembly II-A into Assembly II-C. This change apparently influenced the worth of reflector 3 more than that of reflector 8. Similar effects of B₄C distribution on reflector worth were previously (6-1) observed in Assembly I-E.

A comparison between the fine reflector worths in Core I and Core II is given in Table 6-19. Insofar as possible, the comparison is made between calibrations performed with similar core loadings and reflector configurations. As the values in the table indicate, the worths of the reflectors have increased by ~8% due to the increased leakage and reflection associated with the harder Core II neutron spectrum.

Coarse reflector worths were determined for Assembly II-A. For these measurements, a coarse reflector was lowered, and one of the calibrated fine reflectors was raised to maintain criticality. The other fine reflector was maintained at a constant partially raised position. The results of the measurements are tabulated in Table 6-20. It is evident that the worth of a reflector is less when an adjacent reflector is partially lowered (e.g., reflectors 2 and 4 when 3 is partially lowered, or 7 and 9 when 8 is partially lowered) than when adjacent reflectors are completely raised (e.g., 1, 5, 6 and 10) during the measurement.

This reflector interaction, or "shadowing" effect has been previously observed, and reported in greater detail in reference 6-1. These results reaffirm that the fine reflectors which are used to determine feedback effects should be calibrated with both adjacent reflectors completely raised and, insofar as practical, they should be calibrated with a coarse reflector configuration similar to that which will be used during the experimental program at power.

6.5 IRRADIATED FUEL EXAMINATION

Three experimental test rods were received at the RML of the Vallecitos Nuclear Center from SEFOR on September 24. The test rods which were received are identified as Guinea Pig rods A02 and A16, the third rod is the capture-to-fission ratio rod (CFRR). The CFRR was sectioned and the ampule capsule was retrieved and sent intact to the Chemical Engineering Laboratory for analysis.

Plans for the destructive examination of SEFOR fuel rods A02 and A16 have been formulated, and the work is scheduled to begin in the current quarter.

Figure 6-13. Core Loading for Assembly II-C

Table 6-15
CALIBRATION OF REFLECTOR 3
ASSEMBLY II-A

Reflector Position		Doubling Time (sec)	Reactivity Increment (β)	Cumulative Reactivity (β)
Initial (cm)	Final (cm)			
—	0.31	—	0.0	0.0
0.31	9.59	73.5	8.436	8.436
9.59	15.10	96.6	6.756	15.192
15.10	20.01	90.2	7.149	22.341
20.01	25.01	75.5	8.257	30.599
25.01	30.02	65.5	9.236	39.835
30.02	35.00	58.9	10.025	49.860
35.00	39.07	70.9	8.680	58.540
39.07	43.07	73.08	8.474	67.014
43.07	47.08	72.68	8.511	75.525
47.08	51.01	74.80	8.319	83.845
51.01	54.07	107.69	6.169	90.013
54.07	58.57	67.97	8.973	98.987
58.57	63.06	73.83	8.406	107.393
63.06	67.93	72.97	8.485	115.877
67.93	74.00	62.67	9.558	125.435
74.00	82.12	56.30	10.375	135.810
82.12	95.19	61.22	9.732	145.542

Table 6-16
CALIBRATION OF REFLECTOR 8
ASSEMBLY II-A

Reflector Position		Doubling Time (sec)	Reactivity Increment (β)	Cumulative Reactivity (β)
Initial (cm)	Final (cm)			
—	0.15	—	0.0	0.0
0.15	10.01	65.87	9.196	9.196
10.01	15.08	110.71	6.027	15.223
15.08	19.98	91.72	7.052	22.274
19.98	25.04	74.43	8.352	30.626
25.04	30.00	65.11	9.279	39.906
30.00	35.02	58.6	10.064	49.969
35.02	38.99	78.81	7.978	57.948
38.99	44.07	54.73	10.598	68.546
44.07	47.03	110.60	6.032	74.578
47.03	50.96	76.99	8.129	82.707
50.96	55.13	73.31	8.453	91.161
55.13	59.56	71.55	8.618	99.778
59.56	64.51	68.32	8.937	108.715
64.51	70.03	64.75	9.319	118.035
70.03	76.00	71.22	8.649	126.684
76.00	84.11	66.51	9.127	135.811
84.11	94.96	94.16	6.901	142.711

Table 6-17
 CALIBRATION OF REFLECTOR 3
 ASSEMBLY II-C at 350°F

(Reflector 5 down, 10 at 15.6 cm)

Reflector Position		Doubling Time (sec)	Reactivity Increment (β)	Cumulative Reactivity (β)
Initial (cm)	Final (cm)			
—	0.26	—	0.0	0.0
0.26	12.11	54.2	10.676	10.676
12.11	17.05	111.6	5.986	16.662
17.05	22.06	94.2	6.898	23.560
22.06	27.04	84.5	7.541	31.101
27.04	32.00	66.4	9.139	40.240
32.00	35.98	88.5	7.262	47.502
35.98	40.02	81.1	7.796	55.298
40.02	43.97	83.8	7.592	62.891
43.97	48.07	79.5	7.923	70.813
48.07	53.04	62.6	9.566	80.380
53.04	58.02	66.8	9.096	89.476
58.02	63.16	70.2	8.748	98.224
63.16	68.09	81.5	7.765	105.989
68.09	75.06	59.5	9.947	115.937
75.06	82.60	76.9	8.137	124.074
82.60	95.22	75.9	8.223	132.296

Table 6-18
CALIBRATION OF REFLECTOR 8
ASSEMBLY II-C at 350°F

(Reflector 5 down, 10 at 15.6 cm)

Reflector Position		Doubling Time (sec)	Reactivity Increment (ρ)	Cumulative Reactivity (ρ)
Initial (cm)	Final (cm)			
	0.20			0.0
0.20	10.04	71.0	8.670	8.670
10.04	16.19	84.9	7.512	16.183
16.19	23.08	58.85	10.031	26.214
23.08	28.59	62.4	9.590	35.804
28.59	34.07	57.5	10.210	46.014
34.07	38.11	79.3	7.939	53.953
38.11	42.39	73.6	8.427	62.379
42.39	47.60	55.7	10.459	72.838
47.60	52.53	59.0	10.012	82.850
52.53	57.49	62.2	9.614	92.464
57.49	62.56	66.5	9.128	101.592
62.56	68.58	62.0	9.638	111.230
68.58	75.02	65.9	9.193	120.422
75.02	82.53	72.8	8.500	128.923
82.53	94.96	75.0	8.301	137.224

Table 6-19
COMPARISON OF REFLECTOR WORTHS
IN CORE I AND CORE II

Assembly	Reflector	Reflector Worth	Ratio of Worth Core II to Core I Worth
II-A	3	145.5	
I-I at 760°F	3	132.0 ^(a)	1.10
II-A	8	142.7	
I-I at 760°F	8	131.6 ^(a)	1.08
I-G	8	133.1 ^(a)	1.07
II-C at 350°F	3	132.3	
I-H	3	125.1 ^(a)	1.06
II-C at 350°F	8	137.2	
I-I at 550°F	8	130.0 ^(a)	1.06

(a) Core I worths include a reduction of 1.3¢ to account⁽⁵⁻⁵⁾ for a reduction in the strokes of the fine reflectors between these Core I and Core II measurements. Worths for Core I are the values obtained using the calculated⁽⁶⁻¹⁾ parameters for the in-hour equation to convert measured periods into reactivities.

Table 6-20
REFLECTOR WORTHS IN ASSEMBLY II-A

Reflector	Worth Measured Using Reflector 8 (Reflector 3 at 47.81 cm) (\$)	Reflector	Worth Measured Using Reflector 3 (Reflector 8 at 47.79 cm) (\$)
1	1.41	6	1.40
2	1.29	7	1.25
3	1.46 ^(a)	8	1.43 ^(a)
4	1.23	9	1.26
5	1.39	10	1.44

(a) Worths obtained by period measurements.

REFERENCES

- 6-1. L. D. Noble et al., "Results of SEFOR Zero Power Experiments", GEAP-13588, March 1970.
- 6-2. A. F. Henry, Nuclear Science and Engineering, 3, 52 (1958).
- 6-3. L. N. Ussachoff, Conference on Peaceful Uses of Atomic Energy, Geneva 1955, Paper No. A/Conference 8/P/656.
- 6-4. M. D. Kelley, "SYN-Operational Aspects of the Combined BISYN, PERT, and BICYCL Synthesis Codes", GEAP-13672, December 1970.
- 6-5 G. R. Keepin, "Physics of Nuclear Kinetics", Addison-Wesley, Palo Alto, 1965.

7. SEFOR SAFETY AUDIT AND LICENSING

7.1 OBJECTIVES

The objectives of Safety and Licensing in relation to SEFOR are: (1) to provide an independent safety audit of SEFOR operations, (2) to provide support in assessing safety significance of reactor operations, the experimental program, and modifications to the reactor system, and (3) to handle licensing efforts — special reports required by the license and Technical Specification changes.

7.2 CURRENT RESULTS

Two proposed changes to the SEFOR Technical Specifications were submitted to the AEC-DRL:

Proposed Change No. 7 Incorporation of BRD and SEFOR organizational changes into the Administrative Controls Section of the Technical Specifications.

Proposed Change No. 8 Permit initiation of the Core II transient test program after completing static and oscillator tests at 10 MW and permit a linear interpolation for Core II transient test slug worth as a function of the doppler coefficient; reduce the frequency of fuel surveillance and sodium sampling required during super-prompt transient tests, and clarify the requirements for static and oscillator tests during the Core II test program.

Approval of three changes to the Technical Specifications were received from the DRL:

Change No. 4 Addition of transient flux detectors to the list of equipment required to meet the limiting conditions for operations.

Change No. 5 Increase of the minimum allowable travel time for the Fast Reactivity Excursion Device.

Change No. 6 Reduction in the permissible rate of change of primary sodium temperature and addition of surveillance requirements for the reactor vessel outer head bolts.

Three safety evaluations were completed for:

- (1) Modification Request No. 760 — WRM Picoammeter Protection for the Super-prompt Excursion Tests,
- (2) Instrumented Fuel Assembly VNC 27 assembled without a spacer, and
- (3) Change from Core I to Core II.

The Quality Assurance Program Plan for SEFOR was reviewed and sent to the DRL as was a report summarizing the results of the FRED poison slug cladding temperature analysis.

Two safety audit trips were made to the site to review compliance with the Technical Specifications and also to review transient test procedures, equipment, and results. A response to the report by the Safety Review Committee on their seventh meeting was issued. The thermal model used in the reactor vessel flange stress analysis was reviewed and recommendations were made for improving the model. A draft copy of the Stress Analysis Report is currently being reviewed and prepared for publication.

8. SEFOR SUPPORT

The Plant Engineering effort is a supporting function to the SEFOR Facility. During August and September, the super-prompt critical transients were performed and the Core I Test Program completed. During October, conversion to Core II was accomplished and the zero power tests performed.

8.1 VESSEL STRESS ANALYSIS

A stress report was prepared concerning the analysis of the reactor vessel wall-to-flange transition region. The investigation was undertaken to verify previous analyses and show that the reactor vessel stresses will remain within acceptable values during operation.

The stress analysis was performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1971. Thermal fatigue effects were evaluated at the locations of highest stress.

Conclusions from the study have shown that there has been no significant reduction in vessel fatigue life due to past operations, and the maximum calculated usage factor for the reactor vessel is well within acceptable limits.

Usage factors for the areas of maximum stress are:

Joint Number	Accumulated Usage Factors	
	Through July 27, 1971	Total Predicted
17	0	0
8	0.9	3.0
6	1.7	3.3
5	0.8	5.2

These usage factors cover an envelope of transients which cover all normal operating and accident conditions.

8.2 ADDITIONAL SUPPORT

- The SEFOR Facility Quality Assurance Plan was prepared for covering operation of the facility under 10CFR50.
- Drawings were completed on a shim for the FRED positioner which allows adjustment of the reactivity of the super-prompt transient slug.
- Performance tests were completed for the nitrogen cooling system and air blast coolers in order to provide information relating to performance capability for the SEFOR Follow-On Program.
- A review was completed of the requirements for examination of the Inconel spring material for the tightener rods and a program was outlined covering the examination of three categories of material:
 - a. Material fully formed and heat treated without reactor material
 - b. Material subjected to the reactor thermal conditions but without significant radiation exposure
 - c. Material taken from the spring closest to the core
- An Engineering Review of the 26.5V DC System was completed covering system design, load analysis, and charger performance.
- The design of a modification to the one-half ton grapple was completed which will prevent rotation of the internals during operation.

9. SEFOR FACILITY

9.1 EXPERIMENTAL PROGRAM

9.1.1 Transient Tests (Core I)

During the period from August 14, 1971 until the 30th, the super-prompt transient tests were performed. Interspersed between the super-prompt transient tests were two sub-prompt transient which were used as calibration tests. The testing sequence is shown in Table 9-1 and is described below.

9.1.1.1 Sub-prompt Transient Test I-B (0.8\$ Slug Worth and 1 MW Initial Power Level)

Because the fission detectors were modified by using B_4C as a shielding material instead of lead shot, it was decided to repeat one of the previous subprompt transient tests. This test was a nominal 80¢ transient from 1 MW. There were no electrical signal filters used in any of the transient detector channels. For this test, the super-prompt poison slug was lowered to a positioner position of 6.0 inches. The measured static worth of the slug used for this test was 77¢. From this test, it was concluded that the modified fission detectors behaved very similar to that prior to modification. During this test, an intercomparison of WRM's and fission chambers was made. The results showed very good agreement.

9.1.1.2 Super-prompt Transient Test 1

On August 14, 1971, at 1440, the first super-prompt critical transient test was performed. For this test, the nominal slug worth was 1.15\$ and the initial power level was 2.0 MW. The slug was lowered to a positioner position of 1.40 inches. For channel A (data recording channel), a fission detector was used. For Channel B, automatic switchover was used. Automatic switchover consists of recording the fission chamber data up to a preset power level, then, under computer command, the signals from the gamma chambers are recorded until a pre-selected power level is again encountered for declining power. When the pre-selected declining power level is encountered, the data recording switches from the gamma chambers back to the fission detectors for the remainder of the transient. Preliminary evaluation of the data indicates that test results agree closely with predictions.

9.1.1.3 Sub-prompt Transient I-C (0.8\$ Slug Worth and 1 MW Initial Power Level)

The test was performed to determine the static worth as measured independently by reflectors 3 and 8, and to perform a repeat check of the dynamic worth as measured in Sub-prompt Transient 1B. Calibration with reflector 8 showed the slug worth to be 1.23\$ as measured originally on June 14, 1971. Calibration with reflector 3 resulted in a slug worth of 1.19\$. The dynamic worth was again measured to be the same as in Sub-prompt Test 1B.

9.1.1.4 Super-Prompt Test 2 through 8

These seven super-prompt tests were performed in a routine manner. Results were in good agreement with the predictions.

Test 2 was done with automatic switchover from fission detector to gamma chamber on channel B, and with a fission detector connected continuously on channel A. The fission detector remained on-scale throughout the transient; therefore, in the remaining six transients, switchover was not necessary.

9.1.1.5 Super-prompt Poison Slug Calibrations

Prior to the super-prompt critical tests, the super-prompt poison slug was calibrated by the "rod-swap" method. The data obtained are presented in Figure 9-1. These calibrations were performed by swapping with the fine reflector 3 while the fine reflector 8 remained at a fixed position of 45.00 cm. These measurements gave a value of $1.19\$ \pm 0.015$ for the total reactivity worth of this poison slug. However, the projected worth of the poison slug based upon June 14, 1971 measurements with reflector 8 was 1.23\$. These differences were later determined to be caused by the different reflector interaction effects. It should be noted that the June measurements were performed by swapping with reflector 8, while reflector 3 remained at a fixed position. Reflector 3 was used during the second calibration because the digital position indicator on reflector 8 was out of service.

Table 9-1
SUPER-PROMPT TRANSIENT TEST SEQUENCE

	Reactor Power	Nominal Slug Worth	Date
Subprompt Transient 1B	1 MW	0.8\$	Aug 13
Superprompt Transient 1	2 MW	1.18\$	Aug 14
Subprompt Transient 1C	1 MW	0.8\$	Aug 16
Superprompt Transient 2	2 MW	1.28\$	Aug 17
Superprompt Transient 3	2 MW	1.28\$	Aug 19
Superprompt Transient 4	5 MW	1.18\$	Aug 21
Superprompt Transient 5	5 MW	1.28\$	Aug 24
Superprompt Transient 6	8 MW	1.18\$	Aug 26
Superprompt Transient 7	8 MW	1.28\$	Aug 28
Superprompt Transient 8	8 MW	1.28\$	Aug 30

Figure 9-1. Reactivity Worth of Positioner FPED Slug August 14, 1971

9.1.1.6 Fission Detector Modifications

Transient fission detectors were reinstalled in their appropriate locations in July, after changing the shielding material to B_4C from lead shot to reduce the capture/fission ratio of U-238 in these fission detectors. Subsequent measurements of the voltage outputs from these detectors as a function of the steady state reactor power level show that the background activity (as a result of U-238 captures) was reduced by a factor of 5 as expected. However, the actual responses from these detectors essentially remained the same.

At the steady state reactor power levels of 5 MW and 9 MW, the signals from the fission and gamma detectors were analyzed for 2000 Hz and 60 Hz noise levels. There was no measureable noise (2000 Hz and 60 Hz) in the signals.

9.1.1.7 Fuel Surveillance

Fuel surveillance was performed on one guinea pig rod and one standard rod after each super-prompt transient test. All fuel rod measurements were unchanged within the limits of repeatability of the inspection device. Results are tabulated in Table 9-2.

9.1.2 Fission-to-Capture Measurement (Core I)

A special core rod designed to measure the fission-to-capture ratio in Pu-239, U-238 and U-235 was placed in the reactor on September 7, replacing fuel rod A02. The rod was irradiated at 10 megawatts reactor power for 90 MW days, and returned to Vallecitos Nuclear Center along with fuel rods A16 and A02 in the fuel shipping cask. The irradiation began on September 9.

9.1.3 Air Blast Cooler and Nitrogen Cooling Tests

During startup and power ascension for the fission-to-capture rod irradiation, data were recorded to determine the performance capabilities of the air blast cooler and nitrogen cooling system. The data are needed for follow-on program studies.

9.1.4 Core II Changeover

One-hundred-two BeO tightening rods were removed from the reactor and replaced with steel tightening rods to effect the changeover from Core I to Core II. The six BeO tightening rods in the instrumented fuel assemblies could not be removed and will remain in the reactor during Core II. The associated reactivity loss was regained by removing B_4C rods and replacing them with fuel rods and steel rods.

This changeover was performed in five incremental loading steps. At the end of each of the loading steps, a criticality check was performed. Total reactivity loss due to the changeover is evaluated to be \$2.44.

9.1.5 Oak Ridge National Laboratory (ORNL) Noise Measurements, Core I and II

During the capture-to-fission measurements (Core I) the reactor was operated at 10 MW. ORNL personnel recorded several sets of reactor "noise" data consisting of response of fission detectors and normal and failed thermocouples (T/C) in the instrumented fuel assemblies. The data analysis will be done at Oak Ridge.

With thru-head rods installed, a first set of critical and sub-critical "noise" measurements was then done in Core I.

In Core II, without thru-head fuel rods in the core, another set of "noise" measurements was performed. Subsequently, all of the thru-head fuel rods were reloaded. Then, the final set of "noise" measurements was performed (Core II-C). The data analysis is in progress.

9.1.6 Core II Zero Power Tests

9.1.6.1 Reflector Calibration (Core II-A)

The reflectors were calibrated with the vessel inner head and the thru-head fuel removed.

Table 9-2
FUEL SURVEILLANCE RESULTS
ROD A18

	Pre-irradiation	Superprompt 1	Superprompt 3	Superprompt 5	Superprompt 7
Temperature Corrected Length (in.)	-0.019	-0.047	-0.054	-0.038	-0.056
Temperature Corrected Diameter (in.)	+0.0036	+0.0033	+0.0033	+0.0036	+0.0037
Bow (in. T.I.R.)	0.020	0.066	0.071	0.073	0.076
Bottom Pellet Column Length (in.)	21 3/8	21 3/8	21 3/8	21 3/8	21 7/16
Top Pellet Column Length (in.)	14	14	14	14	13 15/16

ROD 806

Temperature Corrected Length (in.)	-0.001	-0.017	-0.018	-0.023	-0.025
Temperature Corrected Diameter (in.)	+0.0036	+0.0031	+0.0030	+0.032	+0.0031
Bow (in. T.I.R.)	0.016	0.013	0.015	0.012	0.016
Bottom Pellet Column Length (in.)	21 3/8	21 3/8	21 3/8	21 7/16	21 7/16
Top Pellet Column Length (in.)	13 15/16	14	14	13 15/16	14

Table 9-2 (Continued)
FUEL SURVEILLANCE RESULTS

ROD A02

	Pre-irradiation	Superprompt 2	Superprompt 4	Superprompt 6	Superprompt 8
Temperature Corrected Length (in.)	-0.008	-0.036	-0.040	-0.035	-0.036
Temperature Corrected Diameter (in.)	+0.0032	+0.0031	+0.0031	+0.0031	+0.0030
Bow (in. T.I.R.)	0.016	0.024	0.020	0.016	0.020
Bottom Pellet Column Length (in.)	21 3/8	21 5/16	21 3/8	21 3/8	21 7/16
Top Pellet Column Length (in.)	13 15/16	13 15/16	13 15/16	13 7/8	13 15/16

ROD 807

Temperature Corrected Length (in.)	-0.006	-0.028	-0.028	-0.028	-0.033
Temperature Corrected Diameter (in.)	+0.0036	+0.0029	+0.0029	+0.0031	+0.0030
Bow (in. T.I.R.)	0.019	0.018	0.021	0.018	0.020
Bottom Pellet Column Length (in.)	21 7/16	21 3/8	21 3/8	21 7/16	21 3/8
Top Pellet Column Length (in.)	14	14	14	14	14

Fuel rods A16 (removed from the reactor at the end of the subprompt transients) and A02 (removed from the reactor at the end of the superprompt transients) were helium-mass-spectrometer-leak checked and shipped to Vallecitos Nuclear Center. The rods will be destructively examined there.

Fine reflectors 3 and 8 were calibrated differentially using the period method in Core II-A (5 thru-head port locations empty and a total of 97 stainless steel tightener rods loaded).

During reflector calibration, coarse reflector 10 was at 48.2 cm. During reflector 8 calibration, coarse reflector 5 was at 47.8 cm. All other coarse reflectors (excepting 5 and 10 as mentioned above) were in full up position.

Preliminary evaluations showed that the total worth of reflector 3 was \$1.45 while that of reflector 8 was \$1.42.

All coarse reflector worths were measured by "rod-swapping" with calibrated reflectors 3 and 8. The values obtained are given in subsection 6.4.

9.1.6.2 Material Worth Measurements (Core II-A)

In Core II-A, material (reactivity) worth measurements of a standard fuel rod, a B_4C poison rod and a UO_2 rod were made at five different core locations. In addition, the material worth measurements of guinea pig fuel rods were made at two thru-head core locations.

9.1.6.3 Activation Measurements (Core II-B)

The spatial distributions of the fission rates for the isotopes Pu-239, U-238 and U-235 were measured. This was done by irradiating the foils of the above three isotopes placed at selected axial positions in six special foil holder rods. These rods were placed at six core locations. The standard fuel rods that were in these core locations were reloaded into the empty thru-head location in order to gain enough reactivity to reach the reactor power level of 0.75 KW. The foils were irradiated 0.75 KW for 1 hour. The fission product activities of these foils have been counted and data analysis is in progress.

9.1.6.4 Reflector Calibrations (Core II-C)

After installing the reactor vessel head, the thru-head ports were reloaded with specified fuel and B_4C poison rods (Core II-C). With this core configuration fine reflectors 3 and 8 were recalibrated differentially by "period" method. Initial evaluation of total worth for reflector 3 gave a value of \$1.33 and \$1.37 for reflector 8.

During these calibrations, coarse reflector 5 was in the full down position, while reflector 10 was at 15 to 18 cm.

9.1.7 IFA Installation

During the planned plant shutdown, four additional instrumented fuel assemblies were installed. On August 4, VNC No. 24 was installed at core location W5.0S0.0. VNC No. 27 was installed at W0.5N1.0. The following day VNC No. 26 and 27 were installed at core locations E3.0S0.0 and W0.5N1.0 respectively.

Following the installation of four new IFA's, before Core I super-prompt transients, a systematic initial surveillance of all sensors in the IFA's was initiated. This initial surveillance consisted of obtaining Time Domain Reflectometer (TDR) measurements, resistance measurements, and EMF measurements. The reactor conditions for these measurements were as follows:

TDR measurements were obtained at primary sodium temperatures of 400°F, 600°F, and 700°F and also at reactor power levels of 600 watts and 7 MW. The IFA flowmeters were calibrated by varying the main primary flow from 0 thru 5000 gpm. Fuel and coolant thermocouples responses were obtained for primary temperatures of 400°F to 700°F and reactor power levels of 600 watts, 2 MW, 4 MW, 7 MW, and 9 MW. These "response" measurements consisted of resistance values of the sensors leads, EMF values and "static" test outputs on the DAS. On the whole, all of these additional IFA's responded as predicted excepting the following:

VNC-24: inlet coolant T/C appeared erratic (A-B)

VNC-27: Fuel T/C (pin pair R-S) appeared open

VNC-27: Fuel T/C (pin pair V-W) appeared erratic

VNC-27: Flowmeter showed considerably lesser values than expected

VNC-28: inlet coolant T/C appeared open (A-B)

The DAS output gave the same indications excepting the following:

VNC-27: Fuel T/C (pin pair R-S) good response

VNC-26: Fuel T/C (pin pair N-P) reads erratic

During the 10 MW run for the capture-to-fission ratio irradiation VNC-27, fuel T/C (pin pair T-U) read high.

Further TDR and resistance measurement correlation taken after Core II stainless steel rods were installed showed the same behavior as before.

Some thermocouples continue to function on the two IFA's which were installed prior to Core I power ascension. The flowmeters on both of these IFA's have failed.

Following the installation of IFA's, the reactor was heated to 760°F and brought to critical power level of 600 watts. By comparing the critical reflector positions, it was concluded that a reactivity equivalent of 39¢ was lost due to the loading of IFA's. Thus, the projected available excess reactivity of 700°F inlet temperature and 20 MW was -2.0¢

9.1.8 Core II Test Procedures

Core II Test Procedures approved and issued were:

- I-A.1 Replacement of BeO Tightening Rods with Steel Tightening Rods
- I-A.2 Core II Reactor Noise and Sub-Critical Multiplication Measurements
- I-B.1 Material Worth Measurements (at 350°F)
- I-B.2 Reflector Worth Measurements
- I-B.7 Activation Measurements
- I-C.1 Oscillator Rod Calibration
- I-C.2 Zero Power Transfer Function
- I-C.3 Isothermal Temperature Coefficient
- I-C.4 Flow Coefficient
- I-D.3 Reflector Worth Measurements (at 550°F and 760°F)
- II-2 Conventional Oscillator Tests
- II-4 Static Tests
- VII-1 Core I Noise Measurements
- IX-1 Alpha Measurements

9.2 OPERATIONS AND MAINTENANCE

Reactor operation statistics for the quarter are shown below:

	Current Month 9/16-10/15/71	Current Quarter 7/14-10/15/71	Total To Date 10/15/71
Hours Critical (h)	209	473	3,414
Power Generated (MWH)	536	2,514	22,899
Startups	46	58	545
Scrams (See Table 9-1)	5	16	159 (2)
Operating Efficiency ⁽⁵⁻¹⁾ (%)	93.4	73.0	42.5

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

Thirty-two scrams have been intentional, for FRED or other tests.

The 1971 summer outage to install the outer head backup seal, emergency crane retrieval and other maintenance work was terminated August 2, 1971 after refueling cell purity was attained for the inner head removal for instrumented fuel assembly (IFA) installation. The remaining four IFA's were installed in the core by August 8, 1971. The reactor inner head was replaced, the thru-head fuel was returned to the core and preparations for operation completed. During the installation of the IFA's, a total of 25 manned entries were made using the man access suits. A total of 21 hours were spent in the refueling cell in the man access suits for the IFA installation.

After the IFA installation was completed, the vessel head was replaced and extensive checkout and pre-operational calibrations on the IFA's were completed. On August 14, 1971 the first transient test after IFA installation was accomplished. This first transient test was a sub-prompt transient test to compare the B₄C shielded transient detectors with earlier performance when the transient detectors were shielded with lead shot. The sequence of super-prompt transient tests were completed on August 30, 1971.

During the completion of the super-prompt transient tests, it was noted that NaK had been forced into the refueling cell from the NaK bubbler on the refueling cell purification loop. It was surmised to be from the separator. The separator was drained, the spilled refueling cell NaK reacted, the refueling cell purged to air and cleaned and the separator removed from the system and cleaned. The separator was returned to the purification loop. In-cell maintenance was accomplished during the brief outage. When refueling cell purity was again attained, the capture-to-fission ratio (CTFR) experiment was inserted into the reactor and irradiated for 90 MW days. The CTFR experiment was removed from the reactor and shipped to Vallecitos Nuclear Center (VNC) with two guinea pig rods slated for destructive examination. During the CTFR irradiation, special tests on the Air Blast Coolers and Nitrogen Cooling System were performed to assess their capabilities for anticipated follow-on testing.

After testing by ORNL personnel on Core I was completed, the reactor head was removed, change over from Core I to Core II completed and Core II zero power testing initiated. Testing continued uninterrupted until the reactor vessel head was re-installed at which time excessive cover gas leakage was noted. A brief shutdown, while the sodium radiation levels were low, was planned to purge the nitrogen zones to air and allow access by BRD designers to study potential equipment locations for follow-on testing. At this time, it was positively established that the cover gas leak was in the refueling cell. The refueling cell was purged to air and the leak was found to be on the outer head backup seal. The "o" ring gaskets in the backup seal will be replaced.

The inner containment leak test was repeated in July, following the crane emergency retrieval system installation. The leak rate was 4.4 percent per day compared with the 9.0 percent rate measured in June during the annual leak test. The improvement was effected by repairing known penetration leaks and rewelding some defective welds which were found in the nitrogen zone liner plate.

The eleven fuel rods which were returned to NFS in June were returned to GE without repair. Helium leak tests at NFS of the seven room temperature leakers produced negative results. NFS refused to repair the three rods having short upper plenums and the one rod (Serial No. 004) that their tests showed to be a leaker because the rods had been heated to 700°F at SEFOR during the hot-leak checks. The eleven fuel rods were again leak checked at SEFOR at room temperature. Two rods, 004 and 552, were determined to be leakers. The rods will be hot-leak checked (700°F) at a later date.

A total of 150 equipment malfunctions were corrected and distributed as follows:

Mechanical	49
Electrical	55
Instrumentation	46
Total	150

The more significant malfunctions included:

Nitrogen Blower 8064 control switch. SBM switch contacts 4-4T failed to function in the "after trip following pull-to-lock" action.

Reflector 8 Decitrak was damaged due to wires being caught in the movement of the reflector.

Line-to-Storage Tank in the Liquid Radwaste System Plugged.

Freon Leaks in Unit 8118

Broken camroller in Main Air Lock Door.

NaK Bubbler Separator Plugged.

Freon Unit 8116B Compressor excessive oil usage.

Leak Detector 33C-1 Failed.

Scram Solenoid Valve 8B Indication Switch Failed.

Reflector Raise Solenoid Valves 541-7A or 541-7B leaking causing reflector 7 to creep upward.

Half-ton Hoist Grapple failed due to the wire rope, electrical cable pulling out of the grapple. This was caused by a jam nut coming loose.

Argon compressor gear failed due to broken teeth.

A total of 38 plant and equipment modifications were completed, including the following significant items:

Electrical

- Instrument air header pressure alarm
- Man access suit flowmeter (Alarm and control Wiring)
- Argon Reservoir Pressure Control Override Switch
- WRM Picoammeter protection super-prompt excursion test
- IFA junction panel modification
- Improved maintainability — Grapple System
- Programmed flow reduction for super-prompt transients
- Scram contactor time signal to DAS
- Main and auxiliary air blast cooler instrumentation installation
- Auxiliary Primary Loop Leak Detector Substitution 33C-1A for 33C-1
- Duct Heater Control — office building air-conditioning system
- Emergency battery voltmeter
- Lighting above new fuel storage vault

M.P.P. Programmed Flow Reduction time delay and rate following scram
Restoration of WRM 3 Output Circuit to Pre-modification 652 Configuration
Sound power phone system improvements

Mechanical

Reactor cover gas meter
Vertical transfer mechanism
Instrument air header pressure switch
Reactor vessel head seal (backup)
L.R.W. storage tank
Hinged covers for radwaste vaults
Vertical transfer port platform
Extension rod conversion
Crane bay marine hatch test panel
Refueling cell argon purifier separation valve
Swing yoke adapter
Swing yoke
Core II tightener storage rack
Reactor head lifing fixture
LRW purge line
LRW vault fire protection
GRW decay tank valves
Penetration leak check manifold
Safety bridge
Airlock pressure test air supply
Manipulator slave arm lifting bail
Argon compressor seal oil seal relief valve

Gamma ray spectrometry measurements of fission gas components of the reactor cover gas were made routinely during the transient testing. The results continue to indicate levels of xenon and krypton commensurate with tramp uranium in the system.

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

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