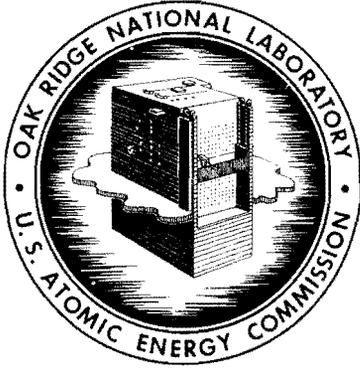


MASTER



## OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION



ORNL-TM-173

*Seif*  
*96*

### SUMMARY OF HRT RUN 25

J. R. Engel  
H. F. Bauman  
J. R. Buchanan  
P. N. Haubenreich  
H. B. Piper  
D. M. Richardson

#### NOTICE

This document contains information of a preliminary nature and was prepared primarily for internal use at the Oak Ridge National Laboratory. It is subject to revision or correction and therefore does not represent a final report. The information is not to be abstracted, reprinted or otherwise given public dissemination without the approval of the ORNL patent branch, Legal and Information Control Department.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

- A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Contract No. W-7405-eng-26

Reactor Division

SUMMARY OF HRT RUN 25

J. R. Engel, H. F. Bauman, J. R. Buchanan  
P. N. Haubenreich, H. B. Piper and D. M. Richardson

Date Issued

**JUL 25 1962**

---

OAK RIDGE NATIONAL LABORATORY  
Oak Ridge, Tennessee  
operated by  
UNION CARBIDE CORPORATION  
for the  
U. S. ATOMIC ENERGY COMMISSION



CONTENTS

|  | <u>Page</u> |
|--|-------------|
| Abstract . . . . .   | vii         |
| Introduction . . . . .                                     | 1           |
| Operations . . . . .                                       | 1           |
| Preliminary Operations . . . . .                           | 1           |
| Hydrostatic Test of Fuel Low-Pressure System . . . . .     | 2           |
| Leak Test of Reactor Shield . . . . .                      | 2           |
| Hydrostatic Test of High-Pressure Systems . . . . .        | 2           |
| Pretreatment and Freeze Plug Checks . . . . .              | 2           |
| Operating Conditions . . . . .                             | 3           |
| Power Operation . . . . .                                  | 5           |
| Establishment of Conditions for 5-Mw Operation . . . . .   | 5           |
| 5-Mw Operation at 1400 psig . . . . .                      | 5           |
| Change in Core-to-Blanket Mixing . . . . .                 | 6           |
| Operation at 1400 psig with High Blanket Concentration . . | 7           |
| Operation at 1700 psig . . . . .                           | 9           |
| Operation at 1250 psig . . . . .                           | 9           |
| Post-Operation Examinations . . . . .                      | 9           |
| Core Examination . . . . .                                 | 9           |
| Corrosion-Specimen Recovery . . . . .                      | 10          |
| Reactor System Storage . . . . .                           | 10          |
| Fuel Recovery . . . . .                                    | 10          |
| Heavy-Water Recovery . . . . .                             | 11          |
| Auxiliary Systems . . . . .                                | 11          |
| Reactor Steam System . . . . .                             | 11          |
| Changes . . . . .  | 11          |

## CONTENTS (Con't.)

|   | <u>Page</u> |
|---|-------------|
| Operation . . . . .                             | 12          |
| Chemistry . . . . .                             | 12          |
| Oxygen and Off-gas . . . . .                    | 13          |
| Components . . . . .                            | 13          |
| Fuel Feed Pump . . . . .                        | 13          |
| Samplers . . . . .                              | 14          |
| Valves . . . . .                                | 15          |
| Freeze Jackets . . . . .                        | 17          |
| Miscellaneous . . . . .                         | 18          |
| Concentration Ratios . . . . .                  | 18          |
| Critical Concentrations . . . . .               | 21          |
| Core Patch Failure . . . . .                    | 23          |
| Containment . . . . .                           | 27          |
| Cell Leakage . . . . .                          | 27          |
| Flange Leakage . . . . .                        | 28          |
| Stack Filter . . . . .                          | 28          |
| Fuel Solution Chemistry and Corrosion . . . . . | 29          |
| Fuel Composition . . . . .                      | 29          |
| Fuel Stability . . . . .                        | 31          |
| Behavior of NAT . . . . .                       | 31          |
| Other Symptoms . . . . .                        | 33          |
| Probable Explanation of NAT Rise . . . . .      | 33          |
| Precipitation in Blanket Samples . . . . .      | 38          |
| Corrosion . . . . .                             | 39          |
| Appendix . . . . .                              | 41          |

## LIST OF FIGURES

| <u>No.</u> | <u>Title</u>  | <u>Page</u> |
|------------|---|-------------|
| 1          | Chart of Operations for Run 25.   | 4           |
| 2          | Effect of Concentration Ratio on Power Distribution in Run 25.  | 9           |
| 3          | Effect of Net Blanket Purge Rate on Concentration Ratio in Run 25.  | 20          |
| 4          | Ratio of Critical Concentrations Predicted by GNU Calculations to Those Observed in Runs 24 and 25 vs Blanket-Core Concentration Ratio. | 22          |
| 5          | Upper Patch and Bolt Fragment as Found in the Core After Run 25.  | 24          |
| 6          | Remnant of Zircaloy-2 Bolt Sticking Through Upper Patch.  | 25          |
| 7          | Scale Model of Upper-Hole Patch Assembly.   | 25          |
| 8          | Last Power Operation Before Increase in Core-Blanket Mixing.  | 26          |
| 9          | HRT Physical Inventories in Run 25.   | 30          |
| 10         | System Conditions and Behavior During an Experiment Resulting in NAT Rise.  | 34          |
| 11         | Illustration of Reversible Power Effect on NAT.   | 35          |
| 12         | Physical Inventories vs Total Power in Run 25.  | 36          |
| 13         | Corrosion Rate of Stainless Steel vs Acid Level in HRT Runs 17 Through 25.  | 40          |

## LIST OF TABLES

| <u>No.</u> | <u>Title</u>   | <u>Page</u> |
|------------|--|-------------|
| 1          | Sample Volumes in Run 25.  | 16          |
| 2          | Some Fuel Solution Parameters During Operation at 5 Mw<br>in Run 25. | 32          |

## ABSTRACT

Run 25 was the final period of power operation of the HRT. The reactor was operated for periods of 62, 8, 52, and 80 hours at 5 Mw with no outward indication of fuel solution instability at a system pressure of 1400 psig and core and blanket average temperatures of 270 and 230°C, respectively. The uranium concentration in the blanket was 1.7 to 2.0 g U/kg D<sub>2</sub>O. Longer periods of operation were prevented by mechanical difficulties, notably with the fuel feed pump.

While the reactor was subcritical after the last of the above runs, the upper patch in the core-tank wall became dislodged, allowing greater core-to-blanket mixing. The resultant blanket uranium concentration was 2.9 g U/kg D<sub>2</sub>O. The reactor was subsequently operated at pressures of 1250, 1400 and 1700 psig with blanket temperatures of 230 and 240°C. Fuel instability was apparent at all conditions at 5 Mw.

The reactor was shut down on April 28, 1961. The experiment was operated at high temperature for a total of 10,866 hours. The system was critical for a total of 8,841 hours and produced 16,295 Mw-hours of power. The fuel, heavy water, and some corrosion specimens were recovered, and the reactor was stored in an assembled state.



## INTRODUCTION

Run 25 was the final period of power operation of the HRT. The objectives and program for this run were essentially the same as for other runs with reverse flow in the core: to operate at 5 Mw for extended periods, if possible, and to continue the study of the fuel instability problem. Operations in runs 23 and 24<sup>1</sup> suggested that the fuel was stable at 5 Mw for at least one set of conditions (1400 psig system pressure, 260°C core temperature and 230°C blanket temperature). However, it had not been possible to operate the system for long periods of time at these conditions because of frequent mechanical difficulties.

All the reactor components which had caused difficulty were replaced during the shutdown which preceded run 25. However, some of the replacement components were used parts which had been rebuilt. This run was started with the objective of operating at 5 Mw for a long time to demonstrate the stability of the fuel solution. In addition, experiments were planned to study the behavior of xenon in the reactor system.

It had been decided, prior to the start of run 25, that the HRT would be shut down by May 1, 1961. This decision had some influence on the experimental program during the latter part of the run, as several short tests were made to provide as much information as possible before the final shutdown.

This report is the last in a series describing the power operation of the HRT. The other reports in the series are listed in the Appendix.

## OPERATIONS

### Preliminary Operations

The in-cell maintenance operations which preceded run 25 were completed on March 14, 1961. These operations included the installation of two freeze jackets on the fuel feed line and the replacement of the fuel letdown heat exchanger, three high-pressure valves and two remote diaphragm pump heads.<sup>2</sup> With the maintenance work complete, a period of preliminary operations was started in preparation for the next power operation. The items accomplished in this period were hydrostatic tests of the high- and low-pressure fuel systems, pressure and leak tests of the shield, pretreatment of the stainless steel surfaces in the high-pressure systems and a checkout of the newly installed freeze jackets on the fuel feed line.

---

<sup>1</sup>H. F. Bauman et al., Summary of HRT Runs 22, 23 and 24, ORNL-TM-106 (Mar. 6, 1962).

<sup>2</sup>Ibid.

### Hydrostatic Test of Fuel Low-Pressure System

The first operation was a hydrostatic test of the fuel low-pressure system, because the shield could not be completely sealed until this was completed. It was desired to observe the cracked tee in the fuel feed line under pressure to determine the effectiveness of the repair. The Omniscope was inserted through the top of the shield and the opening was sealed with masking tape so that a slight negative pressure (5 in. H<sub>2</sub>O) could be maintained in the cell while the test was in progress. The fuel low-pressure system was first pressurized to 500 psig with one of the freeze plugs which isolated the leaking tee (FP-159) unfrozen. Under this condition a leak of about 3 cc/min was observed. This test proved that the tee was adequately supported to prevent complete rupture under high pressure, even without the protection of the freeze plugs. Then the tee was completely isolated with freeze plugs and no leak was observed with the pressure at 500 psig for one hour.

### Leak Test of Reactor Shield

After the viewing equipment was removed, the work of sealing the shield was started. Some difficulty was encountered (see pages 27, 28) and a satisfactory condition was finally achieved on March 23. The rate of inleakage was measured to be 3.1 liters/min with the shield evacuated to one-half atmosphere. Leakage through the top of the shield was 170 cc/min with an internal pressure of 15 psig.

### Hydrostatic Test of High-Pressure Systems

The hydrostatic test of the high-pressure systems was carried out while the pan-sealing operation was in progress. The systems were satisfactorily tested at 2675 psig, but it was found that the blanket dump valve (PCV-252) failed to open on a dump signal (see page 15). It was necessary to reopen the shield above the valve and reconnect one of the valve operator air lines. With this correction, all the pressure-associated safety interlocks on the high-pressure system were checked to work properly.

### Pretreatment and Freeze Plug Checks

The pretreatment of the reactor high-pressure systems was undertaken after a satisfactory shield leak rate had again been established. It was necessary to form a protective oxide film on the new surfaces in the fuel letdown heat exchanger and to fortify the film in the rest of the system. This was accomplished by circulating a subcritical concentration of fuel solution at 270°C for about 50 hours. After about 24 hours of the pretreatment, there was evidence (from thermocouples on the jackets) that the freeze plugs around the leaking tee in line 107 had thawed. The pretreatment was continued while efforts were made to re-establish the freeze plugs (see pages 17, 18). After the completion of the pretreatment, four more days were spent in trying to freeze the plugs around the leak and prove that they were frozen. It was finally

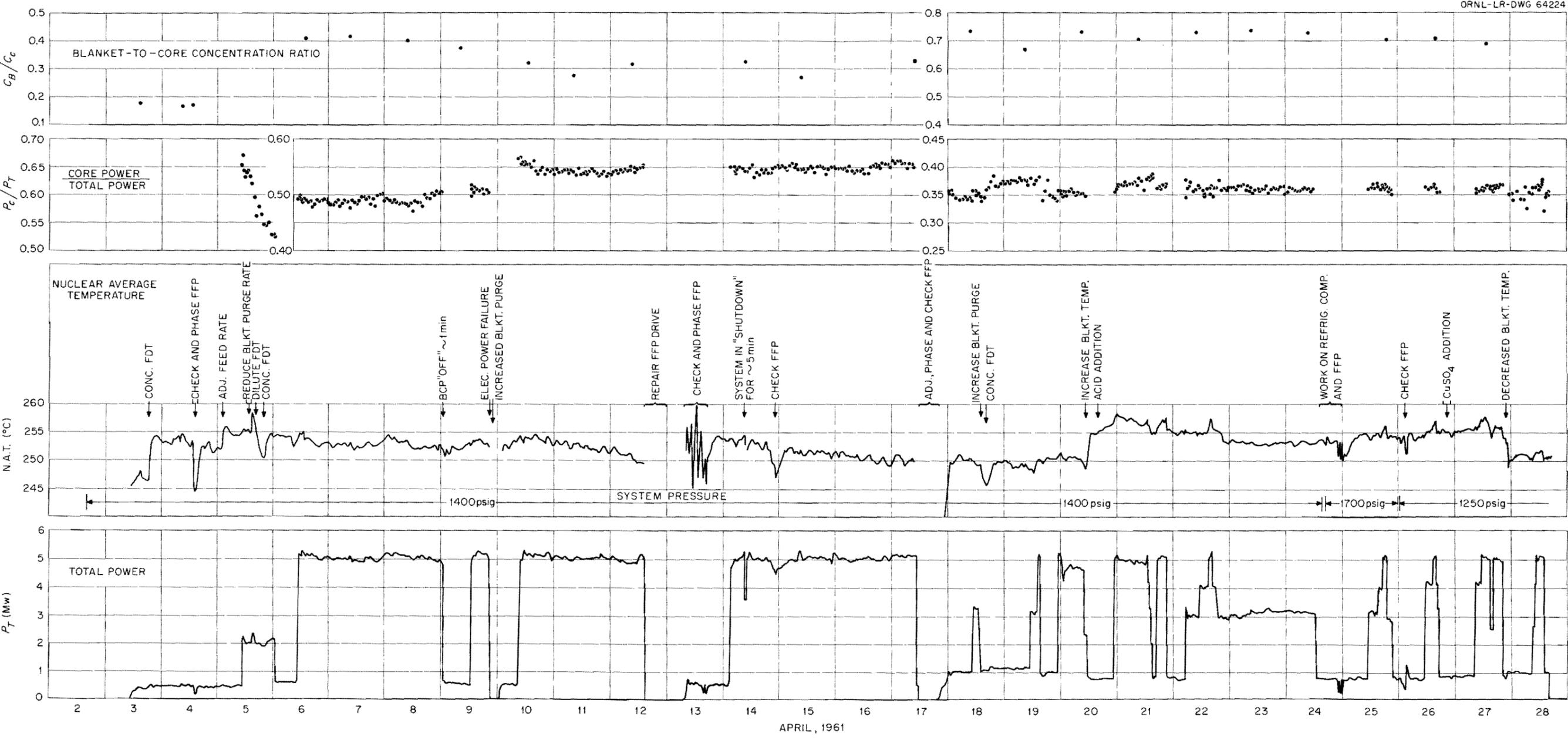


Fig. 1. Chart of Operations for Run 25.

concluded that the only means of proving the existence of the ice plugs was to operate the reactor at power and observe the fission-product activity in the cell air.

A reactor startup for power operation was begun on March 30, 1961, but the achievement of criticality was delayed by two failures in the reactor steam system on consecutive days (see page 12 ). The reactor was made critical on April 3, 1961.

#### Operating Conditions

The operating conditions in run 25 were dictated by the experimental program and the events which occurred during the run. Most of the changes were made after the upper core-tank patch failed (see page 6 ), causing the blanket concentration to increase. The core average temperature throughout the run was maintained between 265 and 270°C, and the blanket average temperature was held at 230°C except for a short time when it was raised to 240°C. The pressure was held at 1400 psig except for some experiments near the end of the run at 1250 and 1700 psig.

This run was started with a nominal blanket uranium concentration of 1.0 g U/kg D<sub>2</sub>O. However, soon after the start of the run, the blanket purge rate was adjusted to give a nominal blanket uranium concentration of 1.5 g U/kg D<sub>2</sub>O, and this concentration was maintained until the patch failed. After that, higher blanket concentrations prevailed.

The "Chart of Operations" (Fig. 1) shows some of the more important system parameters for the operating portion of run 25. The nuclear power, system pressure, nuclear average temperature, core-to-total power ratio and blanket-to-core concentration ratio are plotted as functions of time. Both the total power and the power ratio were calculated from the temperature rise of the primary circulating streams. Since this method has limited accuracy at lower powers, the power ratio is shown only for the periods when the total power was greater than the system heat losses. The nuclear average temperature (NAT) is a weighted average of the core and blanket inlet and outlet temperatures with the weighting factors chosen so that the NAT varies in the same way as the critical temperature. The concentration ratios which are plotted are the averages of the uranium and copper analytical results. For the periods between samples, the power-ratio plot shows the general trend (inverted) of the concentration ratio. Particular aspects of the Chart of Operations are discussed in various sections throughout this report.

The term "heat-loss power" is used extensively in this report. This quantity is a variable which depends upon the system conditions. Heat-loss power, as used in this report, is determined by the core power level which is required to make up for all heat losses from the core loop. These include heat transfer to the blanket and heat removal by the fuel heat-exchanger blowdown and steam leakage as well as the usual heat losses through pipe walls. The power generated in the blanket contributes to the reactor power level and is determined by the blanket-to-core concentration ratio and the core power. In run 25, when the core-heat-loss

condition was established, the blanket power was greater than the heat losses, and it was necessary to remove steam from that heat exchanger in order to maintain the desired blanket loop temperature. Thus, the total reactor power at "heat-loss" conditions was greater than the conventional heat losses. During the first part of run 25, heat-loss power was about 400 kw--only slightly higher than the conventional heat losses. However, with the higher blanket uranium concentration that prevailed after the core-patch failure, the fraction of power generated in the blanket increased so that heat-loss power was nearly 1 Mw.

### Power Operation

#### Establishment of Conditions for 5-Mw Operation

The reactor was made critical and raised to heat-loss power (about 400 kw) on April 3, 1961. At this level the fuel-dump-tank weight was adjusted to give the desired conditions: core average temperature, 270°C and blanket average temperature 230°C. The fuel feed rate was 7 lb/min and the blanket purge rate was 6 lb/min, with both remote heads of each feed pump in service. These conditions were maintained for 48 hours, and three pairs of high-pressure samples were obtained. In addition, a sample of the reactor off-gas (about 150 liters) was isolated in the low-pressure system of the chemical plant to provide a base line for xenon studies to be made later in the run.<sup>3</sup> While at heat-loss power some difficulty was encountered with the fuel feed pump (see page 13); the east head was taken out of service and operation was continued using only the west head.

On April 5 the power was raised to 2 Mw. At this level the core-to-total power ratio indicated a blanket uranium concentration of 1.1 g U/kg D<sub>2</sub>O, which was confirmed by the results of the three samples at heat-loss power. It was decided, however, that this concentration did not provide adequate insurance against solution instability in the blanket region, and a minimum blanket uranium concentration of 1.5 g U/kg D<sub>2</sub>O was prescribed. The blanket purge rate was reduced to allow this concentration to build up, and the fuel-dump-tank weight was adjusted to hold the core average temperature at 270°C. After 14 hours at 2 Mw the power was reduced to heat-loss to allow all conditions to stabilize in preparation for operation at 5 Mw.

#### 5-Mw Operation at 1400 psig

The power was raised to 5 Mw on April 6, and the reactor operated stably at this level for 62 hours. At that time the operation was interrupted by a brief stoppage of the blanket circulating pump. The pump was stopped by action of the overcurrent relay, but no cause for the action could be found and the current was normal when the pump was

---

<sup>3</sup>W. D. Burch and O. O. Yarbro, Homogeneous Reactor Program Progress Report for Period From December 1, 1960 to May 31, 1961, ORNL-3167, p 11.

restarted. Because of the upset in system conditions caused by the pump stoppage, the reactor was held at heat-loss power for 12 hours to allow conditions to stabilize again. Operation at 5 Mw was then resumed and continued for about 8 hours until an electric power failure in the building (during a storm) caused an interruption. Throughout this period of operation, the core-to-total power ratio indicated a blanket uranium concentration considerably greater than the minimum required. Results from four pairs of high-pressure samples showed an average of 2.04 g U/kg D<sub>2</sub>O.

As a result of the power outage, the system was subcritical for 4 hours and then operated 8 hours at heat-loss power. During this time the blanket purge rate was increased to lower the blanket uranium concentration, and the fuel-dump-tank weight was adjusted for core-temperature control. Another sample of reactor off-gas was isolated in the chemical plant while at heat-loss power. The power was again raised to 5 Mw on April 10 and maintained for 52 hours. The operation, in general, was very steady, but the last 6 hours was characterized by a steady drop in nuclear average temperature, which finally made it necessary to reduce the power. There was evidence that the temperature drop was a result of poor fuel-feed-pump performance, but efforts to improve the pumping rate, while the reactor was at power, were unsuccessful. The reactor was made subcritical on April 12, and the fuel feed pump was shut down for maintenance of the drive unit (see page 14). Circulation in the high-pressure systems was continued and the temperature was maintained with steam supplied by the package boiler.

The reactor was subcritical for 18 hours for the fuel-feed-pump repair, then was operated at heat-loss power for 18 hours to allow steady conditions to be established. Both heads of the pump were tried and operation was continued with only the west head in service even though some check-valve leakage was apparent. The power was raised to 5 Mw on April 14 and held for a total of 80 hours with only one minor interruption. This occurred after 5 hours at 5 Mw when an adjustment of the control action of the blanket level controller caused a disturbance which resulted in a decrease in NAT. In order to avoid the possibility of radiolytic-gas bubble formation in the core, the power was reduced to 3.5 Mw for 45 minutes until the temperature began to recover. About 16 hours later the NAT began to drop because of a decrease in the fuel feed rate. The normal rate was restored and the temperature recovered without interrupting the power operation.

At no time during this period of operation, or the one preceding it, was there any observable evidence of fuel-solution instability, either from the system behavior or from the fuel samples. The average concentration of uranium in the blanket from six samples was 1.7 g U/kg D<sub>2</sub>O.

#### Change in Core-to-Blanket Mixing

On April 17 a rapid decrease in NAT began and could not be checked even though it was evident that the cause was reduced output of the fuel

feed pump (see page 23). The power was reduced<sup>4</sup> and the reactor was made subcritical at 1120 to investigate the pump behavior. Measurement of the fuel feed rate at 1145, after rephasing the pump and increasing the stroke, gave a value of only 5.8 lb/min instead of the 7 lb/min which had existed at steady reactor operation. Both heads of the fuel feed pump were put in service in an effort to raise the feed rate to 7 lb/min. However, the intermediate system of the east head was found to be leaking inside the cell and this head was taken out of service. The fuel-dump-tank weight was then lowered to permit operation at normal temperature with a fuel feed rate of 4 lb/min. The rate was adjusted to this value, but, after a short time, it began to increase spontaneously. Finally (about 2300) it was possible to re-establish the 7 lb/min feed rate with a shorter pump stroke than had been required before the pump faltered.

The reactor was made critical while the fuel feed rate was still increasing, but the critical temperature was much lower than would be expected for the existing conditions of feed and purge rate and dump-tank weight. It was also noted that heat-loss power was nearly 1 Mw and that the core-to-total power ratio was about 0.35. The values for heat-loss power and power ratio prior to the interruption were about 0.5 Mw and 0.55, respectively. Both of these facts indicated a large increase in the mixing between the core and blanket. It was found later (see pages 23-27) that the increased mixing was caused by the failure of one of the core patches. On April 18 the power was raised to 3 Mw for about 2 hours to show that the power ratios measured at 1 Mw were accurate. Further evidence of the increased mixing was obtained from sample results which showed a blanket uranium concentration of 2.9 g U/kg D<sub>2</sub>O, 0.74 of that in the core.

#### Operation at 1400 psig with High Blanket Concentration

After the increased mixing had been verified, the reactor was operated at heat-loss power for about 20 hours. During this time the blanket purge rate was raised to 6 lb/min (from 4 lb/min) to reduce the blanket concentration as much as possible. However, it was found that the maximum purge which could be maintained was 5 lb/min and the system was allowed to reach steady state with that purge. The core-to-total power ratio leveled out at 0.35, indicating a blanket-to-core concentration ratio of 0.7. The blanket average temperature was held at 230°C and the core average temperature was brought to 270°C by lowering the fuel-dump-tank weight to 350 lb.

Two questions were raised at this point about the operability of the reactor at 5 Mw under the above conditions. One of these was concerned with the possibility of radiolytic-gas bubble formation in the blanket. Because of the lack of information about the flow pattern and temperature distribution within the blanket vessel, it was not possible to calculate, with any degree of certainty, the threshold power for bubble

---

<sup>4</sup>Just before the power was reduced, a sample of reactor off-gas was isolated in the chemical plant.

formation. It appeared, however, that this threshold might be exceeded at 5 Mw if the blanket average temperature were held to 230°C. The other question was concerned with the ability to pass all the steam produced in the blanket heat exchanger through the blanket steam-withdrawal valves. Since these valves were sized for a significantly smaller steam flow, it was calculated that their capacity would be exceeded at full power with 65% of the heat generation in the blanket.

The reactor power was raised to 5 Mw for a short time on April 19 to obtain direct answers to the above questions. There was no evidence of bubble formation, and, although the blanket steam valves were nearly full-open, there was no blanket temperature rise to indicate that the valve capacity had been exceeded. The power was then reduced to heat-loss to allow more time for the establishment of steady conditions. After about 7 hours the power was raised to 5 Mw for extended operation. However, the blanket-steam-valve capacity was exceeded this time and it was necessary to lower the power to 4.6 Mw to keep the blanket temperature from rising. The reactor was operated at this level for 9 hours; then the power was lowered to heat loss and the blanket average temperature was raised to 240°C.

The reasons for increasing the blanket temperature were, first, to increase the density of the steam produced in the blanket heat exchanger so that the reactor could be operated at full power and, second, to reduce the likelihood of radiolytic-gas bubble formation in the blanket. While the system was at heat-loss power for this change, 14 moles of H<sub>2</sub>SO<sub>4</sub> was added. This addition raised the acid concentration to the highest level in the entire operation of the HRT, for the purpose of studying the effect on fuel stability (see page 29 et seq.).

After steady-state conditions had been attained at heat-loss power, the power was raised to 5 Mw and held there for 14 hours. Shortly after the power was raised, the NAT rose about 2°C and remained above the expected temperature by this amount until the power was reduced. The NAT dropped back to the normal level when the power was lowered. After a short time (approximately 2 hours) at heat-loss power, the power was again raised to 5 Mw. This time the NAT rose and continued to rise. After 5 hours the NAT rise (3.2°C) had exceeded the limit set for safe operation and the power was reduced to heat-loss for re-establishment of steady conditions. The NAT dropped 2°C within 30 minutes of the power decrease. A third attempt was made to operate at 5 Mw with the same base conditions. This time the power was raised in steps with several hours of operation at 3 Mw and 4 Mw. Operation was normal until the power was raised from 4 to 5 Mw; then the NAT began to rise, even more steeply than on the other attempts. The power was reduced to 3 Mw to restore the normal temperature, and the reactor operated steadily and, apparently, stably at that power for 42 hours. At the end of that time an off-gas sample was isolated and the power was reduced to heat-loss so that the system conditions could be changed.

### Operation at 1700 psig

One theory that was advanced to explain the observed instability was that, because of the unprecedented high blanket power, boiling was occurring in the blanket, resulting in fuel deposition on the core-tank wall. It was suggested that this condition might be alleviated by operating at higher system pressures. Therefore, the pressure was raised to 1700 psig and steady conditions were established at heat-loss power with the core at 270°C and the blanket at 240°C. The reactor power was raised to 3 Mw and then to 4 Mw with no indications of fuel instability. However, a rapid NAT rise ensued when the power was raised to 5 Mw. Normal temperatures were restored by lowering the power to 3 Mw.

### Operation at 1250 psig

For the next series of experiments, the system pressure was set at 1250 psig. Power operation at this pressure was stable at 4 Mw, but was accompanied by a rapid NAT rise at 5 Mw. To eliminate the possibility that excessive radiolytic-gas concentrations in the solution were responsible for the fuel instability, an addition of 13 moles of  $\text{CuSO}_4$  was made to the fuel solution. This approximately doubled the concentration of recombination catalyst in the solution and significantly reduced the radiolytic-gas partial pressure at power. Two more attempts were then made to operate at 5 Mw with the same results that had been observed previously. Following this experience, the blanket temperature was again reduced to 230°C to improve the cooling of the core-tank wall. Fuel instability in the form of a NAT rise was once more observed at 5 Mw. This experiment ended the experimental work and the reactor was shut down on April 28, 1961, in accordance with the pre-established schedule of final shutdown by May 1. (The shutdown date was set by factors other than the completion of all useful experimental work.) The reactor was critical for 576 hours and produced 1,681 Mw-hours of heat in run 25. The total operating time above 200°C for the reactor experiment was 10,866 hours. The system was critical for 8,841 hours and produced 16,295 Mw-hours of power.

### Post-Operation Examinations

Because run 25 was the final operation of the HRT, a number of special examinations have been made or planned since the shutdown.

### Core Examination

Since the cause of the increased mixing during run 25 was of major interest, an inspection of the inside of the core tank was made shortly after the shutdown. The results of this inspection, relative to the core-to-blanket mixing, are described on page 23. In addition, the inside surfaces of the core were photographed and wall-thickness measurements were made. A report of these results will be issued by the Metallurgy group.<sup>5</sup> Plans were also made for the removal of portions of the core tank for metallographic examination. This work will be completed in FY-1963.

---

<sup>5</sup>A. Taboada, personal communication.

### Corrosion-Specimen Recovery

Corrosion specimens had been placed in a number of strategic locations in the HRT piping system during the initial installation. Specimens were removed from various locations, and replaced with new ones, throughout the operation of the reactor. At the end of run 25, corrosion-specimen assemblies were recovered from the following locations:

- 1) shell side of the blanket heat exchanger,
- 2) shell side of the fuel heat exchanger,
- 3) suction of the blanket circulating pump,
- 4) suction of the fuel circulating pump,
- 5) inlet to the blanket heat exchanger,
- 6) inlet to the fuel heat exchanger, and
- 7) core access flange.

In addition, an assembly of canned, metallurgical specimens for radiation damage studies was removed from the blanket pressure vessel. Specimens were recovered only if they were expected to provide substantial amounts of information and could be easily removed. All the specimens listed above were removed at flanged connections which were subsequently closed to seal the system. Those specimens which required destructive operations for their recovery were left in place. The examination of the recovered specimens is incomplete at this writing.

### Reactor System Storage

The entire HRT system was stored in a standby condition. The primary system piping was left intact and all flanges, which were opened for the recovery of corrosion specimens, were resealed and leak-tested. Several components (feed pump heads and valves) which had been replaced in earlier operations, but which were still highly radioactive, were moved from the storage pool to the reactor cell for storage. The reactor-cell seal pans were welded in but the cell was not leak-tested. All the reactor-associated equipment, including auxiliary systems and instruments, was serviced and left in standby condition.

### Fuel Recovery

Three charges of fuel were used in the operation of the HRT. These were added at the start of runs 12,<sup>6</sup> 14<sup>7</sup> and 24;<sup>8</sup> new charges were made

---

<sup>6</sup>Run 12 was the initial critical experiment in the HRT.

<sup>7</sup>J. R. Engel et al., Summary of HRT Run 14, ORNL CF-59-6-96 (June 8, 1959).

<sup>8</sup>H. F. Bauman et al., op. cit.

because of the build-up of excessive concentrations of nickel in the fuel. When a new fuel charge was added, the spent fuel was isolated in the fuel storage tanks until it could be removed for recovery of the uranium.

The first fuel charge was removed from the reactor system on September 26, 1958,<sup>9</sup> and was subsequently processed in the Oak Ridge National Laboratory fuel processing pilot plant. The second charge was still in the fuel storage tanks at the end of run 25. Most of the heavy water was removed from the fuel, and the two charges were stored separately in the fuel and blanket storage tanks. The fuel in the storage tanks (containing about 15 kg of uranium) was processed for uranium recovery in the fission-product pilot plant at the Oak Ridge National Laboratory, beginning in January 1962.

#### Heavy-Water Recovery

The heavy water associated with the HRT fuel was removed from the system in August 1961. The water was separated from the fuel solution, by evaporation in the fuel dump tanks, and collected in the blanket dump tanks. About 250 lb of heavy water was left with the fuel to permit transfer of the separate charges to the storage tanks. The fuel dump tanks were rinsed to the fuel storage tanks with normal water.

The heavy water condensate in the blanket dump tanks was transferred, in batches, to the waste evaporator where it was re-evaporated. The condensate was then passed through a mixed-bed ion exchange column for final cleanup and collected in aluminum drums. The final product was completely free of soluble contaminants, but the tritium content was 11 millicuries/ml.

#### AUXILIARY SYSTEMS

The successful operation of the HRT is strongly dependent upon the satisfactory performance of auxiliary systems. These systems include process and service steam, cooling water, refrigeration, instrument air, AC and DC power, general instrumentation and oxygen and off-gas. For the most part, the operation of the auxiliary systems is completely routine and, therefore, not of particular interest in a report dealing with the reactor operation. The operation of the auxiliary systems, which are of unusual interest because of non-routine behavior or because of particularly close connection with the reactor, is described below.

#### Reactor Steam System

##### Changes

Several changes were made in the reactor steam system before and during run 25.

---

<sup>9</sup>R. H. Winget, Transfer of Initial Fuel from the HRT and Recovery of the Associated Heavy Water, ORNL CF-58-10-44 (Oct. 3, 1958).

Before operation, a new plug and liner were installed in the main fuel steam-block valve, HCV-537. Previously, this valve had occasionally failed to seat tightly, especially when cold. Following the installation, both the fuel and blanket steam block valves were tested hydrostatically at 800 psig; the leakage through the valve seats was measured to be 1.1 liters/min for the fuel valve and 0.6 liter/min for the blanket. This leakage rate was acceptable since the steam activity block-and-vent system provides a vent to the reactor cell for any reasonable leakage from the block valves.

A control valve was installed in the water to the deaerator vent condenser, to control the feedwater level in the deaerator surge tank by varying the rate of condensation of service steam which provides feedwater makeup. The level had previously been controlled manually.

During the reactor operation, on April 13, an improved deaerator-steam pressure-control system was installed. (The previous pressure controller had failed in run 22.) A new primary element was installed on the deaerator, with an indicator-controller on the steam panel.

The improved control of the deaerator pressure and level made possible a more precise control of the feedwater composition.

### Operation

The reactor steam system operated very well throughout run 25, except for two valve failures during the reactor startup.

On March 31, the reactor steam system was shut down to repair a steam leak at the flange of HV-317A, the manual shutoff valve for reactor steam to the turbine. The flange ring groove, which was slightly eroded, was lapped, and the valve was reinstalled with a new O-ring. The system was then tested with steam from the package boiler at 750 psig and found to be tight.

The reactor startup was resumed, but on the next day, April 1, it was found that the blanket feedwater valve LCV-546 would not open, and the steam system was shut down again. The valve stem was found to be broken about 1/4 in. above the valve plug. The stem was rethreaded and reinstalled with the same plug.

### Chemistry

The reactor steam system was operated with continuous addition of hydrazine to remove radiolytic oxygen and with buffered phosphate to control the corrosion of carbon steel. The concentration of chloride in the letdown was very low, rarely exceeding 0.1 ppm. The rate of addition of hydrazine was very generous at  $1.77 \times 10^{-5}$  parts  $N_2H_4$  per part steam produced. The concentration of oxygen in the reactor steam averaged less than 15 ppb.

As in runs 22, 23 and 24, the blanket heat exchanger was operated at a lower temperature than the fuel heat exchanger in order to maintain

the blanket average temperature below that of the core.<sup>10</sup> An accumulation of hydrazine in the blanket heat exchanger was again observed, and the concentration of hydrazine in the blanket heat exchanger blow-down was about seven times that in the blanket heat exchanger steam.

#### Oxygen and Off-gas

Oxygen was injected into the core and blanket high-pressure systems at 1.5 and 1.0 SLPM, respectively, throughout run 25. As in the past, the oxygen was supplied from cylinders with the required pressure being provided by a diaphragm-pump compressor. Although there were no interruptions or serious malfunctions, frequent maintenance of the oxygen compressor was required. Otherwise the oxygen addition system performed routinely during the run.

All three charcoal adsorber beds were in service, in parallel, throughout run 25 for off-gas holdup. There was no evidence of overheating in the beds or of activity breakthrough. On four separate occasions, large samples (about 150 liters) of reactor off-gas were isolated in the chemical plant low-pressure system. The gas was taken off upstream of the charcoal beds and small samples were subsequently removed for analysis in connection with the reactor xenon studies.<sup>11</sup> After sampling, the gas was passed through the chemical-plant adsorber bed and vented to the stack. Since this gas did not go through the normal off-gas metering station and since the volume is not accurately known, no comparison of oxygen input and discharge can be made for run 25.

The deuterium content of the discharge from the reactor charcoal beds was continuously monitored. The average deuterium concentration was 2.1% and the maximum was 3.6%.

#### COMPONENTS

Satisfactory performance of all the reactor components is vital to the operation of the system. However, the performance of an individual component, or class of components, is described below only if that performance was abnormal in some respect.

##### Fuel Feed Pump

The diaphragm pumps operated without difficulty in run 25 except for the fuel feed pump.

Run 25 was started with both heads of the fuel feed pump in service. Although both heads were newly installed before the beginning of the run, they were not new heads because none were available at the time. The

---

<sup>10</sup>H. F. Bauman et al., op. cit.

<sup>11</sup>W. D. Burch and O. O. Yarbrow, op. cit.

west head was a former blanket-feed-pump head (No. 14) which had served 4035 hours before it developed an intermediate-system leak in run 22. The leak was repaired and the head installed with new flanges for run 25. The east head was a former non-radioactive-test head which was equipped with new check valves for installation in the reactor. Pumping was maintained with one or the other of these heads throughout run 25, but neither gave satisfactory service.

On April 4 the east head began drifting out of phase. It was re-phased three times, with only temporary improvement, before it was removed from service and pumping continued with the west head only. On April 12 the pumping rate with the west head began to decrease. The head was rephased several times with no improvement. Check valve leakage was suspected.

Several changes were made at this time in an effort to improve the overall operation of the fuel feed pump. An improved slide valve and a new east operate-standby selector valve were installed in the oil system. (The removed selector valve was later inspected and found to be in good condition.) The west pulsator was removed and inspected; a rough spot was found on the pulsator and it was replaced; a new O-ring gasket was installed in the pulsator housing. A water tank was installed in which both pulsator housings were submerged to prevent entry of gas into the intermediate systems through the pulsator-housing seals.

An attempt was made to resume pumping with both heads, but the west head would not pump steadily; the sound of the check valves indicated that the suction check valves might be leaking. Operation was resumed using the east head only. Within 16 hours the east head had drifted out of phase. The west head was rephased and it started pumping fairly well; operation continued for the next four days using the west head only.

On April 17 the feed rate dropped off again, the west head was re-phased and the stroke increased to maximum. The east head was tried again briefly, but apparently was losing phasing water inside the cell and would not remain phased. Even at maximum stroke, the west head would not pump the normal feed rate of 7 lb/min, so the feed rate was lowered to 4 lb/min. While the feed rate was low, the fuel-dump-tank solution was concentrated to hold the normal high-pressure system concentration and temperature. After a few hours, the pumping rate of the west head increased spontaneously, and the feed rate was readjusted to the original 7 lb/min. The west head continued in service (although the pumping rate was somewhat unsteady) until the final shutdown of the reactor on April 28.

The total reactor service of the east head (No. 28) was 382 hours, and of the west head (No. 30) 754 hours.

#### Samplers

In general, the samplers functioned well in run 25. However, the blanket high-pressure-sampler inlet valves, HCV-236 and HV-235, developed slight leakage through the seats.

On April 11 the leakage through these valves caused the pressure to rise in the isolation chamber while a sample was being drained to the flask. The high-sampler-pressure interlock (set at 9.5 psig) actuated the mechanisms which lowered the sample flask from under the drain valve. The operator immediately closed the drain valve. When the flask holder was opened at the analytical laboratory, several drops of solution were observed on top of the flask diaphragm. The flask contained a normal sample volume, 9 ml.

For the remaining samples during the run, the blanket isolation chamber pressure was observed closely to be sure that the inlet valves were seated tightly enough to prevent an appreciable pressure rise, before the drain valve was opened. Between samples, a plug was frozen in the blanket high-pressure-sampler inlet line to minimize erosion of the inlet valves which might result in increased leakage.

The volume distribution for the reactor samples in run 25 is shown in Table 1. Ninety-eight percent of the samples were of the required 5-ml minimum volume.

#### Valves

The reactor valves performed well in run 25; only minor difficulties were encountered.

During the startup, on March 13, it was discovered that the loading and actuation air lines to the fuel dump valve had been interchanged when they were reconnected following replacement of the fuel letdown heat exchanger. The valve operation appeared to be correct at the time the air lines were reconnected; the anomaly may have been due to an error in the manipulation of the temporary air supply to the valve while the reactor cell was flooded. In any event, the normal operation of the valve was restored by simply interchanging the air lines at the north-shield face without opening the cell.

During the test of the reactor high-pressure interlocks on March 19, it was discovered that the blanket dump valve failed to open in response to a normal air signal. Investigation showed that one of the air lines in the cell was disconnected. The cell was opened and the connection was completed with the aid of a remote tool. Since the original connections had been double-checked following the installation of the valve, it was concluded that the disconnect was probably worn or faulty and had failed to seat properly. (A thorough examination of the disconnect was not attempted because of the high radiation background.) However, the final connection was secure, and did not cause any further difficulty.

The fuel letdown valve stuck in the open position during the pre-treatment of the reactor on March 26, forcing the depressurization and cool-down of the reactor. After a short time, the letdown valve returned to normal operation spontaneously, and no further sticking was observed.<sup>12</sup>

---

<sup>12</sup>A similar incident had occurred once previously, on January 14, 1959, in run 18.

Table 1. Sample Volumes in Run 25

| Volume<br>Range<br>(ml) | Number of Samples           |          |                            |          | Totals   |
|-------------------------|-----------------------------|----------|----------------------------|----------|----------|
|                         | <u>High-Pressure System</u> |          | <u>Low-Pressure System</u> |          |          |
|                         | Fuel                        | Blanket  | Fuel                       | Blanket  |          |
| 0 - 5                   | 0                           | 1        | 0                          | 0        | 1        |
| 5 - 8                   | 4                           | 5        | 0                          | 0        | 9        |
| 8 - 10                  | 22                          | 14       | 0                          | 0        | 36       |
| 10 +                    | <u>0</u>                    | <u>5</u> | <u>3</u>                   | <u>0</u> | <u>8</u> |
| Totals                  | 26                          | 25       | 3                          | 0        | 54       |

### Freeze Jackets

During pretreatment operations preliminary to run 25, the freeze plugs isolating the cracked tee in the fuel feed line<sup>13</sup> thawed. The chemical-plant refrigeration unit was supplying Freon to the freeze jackets at approximately  $-20^{\circ}\text{C}$ . The return refrigerant, which had been approximately  $-5$  to  $-10^{\circ}\text{C}$ , rose to  $0^{\circ}\text{C}$  at about 2200 on March 25. Attempts to refreeze the plugs while the liquid in the tanks was boiling were unsuccessful, so the high-pressure system was cooled and depressurized on March 27, to allow the dump tanks to be cooled and to prevent fluid from leaking down from the high-pressure system.

The fuel was transferred from the dump tanks and then condensate was added to improve the conditions for freezing. The quantity of water was small enough (approximately 160 lb) to prevent circulation by convection through the dump-tank evaporator legs. The load on the chemical-plant refrigeration system (from other freeze jackets) was reduced so that the Freon going to the freeze jackets was at  $-38^{\circ}\text{C}$  and that returning, at  $-34^{\circ}\text{C}$ . Refrigerant flow through each jacket was about 1 gpm. When the heating steam was turned back on the dump-tank evaporators, the return refrigerant temperature started rising, however, and within 30 minutes indicated that the plugs had thawed. The steam was turned off for another attempt at freezing the plugs. This time water was added to the reactor cell to the 807.3-ft elevation in order to immerse the jackets and provide better thermal contact with the pipe. Refrigerant flow was sent to the horizontal jacket for several hours without flow to the vertical one to further improve the conditions for freezing. Refrigerant went to the jackets at about  $-38^{\circ}\text{C}$  and left at  $-20^{\circ}\text{C}$ . After 24 hours the cell was drained and steam was turned on to the evaporators again. The thermocouples again indicated that the ice plugs were not maintained.

On March 29, the freeze-jacket coolant supply was switched to a portable dry-ice-kerosene unit which could supply coolant at  $-60^{\circ}\text{C}$ . The flow was greater than 1 gpm to each bundle. After 16 hours, steam was turned on to the dump-tank evaporators and the temperatures at the freeze bundles rose to  $0^{\circ}\text{C}$ . Since the portable unit required constant attention, it was shut down and a change back to the chemical-plant unit was effected. The reactor cell was again flooded for about 16 hours to immerse the freeze bundles in water. Refrigerant was supplied to each bundle at  $-40^{\circ}\text{C}$  at a flow of about 1.3 gpm.

Startup for run 25 was resumed on March 30 without repeating any of the special tests for ice plugs, since there was some doubt that the thermocouples indicated the true situation there. The best test of whether or not the remedial action at the leaky fitting had stopped the fuel leak appeared to be to operate the reactor while closely watching the gaseous-activity level in the cell. This was done and

---

<sup>13</sup>H. F. Bauman et al., op. cit.

throughout the experimental program the total gaseous-fission-product activity level never exceeded 4 curies, nor was any fresh fission-product activity detected in the cell sumps.

#### MISCELLANEOUS

A number of aspects of the reactor system operation are observed on a routine basis. In addition, there are sometimes special problems which arise in the course of a run. These subjects are discussed in the succeeding sections.

#### Concentration Ratios

The ratio of solute concentrations in the core and blanket high-pressure systems depends on the rate of solution mixing between the two regions and on the net blanket purge rate. The purge rate can be separately controlled, but the mixing rate is determined by the hole configuration and the pressure drop across the holes. (The purge rate also exerts some influence on the mixing rate.) Since the system pressure drops do not vary widely, any uncontrolled change in the mixing rate normally indicates a change in the hole geometry.

The concentration ratios, from which the mixing rates are calculated, are normally based on the analyses of core and blanket samples. However, an alternate method may be used to obtain trends in the concentration ratio between sample times. This is based on the calculated relation between the concentration ratio and the power ratio. Figure 2 shows the relation between these two variables for the general operating conditions of run 25, calculated by the GNU code, a multigroup neutron diffusion calculation. The experimental results from run 25 samples are also shown. The greater scatter in the data at heat-loss power is probably due to the greater relative error in the power determination at low levels.

In normal operation the concentration ratio is fixed by the operating temperatures and the desired blanket concentration. This ratio is then established by adjusting the blanket purge rate. In run 25 the minimum blanket uranium concentration was set at 1.5 g U/kg D<sub>2</sub>O after a short period of operation with the concentration at 1.1 g U/kg D<sub>2</sub>O. There were periods of operation with 2.0 and 1.7 g U/kg D<sub>2</sub>O in the blanket, and, after the failure of the upper patch, the blanket concentration reached 3 g U/kg D<sub>2</sub>O.

The observed correlation between net blanket purge rate and concentration ratio is shown in Fig. 3 for the period of operation in run 25 up to the time of the patch failure. A least-squares line was drawn through the data to permit evaluation of the back-mixing rate. The back-mixing was evaluated at a net blanket purge rate of 4 lb/min for comparison with previous observations. A rate of 1.8 lb/min was obtained, compared with 1.3 lb/min in runs 23 and 24 and 0.9 lb/min in run 22.<sup>14</sup>

---

<sup>14</sup>Ibid.

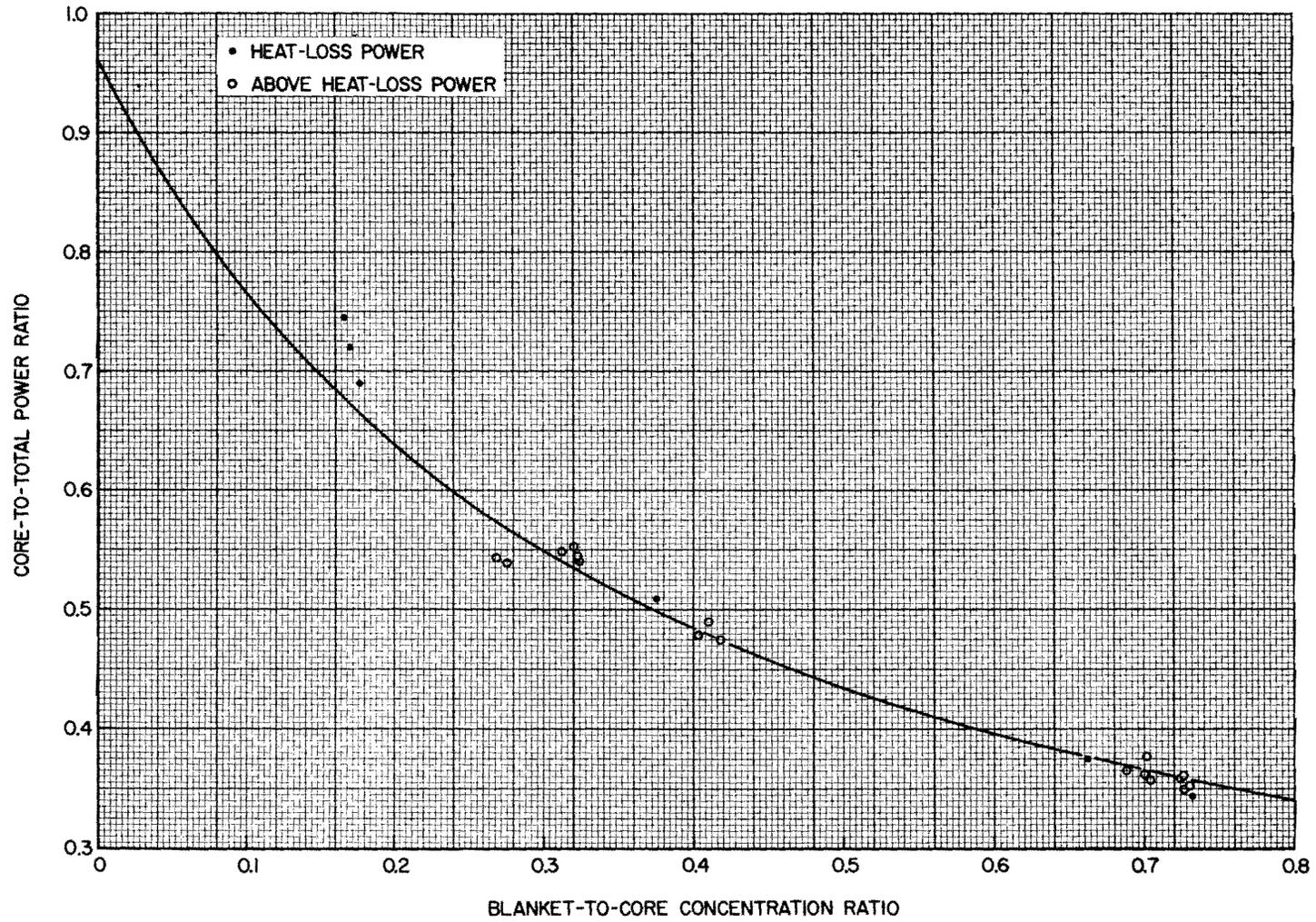


Fig. 2 Effect of Concentration Ratio on Power Distribution in Run 25

UNCLASSIFIED  
ORNL-LR-DWG 63814

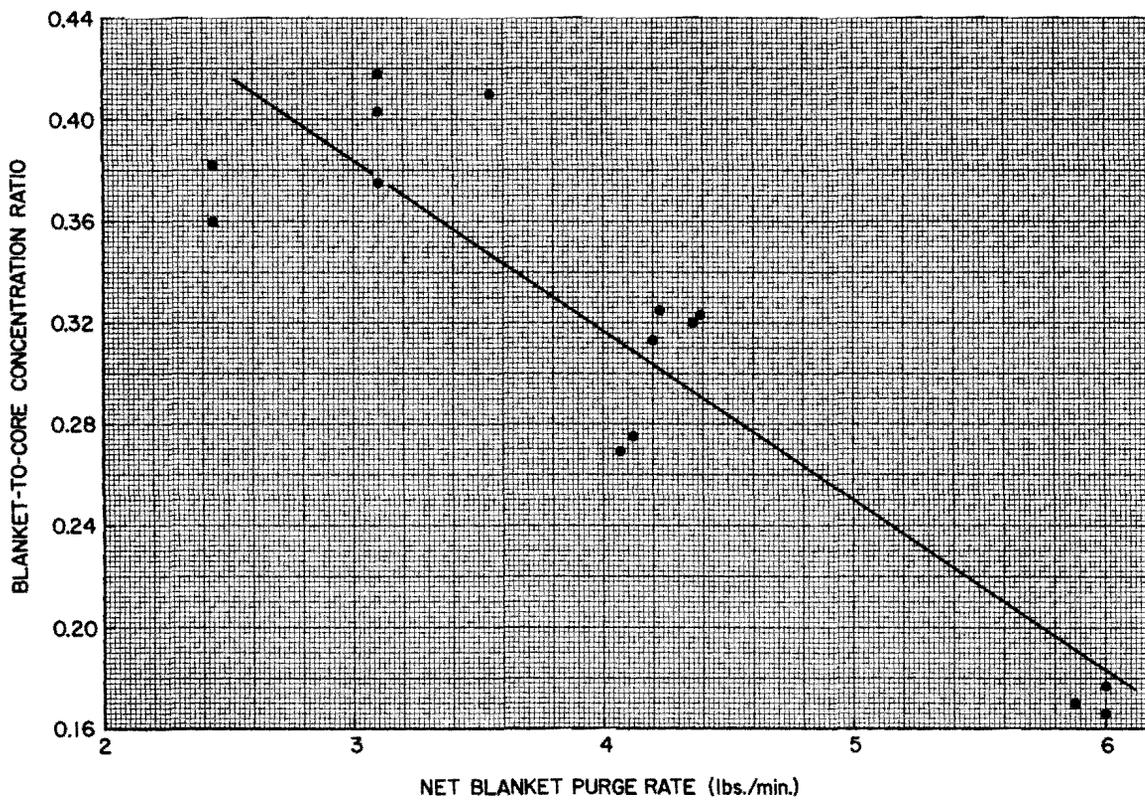


Fig. 3 Effect of Net Blanket Purge Rate on Concentration Ratio in Run 25

These values indicate that the condition of the patches was growing progressively worse, even before the upper patch fell out. The back-mixing rate increased to about 14 lb/min after the failure of the upper patch.

#### Critical Concentrations

Each time a pair of high-pressure samples was taken during run 25, the core uranium concentration indicated by the sample analysis was compared with the predicted critical concentration for the conditions at the time of the sample. The purpose was to show possible effects of non-circulating uranium and, secondarily, to check the accuracy of the analysis and the prediction.

The critical concentration was predicted from the core temperature, the blanket temperature and the blanket/core concentration ratio at sample time. Criticality relationships developed from GNU multigroup neutron diffusion calculations were used. The curves had agreed closely with observed concentrations in run 24, even though the GNU calculations were made for lower enrichment and more poisoning than existed in run 24.<sup>15</sup> Since no intentional changes were made in the fuel between runs 24 and 25, it was expected that the curves would be applicable for run 25. The "observed" U-235 concentration was obtained from the concentration of total uranium shown by sample analysis, corrected for analytical bias<sup>16</sup> and multiplied by the book enrichment of the circulating uranium. The book enrichment was computed from the enrichments and amounts of fuel added at the beginning of run 24 and in solution at that time.

Thirteen pairs of high-pressure samples were taken during the first two weeks of run 25 while the blanket/core concentration ratio ranged from 0.18 to 0.42. The ratio at the time of the remaining ten pairs was between 0.66 and 0.73.

For the first group of sample pairs, the ratio of predicted/observed concentration was  $0.972 \pm 0.012$  (mean and standard deviation). The ratio for run 24 samples was significantly higher:  $0.997 \pm 0.009$ . The difference is probably due in part to the generally higher blanket/core concentration ratios which existed in run 25. As shown by the run 25 data in Fig. 4, the error in the prediction appears to be greater at higher blanket/core concentration ratios. Even allowing for this effect, the run 24 points are still unaccountably higher than those for run 25. Change in enrichment of uranium between the two runs could explain the difference. The fuel added at the beginning of run 24 contained 93% U-235, replacing fuel of 83% enrichment. Thus there was a chance that the fuel enrichment decreased between runs 24 and 25 as a result of mixing with more of the old fuel. (The fuel was transferred into and out of the

---

<sup>15</sup> Ibid.

<sup>16</sup> In run 25 the ratio of analysis to actual uranium concentration in thirteen core control samples was  $1.006 \pm 0.012$ .

UNCLASSIFIED  
ORNL-LR-DWG 63815

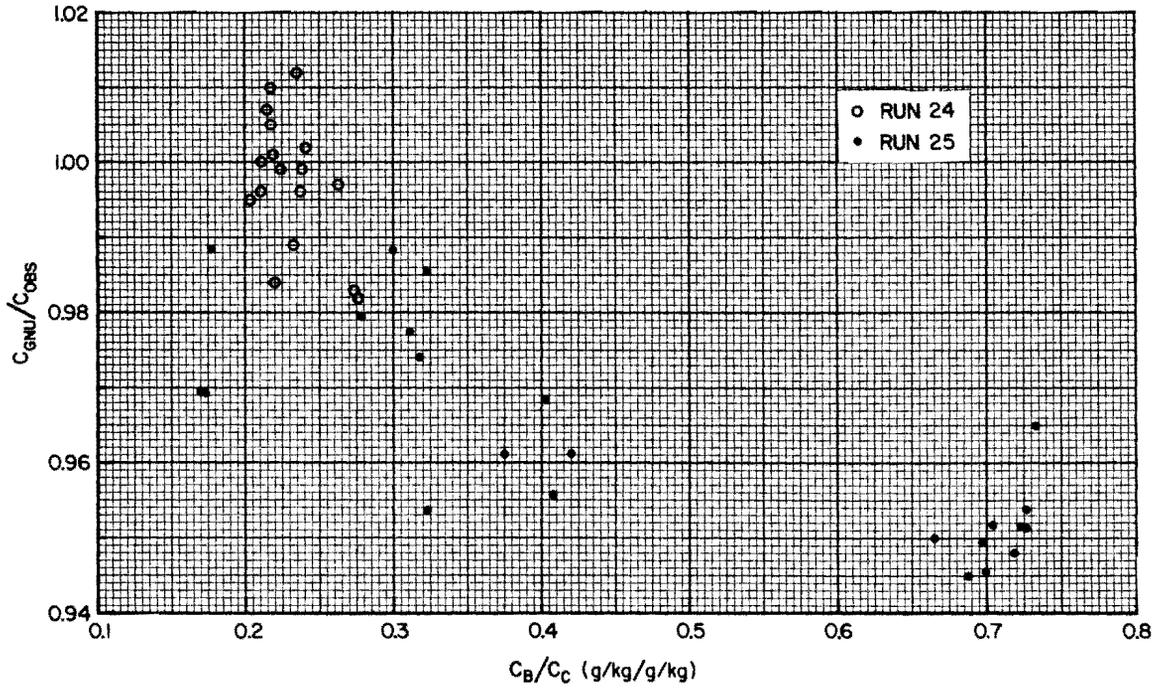


Fig. 4 Ratio of Critical Concentrations Predicted by GNU Calculations to Those Observed in Runs 24 and 25 vs. Blanket-Core Concentration Ratio

blanket storage tanks between runs.) Unfortunately no samples were assayed for uranium isotopic composition during run 25. (It was expected that an assay would be forthcoming when the fuel was processed after run 25. The fuel has not been removed from the reactor as of this writing, however.) If the actual enrichment were lower than the book, the "observed" U-235 concentrations would be high and the ratio of predicted/observed would be low.

#### Core Patch Failure

On April 17 the mixing between the core and blanket increased sharply, causing the blanket concentration to rise from 0.3 to 0.7 of that in the core. This increase was explained after the run was concluded, when inspection of the core disclosed that the patch had fallen out of the upper core-tank hole. As shown in Fig. 5, the patch was found lying on the top diffuser screen, with the head portion of its retaining bolt lying nearby.

The patch and the bolt fragment were recovered and examined by the Post-Irradiation Examination Group of the Metallurgy Division. The bolt was found to have been melted in two on the blanket side of the patch near the toggle nut. Figure 6 shows the bolt fragment re-inserted through the patch, and Fig. 7 shows, for reference, a mockup of the patch and bolt. Examination showed that an area on the surface of the patch adjacent to the bolt had been raised to approximately 800°C, and had cooled fairly slowly (in more than a few seconds).<sup>17</sup> The core side of the patch and the surfaces in contact with the bolt and the tank wall were in good condition.

The time at which the bolt melted cannot be exactly established. The most likely explanation is that uranium accumulation on the bolt caused film boiling, overheating and melting at some time when the reactor was at high power. The increase in blanket concentration which resulted from the patch's falling out occurred while the reactor was subcritical, however, and it appears that the power was lowered before the concentration increase began.

Figure 8 shows reactor temperatures and powers and an index of the fuel feed rate at the beginning of the subcritical operation during which the blanket concentration increased. The reactor had been operating at 5 Mw for 79 hours, and conditions were steady at the beginning of the 2-hour period shown here. When the decrease in NAT was noticed at 1030, the fuel feed rate was suspected and the fuel feed cooler temperatures were checked. It was found that the feed rate was decreasing. The by-pass line, which was installed to free the pump when gas-bound, was opened but with no effect. The steam withdrawal was shut off at 1043 to drop the power. In the next half hour the blanket

---

<sup>17</sup>M. L. Picklesimer, personal communication.

UNCLASSIFIED  
PHOTO 53828

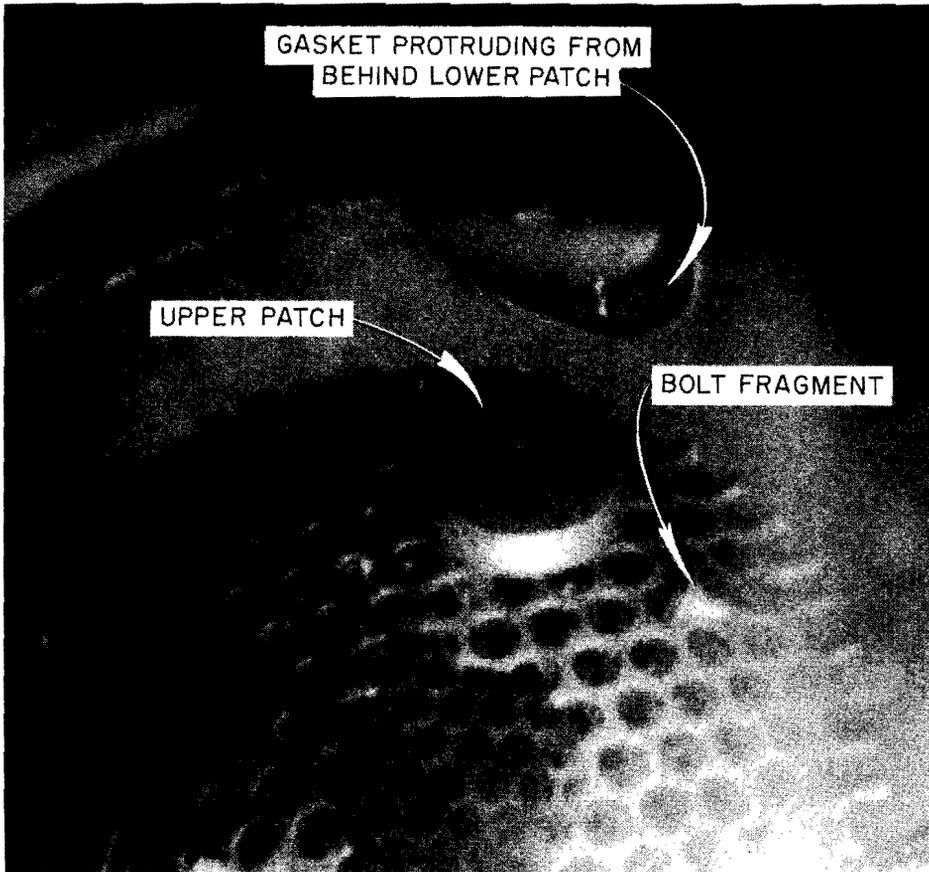


Fig. 5. Upper Patch and Bolt Fragment as Found in Core After Run 25.

UNCLASSIFIED  
PIE 913A



Fig. 6 Remnant of Zircaloy-2 Bolt Sticking Through Upper Patch

UNCLASSIFIED  
PHOTO 54063R1

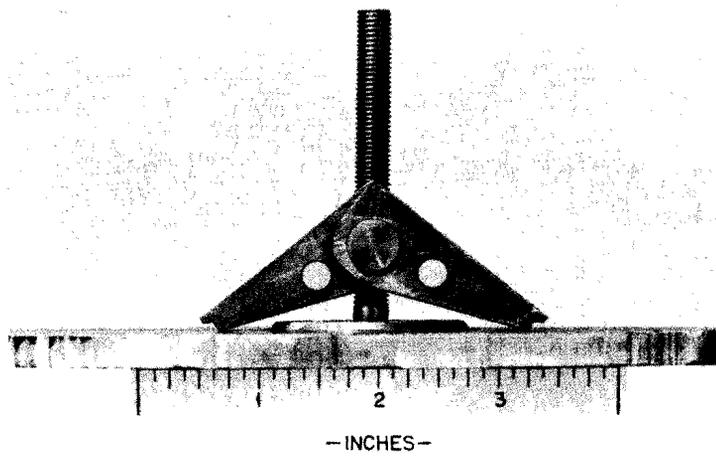


Fig. 7 Scale Model of Upper Hole Patch Assembly

UNCLASSIFIED  
ORNL-LR-DWG 63817

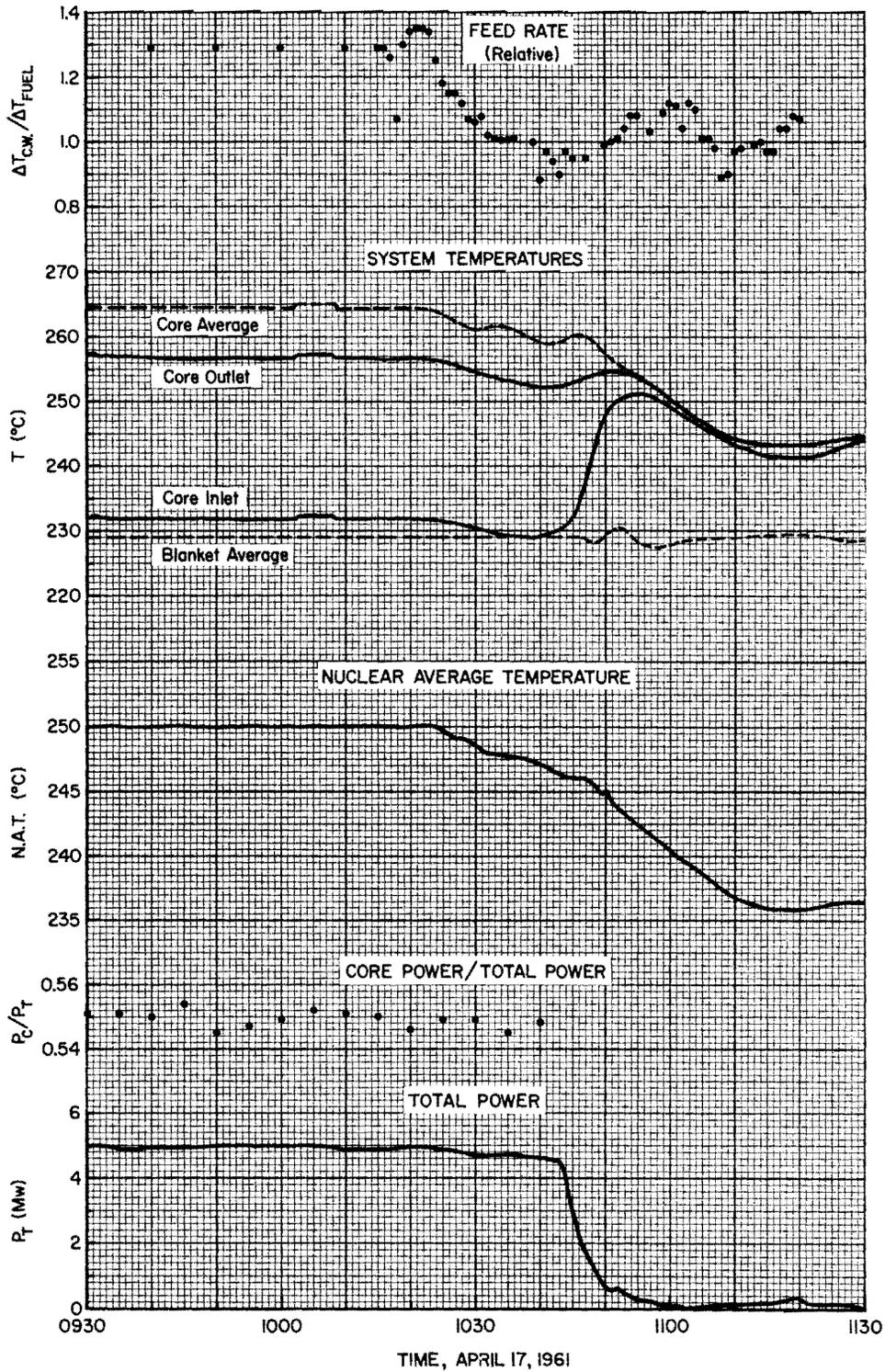


Fig. 8 Last Power Operation Before Increase in Core-Blanket Mixing

purge was reduced, the fuel feed pump stroke was lengthened, and the fuel feed bypass was opened twice to vent any gas trapped in the pump head. It was decided to take the reactor subcritical, so at 1120 the package boiler was started to heat the fuel above the critical temperature.

When the reactor was next made critical, at 2043 on the same date, the ratio  $P_c/P_t$  was found to be only 0.35, indicating a very large increase in blanket concentration during the subcritical operation. The increased blanket/core concentration ratio caused the critical temperature to be 13°C lower than before the upset. The decrease in NAT which started at 1023 was not caused by increased core-blanket mixing, however. The ratio  $P_c/P_t$  shows that the concentration ratio did not change significantly before the power was dropped. Calculations of the effect of the observed changes in fuel feed rate on core concentration indicate that, within the accuracy of the calculations, the NAT decrease was what should result from the feed rate changes alone.

Thus it appears that the core-tank patch did not fall out before the power was lowered, certainly not more than 5 minutes before, else the power ratio would have shown the effect. If, as seems most likely, the bolt melted earlier, while the power was at its highest, the patch may have remained stuck in place and became loose only when the power was lowered and the tank cooled.

The examination of the core interior showed the patch over the lower core-tank hole still fastened in place. The gold gasket was out of its proper position and was hanging down below the patch. Undoubtedly the leakage past this patch was greater than shortly after it was installed, which may explain the progressive increase in core-blanket mixing during runs 22 through 25.

## Containment

### Cell Leakage

Considerable difficulty was experienced in satisfactorily sealing the reactor-cell roof pans prior to the run 25 startup. The first leak test was made on March 15, shortly after the last roof-pan weld was completed. Several gross leaks (>10 liters/min each) were found when the cell was pressurized to 5 psig. Repairs were made and, early on the following day, the leakage was again tested at 5 psig; it was still excessive. Repairs were made and another test was made on the 16th, with leakage still being excessive.

Up to this time, a thermosetting resin had been used extensively to seal small cracks in the seal pans. In many cases the cracks were difficult to reach for rewelding, or they could not be located precisely, and relatively large areas were covered with the resin to insure sealing. The resin was also applied to some areas which were only questionable. Adequate sealing was achieved by this method, but the resin did not have the structural strength of the steel pans. Many of the leaks found in

the cell tests for run 25 were in areas that had not been affected by the preceding maintenance period. These apparently resulted from failure of old resin seals due to flexing of the pans. The resin was removed from these areas and the leaks were repaired by welding. Subsequent use of the resin was limited to very small leaks. Another factor which contributed to the sealing problem was the fact that the fit of the upper concrete slabs was very tight, with almost no gap existing between most slabs. This frequently made it necessary to use hydraulic jacks to force the last slabs in a bay into position, which may have caused excessive strains in the pan seams.

Leak tests were repeated on March 17, after the previously located leaks were repaired. Leakage was greatly improved but was still excessive; the total leakage at 5 psig was 328 cc/min, and at 15 psig was 965 cc/min. Additional leaks were opened at 15 psig. After the leaks were repaired, a total of 600 cc/min leakage was measured at 15 psig. However, new leaks were again opened because the leakage, as the pressure was lowered, was 1200 cc/min at 5 psig. During this test a leak of about 8 liters/min was found through the line connecting the cell and the east instrument cubicle. The cubicle rupture disc was replaced and the drain valve was cleaned out to eliminate this leak.

A new approach was used when the cell was opened to reconnect the air line to the blanket-dump-valve operator (see page 15). For the first time only a flat section of the roof pan was cut out, so that the welds to close it did not occur at a pan seam or concrete slab edge. After a leak, located at the new weld at 1 psig, was repaired, the shield was satisfactorily leak tested. Roof-pan leakage rates were 170 cc/min at 15 psig and 17 cc/min at 5 psig, acceptably low.

The cell was evacuated to 0.5 atmosphere to start the vacuum leak test on March 21. An initial high leakage of 15 liters/min was determined. It was soon discovered, however, that the cell ventilation valve (HV-532A) was leaking grossly. After the valve gasket was repaired, a satisfactory leak rate of 3.1 liters/min was determined on March 23.

Throughout run 25 operation, the cell leakage was approximately 3 liters/min with the cell maintained at 0.5 atmosphere. The leakage rate was corrected for a normal nitrogen purge of 3 liters/min and a demineralized-cooling-water leak of about 0.5 liter/min.

#### Flange Leakage

During run 25 there was no significant individual flange leakage. Total leakage from the flange leak-detector system ranged between 20 and 30 cc/day during periods of steady reactor operation.

#### Stack Filter

All gaseous wastes from the HRT system and the building ventilation air are ultimately passed through a series of filters<sup>18</sup> and

---

<sup>18</sup>H. F. Bauman et al., op. cit.

released to the atmosphere from a 100-ft-high stack. The filter system consists of an absolute particulate filter, a bed of silver-plated wire mesh followed by an array of charcoal canisters for iodine removal and a second absolute filter. When small amounts of radioiodine were detected in the filtered stack gas, a limited testing program was planned to evaluate the effectiveness of the filter system. These tests were performed while run 25 was in progress.

Two continuous gas samplers were installed at the stack--one just upstream of the iodine traps and one downstream of the entire filter system. In the first test 10 millicuries of iodine-131 was injected into the vapor space in the 1000-gal waste tank to simulate a spill of fuel solution in one of the sampler cavities, all of which drain to this tank. This test was inconclusive because only about 1% of the injected I-131 reached the sample point upstream of the main iodine traps. The vent line which carries gas from the waste tank to the stack contains a separate iodine trap which probably held up much of the iodine. In addition, some may have been retained on the pipe walls and the first particulate filter. In the second test a 10-millicurie charge of I-131 was injected just upstream of the filter. This time 10% of the iodine reached the first sample point and the rest was, presumably, held on the first particulate filter. Only about 1% of the iodine which reached the first sample point escaped into the stack. This test indicated that a minimum decontamination factor of  $10^3$  could be expected from the stack filter alone in the event of a significant activity release. This factor is considerably higher than those calculated for the small releases which prompted the investigation. It was, therefore, tentatively concluded that very small concentrations of activity were less efficiently removed from the stack gas. In either case the total activity release was very small.

## FUEL SOLUTION CHEMISTRY AND CORROSION

### Fuel Composition

The physical inventories of solution constituents taken during run 25 are plotted against operating time above 200°C in Fig. 9. Each inventory is based on a pair of samples from the fuel and blanket high-pressure systems and the weights, volumes, temperatures, pressures, and feed and purge rates which determine the concentrations and amounts of solution in all parts of the reactor.

The average levels of uranium, copper and sulfate inventories at the beginning of run 25 were 96% of the average levels calculated during the part of run 24 when letdown heat exchanger leakage did not affect the inventory calculations. About 2% reduction between the two runs was expected on the basis of the amount of fuel recovered from the reactor cell sump after the leak which terminated run 24. The additional 2% difference may be the result of incomplete recovery from the storage tanks. The initial inventory of nickel in run 25 was

UNCLASSIFIED  
ORNL-LR-DWG 63818

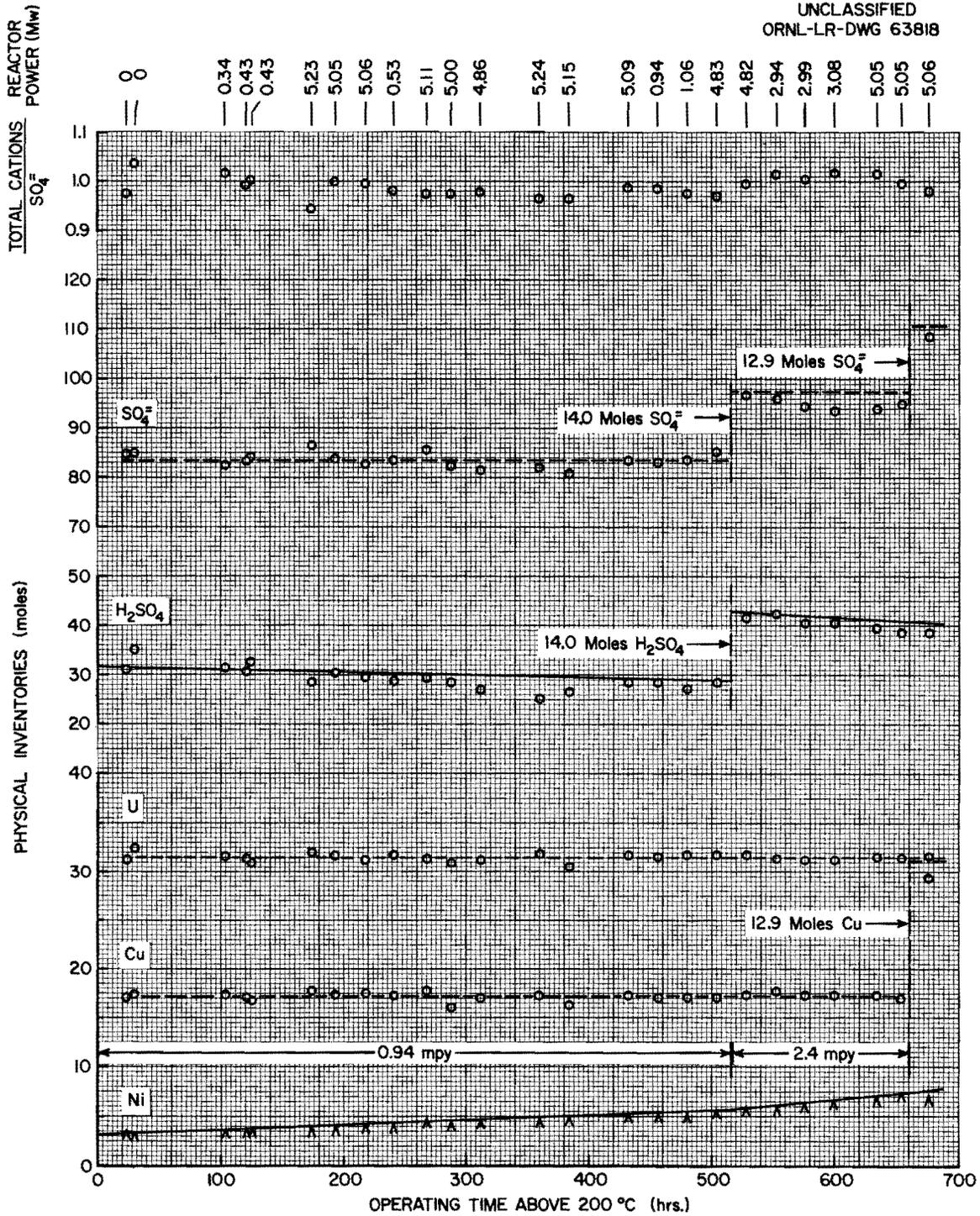


Fig. 9 Physical Inventories in Run 25

3.2 moles, about 0.9 moles higher than would be calculated for 96% carryover of run 24 fuel.

Two additions were made to the fuel solution during run 25. Fourteen moles of sulfuric acid was added at 516 hours, and 12.9 moles of cupric sulfate was added at 660 hours, shortly before the final shut-down of the reactor. Both of these additions were made for the purpose of determining their effect on fuel stability. The acid raised the minimum heavy-liquid-phase temperature to more than 10°C above the heavy-water boiling point at 1750 psig, in the hope of being able to operate for the first time at this pressure and 5 Mw without fuel instability. The copper addition doubled the existing concentration to explore the effect of reduced radiolytic-gas concentrations on reactor behavior.

The most important parameters describing the composition of the fuel at several times during the high-power operation are listed in Table 2. The changes resulting from the fuel additions are obvious, as are the changes in blanket/core concentration ratio, power ratio, and blanket concentration which occurred at about 435 hours (April 17). The ionic ratios reported here are computed from the lines representing average inventory trends in Fig. 9. The concentration of the sulfate ion in the blanket solution is taken directly from the results of a blanket sample in each case. The other features of Table 2 will be discussed later (see page 37).

#### Fuel Stability

From the standpoint of overt fuel instability, run 25 was sharply divided into two periods by the loss of the core patch, with the attendant increase in core-blanket mixing, which occurred on April 17. Before this event, the reactor was operated at 5 Mw for periods of 62, 8, 52 and 80 hours. During none of these was there any observable evidence of fuel instability. Afterwards, the power was raised to 5 Mw on ten occasions. In every case, there was a rise of the NAT which could not be explained except by fuel instability and deposition of uranium on the core tank. This NAT behavior was the only symptom of fuel instability observed during the run.

#### Behavior of NAT

During the first part of run 25, the NAT was not significantly upset when the power was raised to 5 Mw or lowered from that power. There were some perturbations, to be sure. But these were only about 1°C or less and were attributable to changes in the bulk mean density of the high-pressure systems as the power and temperatures were changed. When the power was first raised to 5 Mw after the increase in mixing, the NAT rose, in the brief period at 5 Mw, more than could be expected on the basis of past experience. The NAT returned to normal when the power was lowered. Every time the power was raised to 5 Mw thereafter, there was a similar rise of from 1.5 to 3°C, which could not be explained by mixing variations or any other known factor. Therefore, it was concluded that the NAT variations were evidence of fuel instability.

Table 2. Some Fuel Solution Parameters During Operation at 5 Mw in Run 25

| Operating Time<br>200-300°C | Concentration Ratio<br>$\left(\frac{\text{Blanket}}{\text{Core}}\right)$ | Power Ratio<br>$\left(\frac{\text{Core}}{\text{Total}}\right)$ | Reactor Inventory  |   |  |  | Blanket Solution                                     |  |      |      |                    |
|-----------------------------|--|--|--|---|--|--|--|--|------|------|--------------------|
|                             |  |  | $\left(\frac{\text{H}_2\text{SO}_4}{\text{SO}_4^{=}}\right)$ | $\left(\frac{\text{U}}{\text{SO}_4^{=}}\right)$ | $\left(\frac{\text{Cu}}{\text{SO}_4^{=}}\right)$ | $\left(\frac{\text{Ni}}{\text{SO}_4^{=}}\right)$ | Concentration of $\text{SO}_4^{=}$<br>Molal x $10^3$ | Fraction of Saturation at<br>300°C in 3-Component System |      |      | Total of Fractions |
|                             |  |  |  |   |  |  | U  | Cu   | Ni   |      |                    |
| 174                         | .410   | .491   | .364   | .377  | .207   | .048   | 25.1   | .429   | .399 | .107 | .935               |
| 193                         | .409   | .475   | .363   | .377  | .207   | .049   | 26.0   | .428   | .399 | .109 | .936               |
| 218                         | .403   | .482   | .361   | .377  | .207   | .050   | 25.5   | .429   | .399 | .112 | .940               |
| 268                         | .320   | .554   | .358   | .377  | .207   | .053   | 21.8   | .438   | .406 | .116 | .960               |
| 288                         | .275   | .540   | .357   | .377  | .207   | .054   | 20.6   | .443   | .408 | .118 | .969               |
| 312                         | .312   | .543   | .356   | .377  | .207   | .055   | 20.8   | .442   | .407 | .121 | .970               |
| 360                         | .325   | .542   | .353   | .377  | .207   | .058   | 21.2   | .440   | .407 | .127 | .974               |
| 384                         | .269   | .548   | .351   | .377  | .207   | .059   | 20.0   | .446   | .409 | .129 | .984               |
| 432                         | .323   | .546   | .348   | .377  | .207   | .062   | 21.9   | .438   | .405 | .136 | .979               |
| 504                         | .726   | .340   | .344   | .377  | .207   | .066   | 36.4   | .409   | .385 | .154 | .948               |
| 528                         | .702   | .369   | .436   | .323  | .177   | .059   | 41.7   | .344   | .326 | .140 | .810               |
| 634                         | .699   | .362   | .421   | .323  | .177   | .072   | 40.8   | .345   | .327 | .170 | .842               |
| 654                         | .705   | .361   | .419   | .323  | .177   | .074   | 41.1   | .345   | .326 | .176 | .847               |
| 676                         | .687   | .366   | .367   | .285  | .273   | .068   | 46.6   | .300   | .498 | .163 | .961               |

The system behavior accompanying a power increase and reduction during the latter part of the run is shown in Fig. 10. The sharp rise after the power was raised to 5 Mw and the sharp drop just after it was lowered to 4 Mw are obvious. This figure also illustrates some typical reasons why many such experiments in the HRT were not clearcut. The fuel-dump-tank weight was not controlled adequately, and a small part of the NAT rise was caused by the concentration of the dump tanks at 1440. Furthermore, the fuel-feed-rate index<sup>19</sup> shows that the feed rate drifted up and down, which would cause some NAT variation (approximately 2°C per unit change in the index).

A clear illustration of the reversible effect of power on NAT is given in Fig. 11. Note the rapidity with which the NAT shifted on dropping the power. The NAT change is probably not a result of power-dependent changes in the core-flow pattern which might affect the weighting factors on inlet and outlet temperatures. Such changes would be expected to vary continuously with power, but there was no significant NAT shift during the large power changes between 1, 3 and 4 Mw. The large shifts in NAT which occurred when the power was changed from 4 to 5 Mw and from 5 to 3 Mw suggest a threshold phenomenon which started above 4 Mw.

#### Other Symptoms

Besides the behavior of the NAT there were, during the run, no other detectable symptoms of fuel instability. At no time was any abnormal power disturbance observed, and Fig. 12 shows that there was no significant correlation of physical inventories with power. The four inventories shown in this figure at high power (above 4.5 Mw) and at high blanket concentrations were taken when the NAT was from 1.7 to 2.9°C above the NAT expected in the absence of fuel instability. All the other inventories were at times when there was no symptom of fuel instability. The inventory differences are certainly within the scatter of the data. (The standard deviation of the high-power uranium inventories before the increase in mixing is 0.5 moles of uranium.)

#### Probable Explanation of NAT Rise

It appears on the basis of all the evidence that the NAT rise at high power in the latter part of the run was the result of deposition of uranium on the blanket side of the core tank, probably by boiling.

The blanket region is the most likely seat of the trouble, first of all, because when the NAT rise appeared, the blanket power was the highest in the history of the HRT and the core power was much lower than during previous, apparently stable operation.

---

<sup>19</sup>The fuel-feed-rate index is a relative number, approximately proportional to the feed rate, and which is obtained from the indicated fuel and cooling-water temperatures around the feed cooler and the cooling-water flow rate.

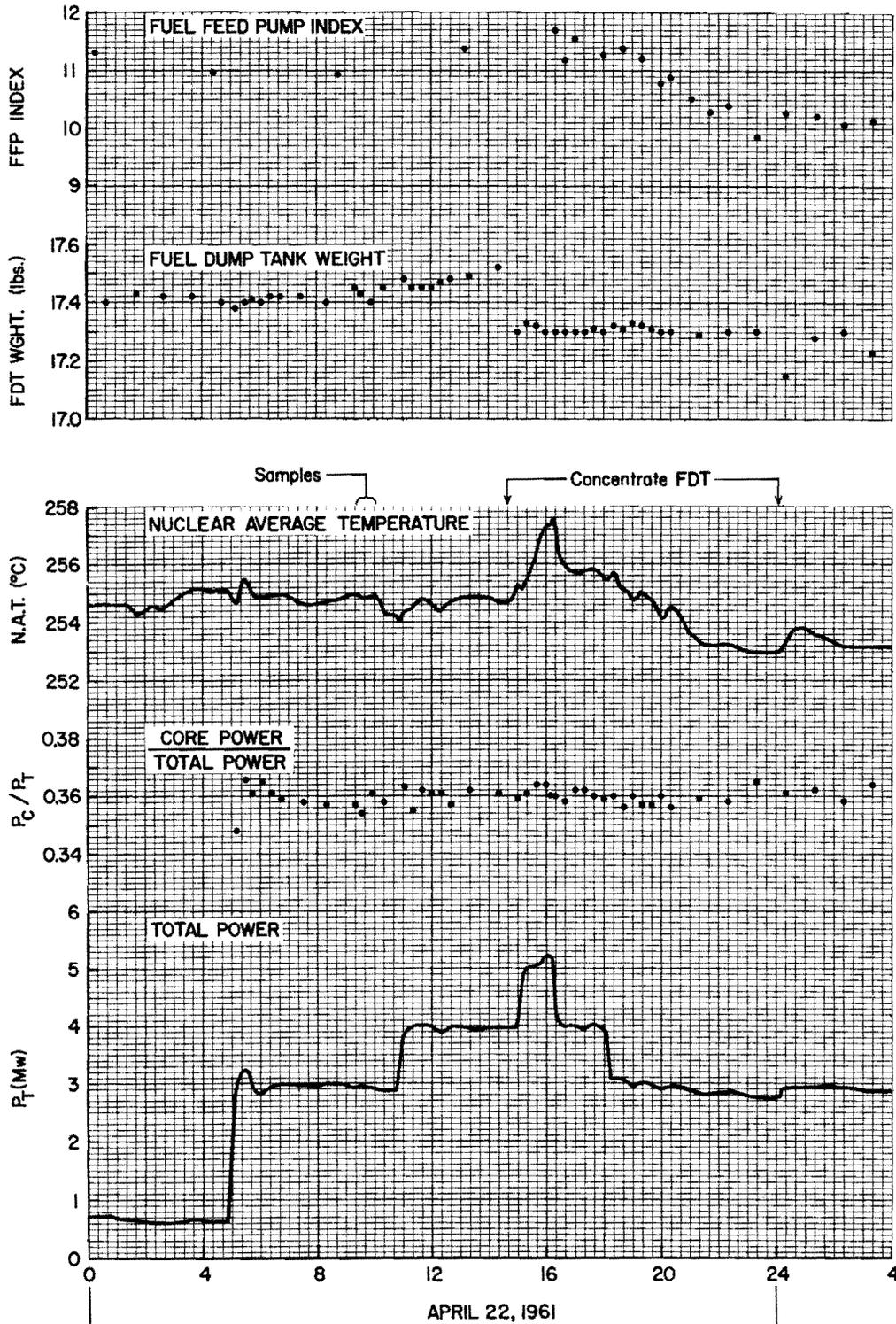


Fig. 10 System Conditions and Behavior During an Experiment Resulting in N.A.T. Rise

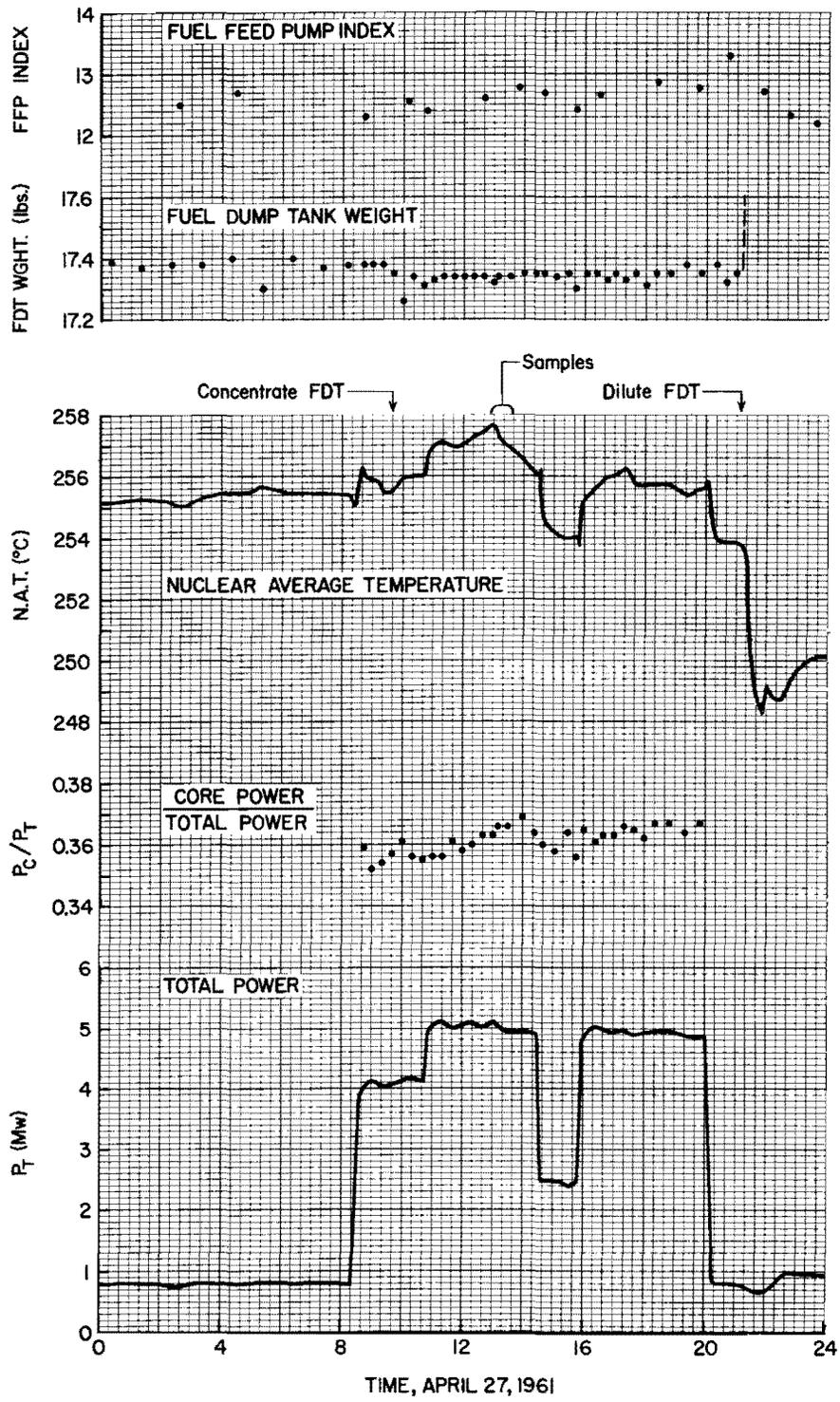


Fig. II Illustration of Reversible Power Effect on N.A.T.

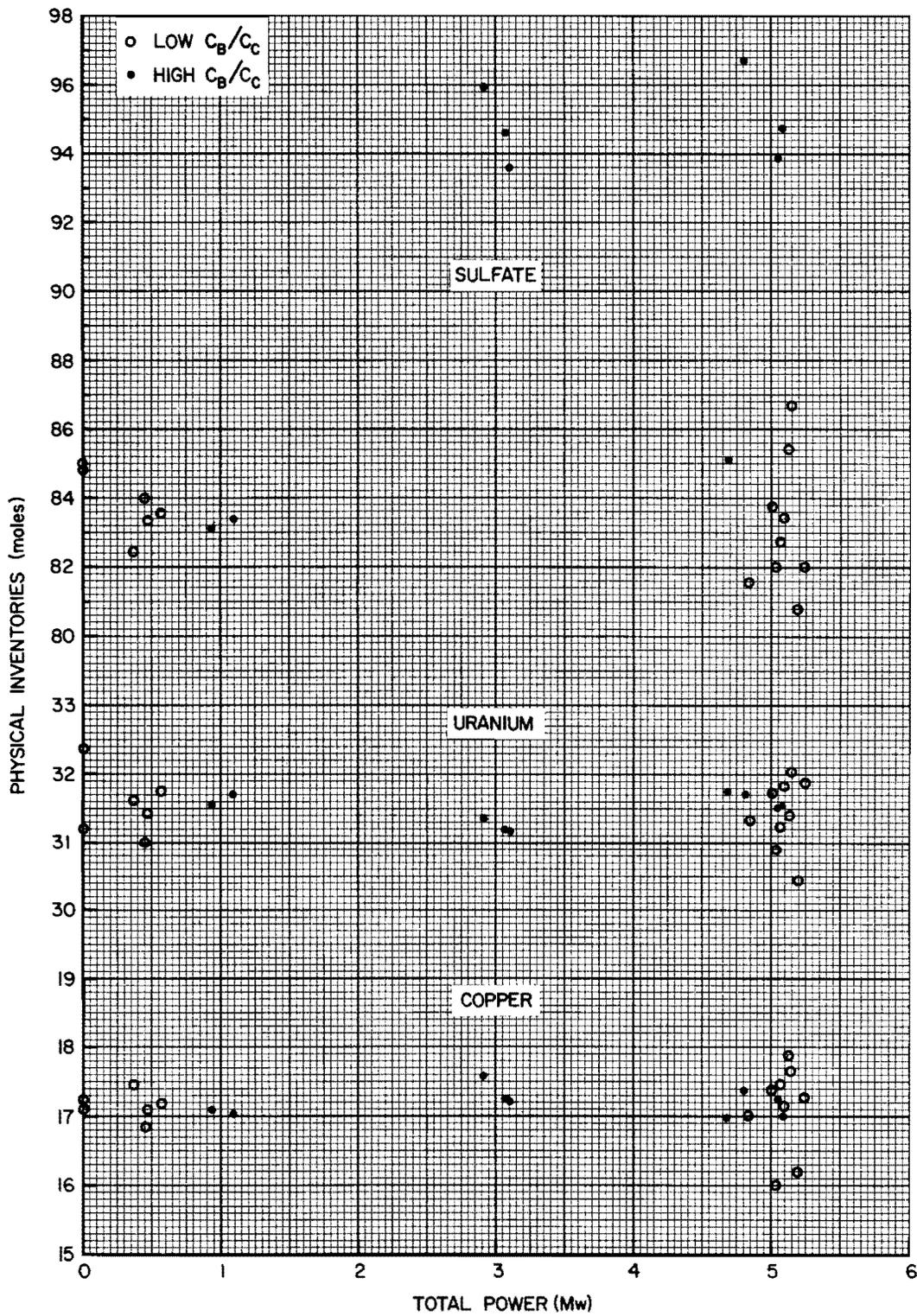


Fig.12 Physical Inventories vs. Total Power in Run 25

Another indication that the instability was in the blanket is the smallness of the changes in physical inventory and the nature of the NAT transients following power changes.

The change in NAT which accompanies relocation of uranium depends on the reactivity worth of uranium at the original and final location. For run 25 conditions, the approximate increase in reactivity resulting from the addition of one gram of U-235 at various locations is: on the core tank,  $1.5 \times 10^{-4}$ ; evenly distributed in the fuel high-pressure system,  $1.3 \times 10^{-4}$ ; and evenly distributed in the blanket high-pressure system,  $0.5 \times 10^{-4}$ . The temperature coefficient of reactivity was about  $-24 \times 10^{-4} \text{ } ^\circ\text{C}^{-1}$ . Thus a change of  $3^\circ\text{C}$  in NAT (the largest observed) would require the movement of about 375 g of U-235 (1.7 moles U) from the fuel high-pressure system to the core wall. The same effect would result from the deposition of only 75 g of U-235 (0.3 moles U) from the blanket solution onto the core wall. The latter change could go undetected in the physical inventories, but the former probably could not (see Fig. 12).

The nature of the transient in NAT when uranium on the core tank comes and goes depends on which side it is deposited upon. If uranium were to go from the core tank to the fuel-high-pressure solution, there would be a small drop in NAT immediately, followed by a gradual decrease about four times as large while the uranium mixed into the dump tanks and blanket. If the uranium were to go into the blanket, on the other hand, there would be a large decrease in NAT while the uranium redissolved, followed by little change (because the mean importance in the core and dump tanks is about equal to that in the blanket high-pressure system). The observed transients appear more like the blanket case.

The fuel solution is more susceptible to hydrolytic precipitation in the blanket than in the core because lower concentration favors this form of instability. Table 2 lists the molal concentration of  $\text{SO}_4^{2-}$  in blanket samples during 5-Mw operation and also the fraction of the solubility limit at  $300^\circ\text{C}$  for each metal in its three-component system at the blanket  $\text{SO}_4^{2-}$  concentration. The sum of these fractions is approximately the fraction of saturation, at  $300^\circ\text{C}$ , in the five-component system:  $\text{UO}_3\text{-CuO-NiO-SO}_3\text{-H}_2\text{O}$ .<sup>20</sup> The combined effects of the slow increase in nickel inventory and the low blanket concentrations resulted in an estimated fraction of saturation at  $300^\circ\text{C}$  of 0.98 during the full-power operation that preceded the failure of the core-patch bolt. Since the velocity of the flow was low on the blanket side of the core vessel, temperatures of  $300^\circ\text{C}$  or higher seem quite possible. Therefore, it is believed likely that the melting of the patch bolt was initiated by hydrolytic precipitation of solids in a poorly cooled region. Hydrolytic precipitation was probably not a factor in the NAT rise effect, which was observed when the fraction of saturation was quite low.

---

<sup>20</sup>W. L. Marshall and J. S. Gill, Journal of Inorganic and Nuclear Chemistry, 22(1) 115-132 (1962).

The most likely explanation of the NAT rise at high powers is boiling deposition on the blanket side of the core tank. This seems possible because of the low velocities and the relatively high power density which existed in the blanket after the failure of the core patch. Although not conclusive, the reduction of pressure to 1250 psig gave some indication of lowering the power at which the NAT rise was encountered. In two experiments at the low pressure, the NAT rose between 0.5 and 0.8°C at 4 Mw, which it had not done at higher pressures.

It does not appear possible that heavy-liquid-phase separation occurred in run 25. The minimum acid/sulfate ratio in the solution during the run was 0.34, at which level the minimum temperature for heavy-liquid-phase separation (in concentrated solution) is at least 10°C above the reactor pressurizer temperature of 308°C which existed during 1400-psig operation.<sup>21</sup> An even larger margin was provided by the acid addition which raised the acid/sulfate ratio above 0.4. This high acid level existed at the time of the experiment at 1700 psig, and provided a margin of more than 10°C above the pressurizer temperature even at that high pressure. A very large margin existed during the experiments at 1250 psig. The only way in which heavy liquid phase could have formed would have been if there had been a considerable lowering of the acid/sulfate ratio in the concentrate during the boiling process. This is believed to be unlikely.

The addition of copper sulfate was to test whether or not the reduced concentrations of radiolytic gas would suppress the NAT rise phenomenon, or whether the increased copper concentration would aggravate some tendency to fuel instability. (The estimated fraction of saturation, at 300°C, with respect to hydrolytic precipitation was raised to 0.96 by the addition.) There was no noticeable effect of the addition. (It should be pointed out that in any high-temperature regions, the enhanced effectiveness of the copper would result in very low radiolytic-gas partial pressures both before and after the copper addition.)

#### Precipitation in Blanket Samples

Solids were observed in two of the blanket solution samples after they had been stored in the analytical hot cells for several hours. These samples were taken at 288 hours and 5.0 Mw and 384 hours and 5.1 Mw. Both of the copper inventories computed from the copper concentrations in these samples were low. Spectrographic analysis of solids from the first sample was: Cu, very strong; Fe, moderate; U, Zr, Ni and Mo, sought but not found. Only traces of Ag, Al, Ca, Cr and Mn were observed. The solids from the second sample were not analyzed.

---

<sup>21</sup>W. L. Marshall et al., HRP Prog. Rep. Aug. 1 to Nov. 30, 1960, ORNL-3061, p 50.

### Corrosion

Although it appears from the inventories of the other fuel constituents that 96% of the fuel solution was carried over from run 24 to run 25, the initial inventories of nickel in run 25 were 0.9 mole higher than would be calculated for 96% carryover. This increase in nickel before the initial run 25 inventories may be the result of corrosion film formation on the fresh stainless steel surfaces in the new letdown heat exchanger and elsewhere. An alternative explanation is corrosion in the low-pressure system between runs, but this was never significant in earlier shutdowns of longer duration.

In the part of run 25 before the addition of acid, the average corrosion rate of stainless steel, based on the total surface area in the high-pressure systems, was 0.94 mpy. This rate was calculated from the rate of change of the ratio Ni/Cu in the physical inventories shown in Fig. 9. In the latter part of the run, the corrosion rate was 2.4 mpy. As shown in Fig. 9, the increase in corrosion rate apparently coincided with the increase in the free acid inventory.

The corrosion rates of 347 stainless steel for all HRT power runs with good solution-composition data can be described by a straight line when the logarithm of the rate is plotted against the molar ratio  $H_2SO_4/U$ . For eleven periods included in runs 17 through 25, the following equation was calculated:

$$\log_{10} (\text{mpy}) \pm 0.4 = -2.028 + 1.803 \left( \frac{H_2SO_4}{U} \right) .$$

This line and the data on which it was based are shown in Fig. 13. The value shown for the ratio  $H_2SO_4/U$  in run 17 was the average of 38 core solution samples at an average power level of 3.3 Mw. Since the ratio at zero power was 0.5 to 0.7 and since the ratio was greater at higher power levels, it would have been more accurate to use a time-and-power-weighted ratio for run 17. This analysis was not performed, however, and would probably result in only a small correction to the average ratio that was used. The values for the ratio  $H_2SO_4/U$  in the other periods were the average ratios based on solution inventories during stable operation. There was a small tendency for the points representing more megawatt-hours of operation to be located above the calculated line.

UNCLASSIFIED  
ORNL-LR-DWG 63822

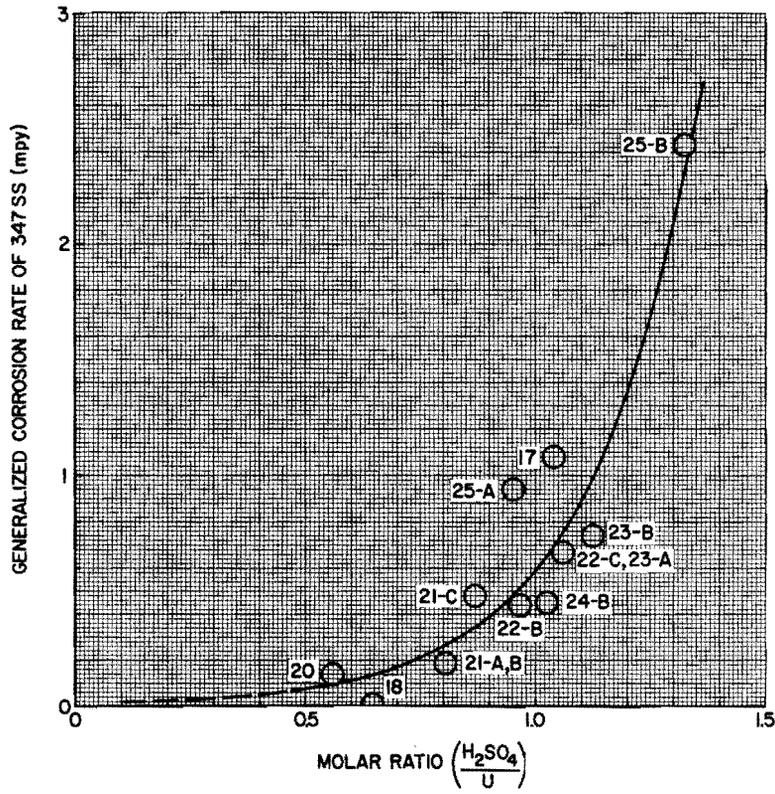


Fig. 13 Corrosion Rate of 347 Stainless Steel vs. Acid Level in HRT Runs 17 through 25

APPENDIX

The following is a list of reports which describe the operation of the HRT from the first power run to the last:

J. R. Engel et al., Summary of HRE-2 Run 13 (Initial Power Operation), ORNL CF-58-10-115 (Oct. 29, 1958).

J. R. Engel et al., Summary of HRT Run 14, ORNL CF-59-6-96 (June 8, 1959).

J. R. Engel et al., Summary of HRT Run 16, ORNL CF-60-4-4 (April 8, 1960).

P. N. Haubenreich et al., Summary of HRT Run 17, ORNL CF-60-8-151 (Aug. 1, 1960).

H. F. Bauman et al., Summary of HRT Run 18, ORNL CF-60-8-152 (Aug. 26, 1960).

H. F. Bauman et al., Summary of HRT Run 19, ORNL CF-61-3-86 (Mar. 16, 1961).

H. F. Bauman et al., Summary of HRT Run 20, ORNL CF-61-7-91 (July 19, 1961).

P. N. Haubenreich et al., Summary of HRT Run 21, ORNL-TM-42 (Oct. 10, 1961).

H. F. Bauman et al., Summary of HRT Runs 22, 23 and 24, ORNL-TM-106 (Mar. 6, 1962).

J. R. Engel et al., Summary of HRT Run 25, ORNL-TM-173 (this report).



Internal Distribution

- |                       |                                      |
|-----------------------|--------------------------------------|
| 1. C. F. Baes         | 44. J. A. Lane                       |
| 2. H. F. Bauman       | 45. C. G. Lawson                     |
| 3. S. E. Beall        | 46. R. A. Lorenz                     |
| 4. M. Bender          | 47. M. I. Lundin                     |
| 5. A. M. Billings     | 48. R. N. Lyon                       |
| 6. A. L. Boch         | 49. H. G. MacPherson                 |
| 7. E. G. Bohlmann     | 50. W. D. Manly                      |
| 8. S. E. Bolt         | 51. W. L. Marshall                   |
| 9. R. B. Briggs       | 52. J. F. McBride                    |
| 10. J. R. Brown       | 53. H. F. McDuffie                   |
| 11. J. R. Buchanan    | 54. A. J. Miller                     |
| 12. W. D. Burch       | 55. R. L. Moore                      |
| 13. R. H. Chapman     | 56. C. S. Morgan                     |
| 14. E. L. Compere     | 57. A. M. Perry                      |
| 15. L. T. Corbin      | 58. M. L. Picklesimer                |
| 16. F. L. Culler      | 59. H. B. Piper                      |
| 17. D. G. Davis       | 60. J. L. Redford                    |
| 18. R. J. Davis       | 61. S. A. Reed                       |
| 19. O. C. Dean        | 62. D. M. Richardson                 |
| 20. D. M. Eissenberg  | 63. M. W. Rosenthal                  |
| 21. J. R. Engel       | 64. H. C. Savage                     |
| 22. J. L. English     | 65. H. W. Savage                     |
| 23. C. Feldman        | 66. A. W. Savolainen                 |
| 24. D. E. Ferguson    | 67. D. Scott, Jr.                    |
| 25. A. P. Fraas       | 68. C. H. Secoy                      |
| 26. C. H. Gabbard     | 69. M. D. Silverman                  |
| 27. W. R. Gall        | 70. M. J. Skinner                    |
| 28. J. P. Gill        | 71. B. A. Soldano                    |
| 29. J. C. Griess      | 72. I. Spiewak                       |
| 30. W. R. Grimes      | 73. J. A. Swartout                   |
| 31. R. H. Guymon      | 74. A. Taboada                       |
| 32. P. A. Haas        | 75. P. F. Thomason                   |
| 33. P. H. Harley      | 76. M. Tobias                        |
| 34. R. J. Harvey      | 77. D. B. Trauger                    |
| 35. P. N. Haubenreich | 78. W. E. Unger                      |
| 36. J. W. Hill, Jr.   | 79. A. M. Weinberg                   |
| 37. E. C. Hise        | 80. O. O. Yarbro                     |
| 38. H. W. Hoffman     | 81. H. E. Zittel                     |
| 39. G. H. Jenks       | 82-83. Central Research Library      |
| 40. P. R. Kasten      | 84-86. Y-12 Document Ref. Section    |
| 41. G. W. Keilholtz   | 87-89. Laboratory Records Department |
| 42. M. J. Kelly       | 90. IRL - Record Copy                |
| 43. J. O. Kolb        | 91-92. RD Library, Bldg. 9204-1      |

External Distribution

- 93. R. E. Pahler, AEC, Washington
- 94. F. P. Self, AEC, ORO
- 95-109. Division of Technical Information Extension
- 110. Research and Development Division, ORO
- 111-112. Reactor Division, ORO

