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ORNL-TM-106

SUMMARY OF HRT RUNS 22, 23 AND 24

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Reactor Division

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H. F. Bauman, J. R. Buchanan, J. R. Engel,
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ABSTRACT

In runs 22, 23 and 24 the HRT was operated with downward flow through the core at powers up to 5 Mw. Because of leakage past the lower core tank patch, uranium could not be prevented from entering the blanket, so the concentration was deliberately kept above 1 g U/kg D_2O. The core power was about 0.6 of the total.

Nuclear power fluctuations were larger than with upward core flow, and the cause was investigated intensively in run 22 at powers up to 1.8 Mw. Run 22 was terminated after 778 hours when a leak developed in a fuel feed pump weld. In run 23 the power was raised stepwise to 5 Mw. This run was terminated after 794 hours by the removal of the fuel, containing the fission and corrosion products accumulated in 10,028 hours of operation. Fuel replacement required three days. Run 24 operations were hampered by a leak which developed between the feed and letdown streams in the fuel letdown heat exchanger. The run was terminated after 546 hours, when a crack developed in a fuel feed line tee and leaked solution into the reactor cell.

The changes in the core flow and heat transfer were evidently effective in eliminating regions of high temperatures and uranium deposition. In all the operations, including more than 100 hours at 5 Mw, there was never any indication of fuel instability. Previous operation at the same pressure (1400 psig) and core mean temperature had resulted in detectable instability at lower powers.

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INTRODUCTION

Following run 21 the HRT was shut down for maintenance and for modification of the core flow pattern. The top five diffusor screens were removed from the core and the fuel circulating pump connections were modified to reverse the direction of flow through the core. The purpose of these changes was to reduce the possibility of hot spots in the core, particularly on the core wall.[1] The purpose, when operation was resumed, was to determine if full-power operation without fuel instability or further damage to the core tank was now possible. System conditions were chosen to improve the chances of trouble-free operation. The reactor pressure was limited to below 1450 psig, so that the boiling point of the fuel solution was well below the minimum temperature for second-liquid-phase formation. The core average temperature was kept below 270°C so that the maximum solution temperature expected in the core should not exceed 290°C,[2] well below the boiling point. As an aid in cooling the core wall, the blanket was operated at 230°C. (The core average was 260°C, so it was necessary to separate the core and blanket steam systems.)

This report covers the period between the end of run 21 and the start of run 25. The summaries of three runs are presented here in a single report, rather than in individual reports as has been the practice, because runs 22, 23 and 24 were closely related both in time and in experimental program. Runs 22 and 23 were extensions of a single experimental program, the stepwise approach to full power to test the efficacy of the core modifications. The two runs were separated by a brief shutdown to replace a fuel feed pump. The run designation was changed from 23 to 24 when the fuel charge was replaced. There was little interruption in the operation; the reactor was not drained during the replacement, and internal recombination experiments started in run 23 were continued in run 24.

REACTOR ALTERATIONS, OBSERVATIONS AND MAINTENANCE

From February 20 to September 30, 1960, the reactor was shut down for alterations, observations and maintenance. A summary of this work follows; detailed descriptions of the work are available in HRP Progress Reports.[3,4,5]

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[1] I. Spiewak et al., HRP Quar. Prog. Rep. July 31, 1960, ORNL-3004, p 29-32.

[2] Ibid.

[3] S. E. Beall et al., HRP Quar. Prog. Rep. Apr. 30, 1960, ORNL-2947, p 3-14.

[4] S. E. Beall et al., HRP Quar. Prog. Rep. July 31, 1960, ORNL-3004, p 1-9.

[5] S. E. Beall et al., HRP Prog. Rep. for Period From August 1 to November 30, 1960, ORNL-3061, p 1-14.

Screen Removal, Core Repair and Flow Reversal

The first step in the reactor alterations was removal of the upper five diffuser screens in the core entrance section. Corrosion had already loosened these screens from the core wall so that they could be pulled free; the screens were picked up one at a time with a special torch-manipulator and cut into 1-1/2-in. wide strips under water. The strips were then fished out into a container with hook tools. The removal of some 145 screen pieces was completed by April 30. Meanwhile, plastic impressions were made of the core holes to aid in fabrication of the plugs.

From May 13 to 18, the core was wire-brushed, also under water, to remove any uranium-bearing scale from the inner surface of the core wall.

On May 26, the first plug was installed in the lower hole. However, the plug did not fit well, and in attempting to remove the plug the toggle bolt galled. By June 16, the lower plug had been removed, by partially sawing and breaking the bolt.

On June 28, the upper hole was sawed to a circular shape with a 1-5/8-in. hole saw. Most of the wall segment removed was retained successfully in the saw and was subsequently examined by the Post-Irradiation Evaluation Group of the Metallurgy Division.[6] The examination indicated that the hole had been formed by melting of the Zircaloy in a rapid temperature excursion of 2 to 5 seconds' duration. The upper hole was reamed to a conical taper, and the upper plug was successfully installed on August 5. Meanwhile, on July 26, a new plug was installed in the lower hole.

The leakage rate through the plugged holes was measured; the leakage through the lower hole, in pounds of water per minute, was equal to five times the square root of the head in inches of water. The leakage through the upper hole was too small to be measurable.

Temporary piping for flow reversal, including a screen filter, was installed on the fuel circulating pump, so that the debris from the screen-cutting and core-brushing could be back-flushed into the filter and removed. Two flushes of the core system were completed by September 2. By September 20, a new fuel circulating pump and a new multiclone were installed with permanent piping to reverse the direction of flow in the fuel high-pressure loop.

Core Observations

Several observations, measurements and photographs were made of the core during the maintenance period.

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[6] F. W. Cooke and M. L. Picklesimer, HRP Quar. Prog. Rep. July 31, 1960, ORNL-3004, p 9-17.

On March 25, the interior of the core from the equator downward was systematically photographed, using the Omniscope. On June 25, following the core-brushing, the same core area was photographed again. The brushed surface had a shiny metallic appearance.

On August 3 and 4, ultrasonic measurements were made of the core-wall thickness. Excluding regions of local damage, the thickness was measured to be 15 to 20 mils less than when the core was first installed.[7]

On September 29, a detailed examination of sections of the core interior was made, using the Omniscope and a Questar telescope for magnification. The surface of pits in the core wall appeared bright and smooth. A complete report of the examination, with photographs, appears in the November 30 Progress Report.[8]

Maintenance

In addition to the work in the core, considerable general maintenance was performed. The replacement of components which failed during run 21, including the fuel feed pump west head and four process valves, is described in the Run 21 Summary Report.[9] The major additional jobs were installation of a new low-pressure transfer line and replacement of the fuel primary and secondary recombiners and the thermocouples on one of the fuel pressurizer heaters. The blanket feed pump east head was replaced during the light water testing. These installations are discussed individually in subsequent sections of this report.

OPERATIONS

Preliminary Operations

On October 2, the reactor was filled with light water (steam condensate) for a shakedown run. The high-pressure systems were hydrostatically tested at 2700 psig. The light water was then circulated at 1400 psig and 260°C core average and 230°C blanket average temperature. The chemical plant was placed on stream to help remove any solids remaining in the system from the core repair operation.

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[7] S. E. Beall et al., HRP Quar. Prog. Rep. July 31, 1960, ORNL-3004, p 18-19.

[8] S. E. Beall et al., HRP Prog. Rep. for Period From August 1 to November 30, 1960, ORNL-3061, p 3-11.

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

Mixing Experiment

Starting October 10, a mixing experiment was performed, using chromic acid as a tracer, to determine the rate of back-mixing through the imperfectly-plugged lower core hole.

Samples were taken to establish base Cr concentrations for the core and blanket regions. A chromic acid solution containing 280 g Cr was added to the fuel system, several samples were taken to make sure that the Cr remained in solution, and an additional 1295 g of Cr was added. The steady-state blanket-to-core concentration ratio was found to be 0.075 for a blanket purge rate of 6 lb/min and 0.222 for 3 lb/min, equivalent to back-mixing rates of 0.4 and 0.8 lb/min, respectively. These data were used to estimate the blanket purge rate required to obtain a given blanket uranium concentration for power operation.

Test of Fuel Recombiners

Following the mixing experiment, the reactor was cooled and depressurized for the replacement of the blanket feed pump west head (the head had failed during the experiment; see page 44) and the multiclone filter, which apparently had become plugged with solids remaining in the system from the core repair operation. While the reactor was shut down, the new recombiners in the fuel low-pressure system were tested by metering O_2 and H_2 directly to the fuel dump tanks. It was found that the efficiency of the primary recombiner was very low at normal fuel dump tank boilup rates, but that recombination was essentially complete in the secondary recombiner (see pages 41-44).

Heavy Water Rinses

Between October 21 and 29, the light water was transferred from the reactor to the waste system, and the reactor was rinsed three times with small quantities of heavy water. The fuel was transferred from the storage tanks to the low-pressure system in preparation for startup.

Pretreatment

On November 2, the reactor was pressurized to 1400 psig and heated to 280°C. Oxygenated D_2O was circulated for 50 hours, followed by a subcritical concentration of fuel solution for an additional 50 hours, to form a passive film on the stainless steel surfaces of the high-pressure system. To minimize hydrolytic precipitation in the blanket, part of the fuel was pumped directly from the blanket dump tank, so that the blanket uranium concentration was quickly raised through the precipitation range (up to about 1 g U/kg D_2O).

Operating Conditions

The reactor power, nuclear average temperature (NAT), pressure, and blanket-to-core concentration ratios are shown in Figs. 1, 2 and 3 for

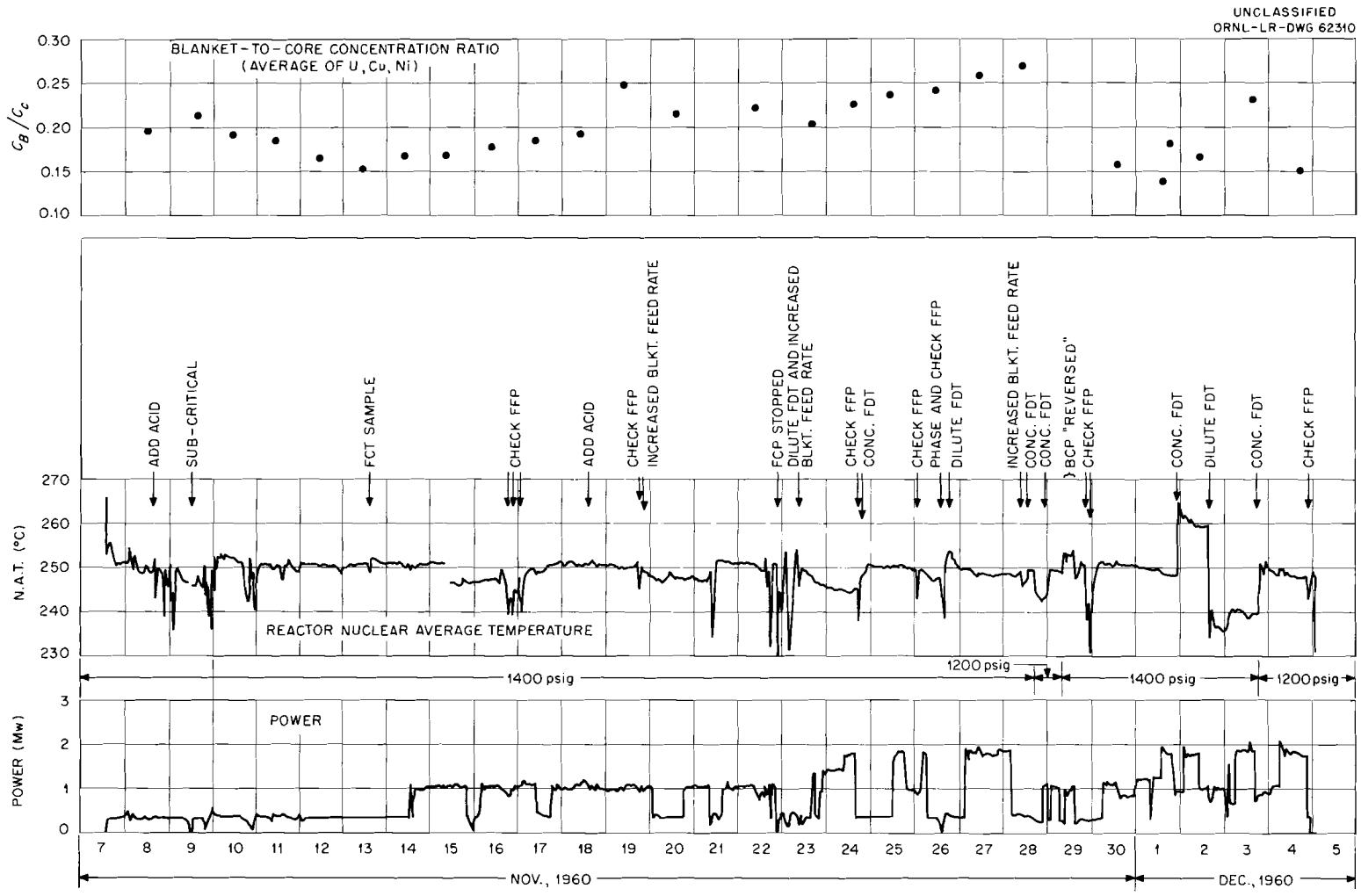

Fig. 1. Chart of Operations for Run 22.

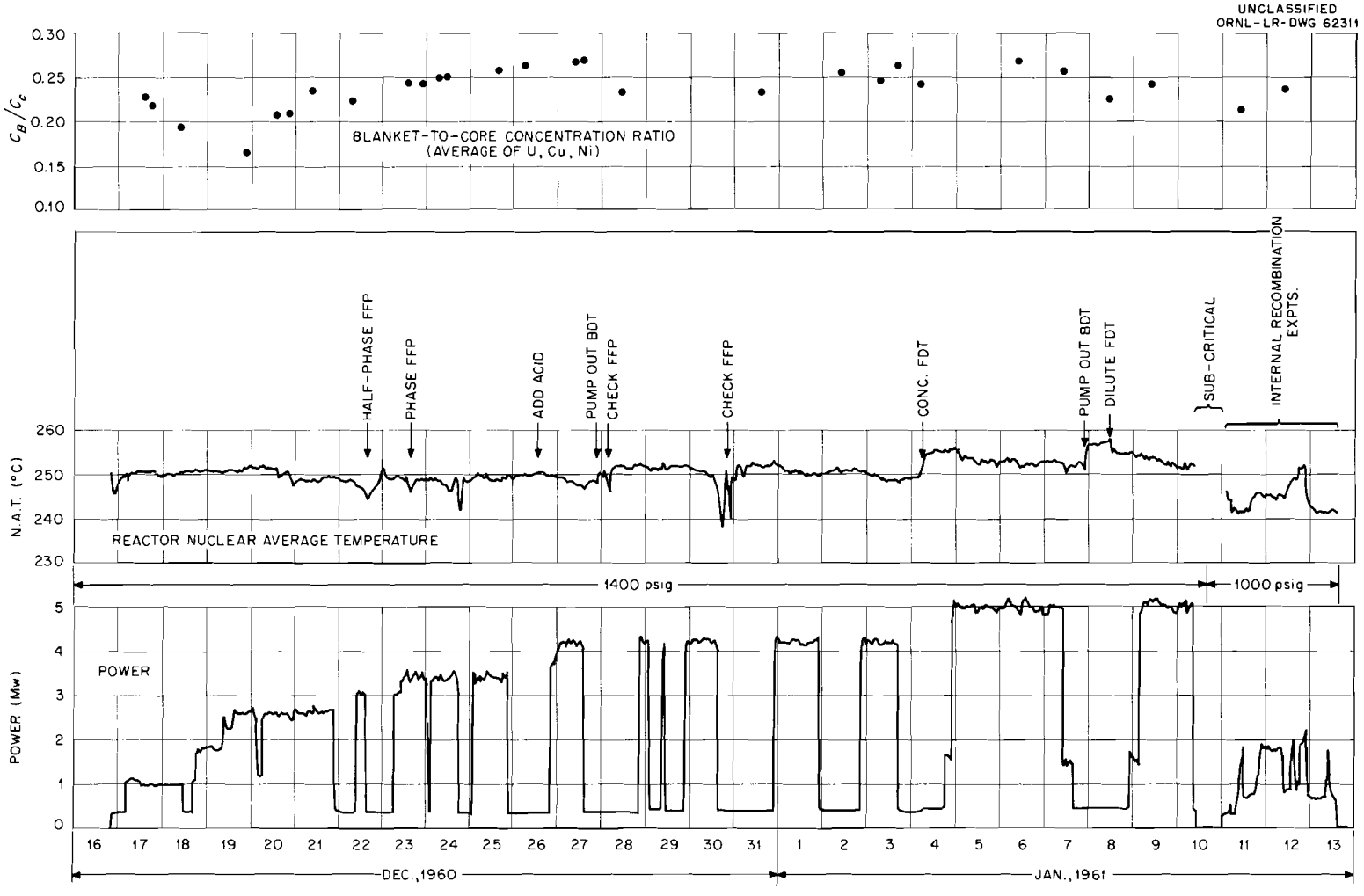

Fig. 2. Chart of Operations for Run 23.

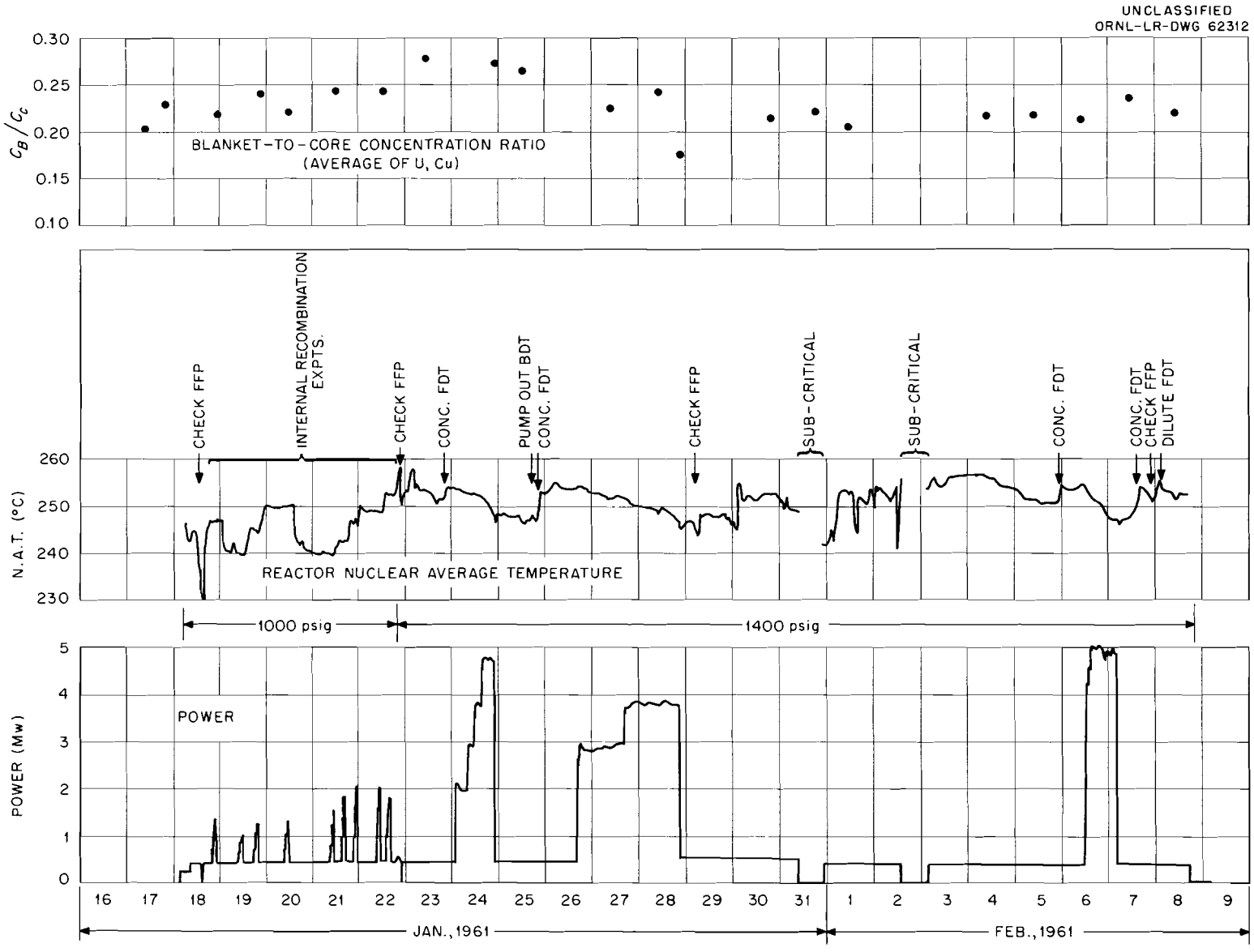

Fig. 3. Chart of Operations for Run 24.

the periods of power operation in runs 22, 23 and 24. The blanket average temperature was held constant at 230°C throughout all three runs. The core average temperature, usually about 260°C, and the reactor pressure, usually 1400 psig, were adjusted as required by the experimental program. The blanket purge rate was adjusted to hold the blanket concentration slightly above 1 g U/kg D_2O.

Since it was intended that the reactor be operated only in the region of fuel stability, the indications of possible instability were observed closely. The reactor high-pressure systems were sampled (usually) daily to detect any loss of uranium from solution. As in previous runs, a limit was set on the permissible change in NAT; the power would be lowered to heat loss if the NAT increased or decreased by 4°C from the temperature at the start of higher power operation. The power would also be lowered if a power pip exceeding 50% of the average power occurred, or if two or more pips exceeding 300 kw occurred in any 8-hr period. During run 22 it was found that power fluctuations that did not appear to be pips caused by instability, regularly exceeded 300 kw and the latter restriction was lifted.

Beginning in run 23, an additional restriction was imposed, based on a statistical method for distinguishing possible power pips from random power fluctuations; i.e., that the power would be lowered to heat loss if the frequency of short, positive, reactor-period fluctuations exceeded a given limit (see pages 71-74). During the stepwise approach to full power, the limit was set for each power level by extrapolation from a statistical analysis of the frequency at the preceding power level. The limit at 5 Mw, for example, was that the number of power fluctuations with a positive period shorter than 8 seconds should not exceed four in two hours.

Power Operation, Run 22

Heat-Loss Power

On November 7, the reactor was made critical by cooling to a blanket average temperature of 230°C and a core average of 260°C. The power was held at heat loss for seven days, while baseline data were obtained for reference during the approach to full power. During this time, sulfuric acid was added to the fuel to bring the core acid concentration to 0.031 molal. Data were taken to determine the ratio of core and blanket temperature coefficients of reactivity. Core and blanket high-pressure system samples were taken daily to provide base inventory data.

Both heads of the fuel feed pump were initially in service; however, the output of the west head became erratic, apparently due to leakage of air into the intermediate system, and from November 9 to 29, only the east head was used (see page 44).

Operation at 1.0 Mw

On November 14, the power was raised to 1 Mw, where it was held, with four interruptions, until November 22.

On November 15, the power was adjusted to various levels between 50 kw and 1 Mw for short intervals while additional data were taken to determine the weighting factors (applied to the core and blanket inlet and outlet temperatures) used in calculating the nuclear average temperature. It was observed that the core inlet and outlet temperatures both decreased as the power was raised, indicating the presence of a region in the core that was at a higher temperature than the core outlet (see pages 51-54).

On November 16, the fuel-circulating-pump purge pump stopped pumping, apparently because the check valves were not seating tightly. The power was lowered to heat loss, and pumping was restored by installing a new drive cam, which gave a faster suction stroke (see page 45). The fuel feed pump rate was readjusted at the same time. On November 17, the power was raised to 1 Mw again; the reactor operated smoothly at this power level for two days. The NAT had decreased by 2.6°C on November 20, apparently because of shifting feed and purge rates. The power was lowered to heat loss, and, since no change in nuclear average temperature (NAT) was observed, after 18 hours, the power was returned to 1 Mw. The power was again lowered to heat loss on the 21st when a blanket pressurizer heater fuse blew, upsetting the system. The blanket purge rate was increased from 3.9 to 4.3 lb/min (to lower the blanket concentration closer to 1 g U/kg D_2O, equivalent to a blanket-to-core concentration ratio of about 0.11), and the core was concentrated to raise the core average temperature to about 265°C; operation at 1 Mw was continued.

Throughout the foregoing power operation at 1 Mw, it was obvious that the magnitude of nuclear power fluctuations was about double that in previous runs with upward flow through the core. It seemed likely that this was a direct result of the changed core flow pattern; however, the phenomenon was studied intensively throughout run 22 and succeeding runs (see pages 66-71).

To aid in studying the power fluctuations, the neutron level signal was transmitted to a data logger in the main Laboratory area to obtain a digital, punched-tape record of the reactor power for subsequent computer analysis. Record tapes were made under various operating conditions throughout the runs.

Several experiments were performed to determine whether the reactor power fluctuations were affected by operating conditions. For brief periods the letdown valve was throttled manually, the pressurizer heaters were shut off, the steam withdrawal rate was varied, the blanket feed pump and fuel feed pump were stopped, and the fuel circulation rate was lowered (by changing the pump frequency from 60 to 40 cps). None of the above changes resulted in a significant change in the power fluctuations.

The study of the power fluctuations indicated that they were random fluctuations rather than "pips," and since there were no indications of fuel instability from observations of the inventory and nuclear average temperature, it was decided to continue the approach to full power operation. Before the power was raised, however, another aspect of operation with downward flow through the core was investigated; namely, the effectiveness of heat removal from the core during a stoppage of the fuel circulating

pump. On November 22, with the reactor power at 1.0 Mw, the circulating pump was stopped. Core inlet and outlet temperatures dropped rapidly when the pump was stopped, but did not cross for about 20 minutes, indicating that downward flow through the core persisted for some time. When the pump was restarted, the inlet and outlet temperatures crossed again, indicating that the flow had either stopped or changed direction while the pump was off. There were no surges in nuclear power caused by stopping and starting the circulating pump.

Following the pump-stop experiment, the blanket purge rate was again increased, from 4.3 to 5.3 lb/min, because the blanket-to-core concentration ratio was still high (0.22).

Operation at Higher Power

After establishing a base NAT at heat-loss power, the power was raised to 1.4 Mw on November 23. Since the fuel feed pump rate appeared to be sensitive to fuel dump tank pressure, a controller was installed to control the dump tank at a constant pressure (usually 18 psia) by throttling the off-gas valves.

After about 24 hours the power was raised to 1.8 Mw; experiments at this power level continued over the next four days.

During this period the reactor power was lowered to 1.0 Mw or to heat loss several times; once to concentrate the fuel dump tank to raise the NAT above 250°C, twice because power fluctuations exceeding the limit of 300 kw were observed, and once because the NAT had declined as a result of a drift in the fuel feed rate.

The relative size of the power fluctuations was observed to increase with power; however, an analysis of the magnitude and frequency of the power fluctuations indicated that even those exceeding 300 kw were random fluctuations rather than pips. For this reason, and since no signs of fuel instability were observed, the 300-kw pip limit was eliminated.

The blanket-to-core concentration ratio had been drifting upward, from 0.17 on November 15, to 0.27 on November 28, despite two small increases in blanket purge rate (see Fig. 1). A heat balance on the blanket dump cooler indicated that some leakage had developed past the seat of the blanket dump valve (in run 23 this leakage was measured to be about 1 lb/min), which decreased the effective blanket purge. The loading pressure on the dump-valve operator was temporarily increased from about 30 to 50 psig to see if the leakage could be reduced, but no effect was detectable. The normal loading pressure was restored after about 2 hours, and the blanket purge rate was increased from 4.3 to 5.9 lb/min to compensate for the leakage.

Experiments on Power Fluctuations

The power fluctuations were most likely caused by variations in the internal recirculation of fuel in the core as a result of the changed flow pattern. However, another hypothesis was that they were the result

of a variable void volume in the core caused by steam bubble formation on hot surfaces. Since steam bubble formation would be strongly affected by the reactor temperature, pressure and flow rates, a series of experiments was performed in which these variables were changed and the resulting reactor behavior observed.

On November 28, the system pressure was lowered from 1400 to 1200 psig, and the power was raised to 1 Mw to investigate the effect of pressure on the power fluctuations; no effect was noted. The pressure was readjusted to 1400 psig at heat-loss power. The blanket circulation rate was lowered from a nominal 260 gpm to 100 gpm by reversing the direction of pump rotation; at 1 Mw no significant effect on the power fluctuations was noted.

Over the next two days an experiment was performed to determine the effects of core power and core inlet-outlet temperature differential on the power fluctuations. The core power and circulation rates were varied as follows:

Power (Mw)	Core Circulation Rate (gpm)	Core Temperature Differential (°C)
0.8	300	6.7
1.2	460	6.7
1.2	300	10
1.8	460	10

A four-hour digital record of the power was obtained at each condition (see page 68).

On December 2 and 3, the reactor was operated at power levels of 1.0 and 1.8 Mw at core average temperatures of 280 and 240°C. On December 4, the core average temperature was returned to 260°C, the pressure was lowered to 1200 psig, and the reactor was operated at 1.0 and 1.8 Mw. The changes in temperature and pressure had no significant effect on the power fluctuations.

Termination of Run 22

At 2100 on December 4, the cell air activity began rising, reaching a peak of 40 mr/hr at the cell air monitor. A fuel sump sample indicated that fresh fission product activity was present in the cell. The run was terminated so that the source of activity could be located.

In run 22, the reactor operated 778 hours, was critical 653 hours and generated 511 Mw-hours of heat.

Replacement of the Fuel Feed Pump Head

After the appearance of fission product activity in the cell, indicating a leak in the primary containment, the reactor high-pressure system was cooled, depressurized, drained and rinsed twice with D_2O condensate. The fuel was stored in the fuel storage tank. Several shield plugs were removed from the top of the cell, and smears of the cell floor and reactor equipment were taken through the dry maintenance shield.[10]

On December 8, smears reading up to 640 mr/hr through 2 inches of steel were obtained from the east head of the fuel feed pump, and inspection disclosed a stained area around the discharge check valve. The pump was observed remotely while pumping against freeze plugs at 1500 psig, but no leakage was visible. However, the surface contamination was ample evidence of leakage, and the pump head was replaced (see page 44).

The leaking head was transferred to the cutting tank of the hot storage pool for examination. The head was pressurized with gas and the leak was located at the weld on the high-pressure side of the discharge check valve.

The cell air monitor piping was decontaminated and reassembled. By December 11, the cell had been resealed; however, it was necessary to locate and repair leaks in the shield roof pans and the cell air monitor piping; an acceptable cell leak test was completed December 14. Meanwhile, the fuel was transferred from the storage tank to the reactor system in preparation for run 23.

Reactor Operation, Run 23

During the 12-day shutdown, run 22 data had been studied in sufficient detail to conclude that the nuclear power fluctuations were not symptomatic of fuel instability (see page 71). Furthermore, the fluctuations were not serious in themselves, because heat capacities and mixing effects smeared them out so that there were no detectable fluctuations in the reactor steam heat removal. Therefore, the stepwise increase in power toward 5 Mw was resumed without delay at the beginning of run 23.

Approach to and Operation at 5 Mw

The reactor was made critical and raised to heat-loss power on December 16, 1960. After waiting several hours to allow all operating conditions to reach steady state, the power was raised to 1 Mw. The system was operated for a few hours at 1.0 and 1.8 Mw to ensure that the reactor behavior was not drastically different from that observed in run 22.

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[10] I. Spiewak *et al.*, HRP Quar. Prog. Rep. Jan. 31, 1960, ORNL-2920, p 13-14.

Starting on December 19, the reactor power was raised in steps of 0.8 Mw until the full power of 5 Mw was reached. However, each major step was made in two parts, with four hours' operation at 0.4 Mw above the previous level to be certain that no gross instabilities had appeared. Operation at each major level was continued until four pairs (fuel and blanket) of high-pressure samples had been obtained and analyzed for chemical studies of the fuel stability. Detailed records of the nuclear behavior of the reactor were made for several hours at each power level. The linear neutron level was recorded in digital form on punched paper tape at the rate of one value per second. These records were subsequently analyzed, using a high-speed electronic computer. In addition to the above, a continuous record of the fine structure of the neutron level was obtained with a Sanborn recorder and the reactor period was continuously recorded on a Brown recorder. These data were treated manually (see pages 71-74) to provide a basis for judging the operational stability before the next power increase was made and to establish criteria for reducing the power under unstable conditions.

The required operating time at 2.6 Mw (the first major step above 1.8 Mw) was completed on December 21. On one occasion the power was reduced to 1 Mw for four hours because of the frequency and size of the power oscillations. However, there was no evidence of instability from the nuclear average temperature, so the power was returned to 2.6 Mw. At the end of this period the power was reduced to heat loss for 12 hours and a pair of samples was taken.

On December 22, the power was raised to 3 Mw as the first part of the increase to 3.4 Mw. After only 4 hours at this power the NAT had dropped 2.5°C. Although there was evidence that the pumping rate of the fuel feed pump had decreased, the power was lowered to heat loss to be sure that the temperature decline was not caused by fuel instability. Since the temperature decline continued at heat-loss power, the pumping rate was measured, found to be low and adjusted to the proper value. The power was subsequently raised to 3.0 and then to 3.4 Mw to accumulate data. There were two interruptions in the operation at 3.4 Mw. The first of these occurred when several short reactor periods were experienced in a short time interval. It was shown, however, that there was no significant deviation from the normal statistical distribution of the periods and operation was resumed. The second interruption was due to a malfunction of the fuel feed pump which resulted in one remote head being taken out of service. At no time did the NAT give any indications which could be attributed to fuel instability.

While the reactor was at heat-loss power, 4.77 moles of H_2SO_4 was added to the fuel solution to reduce the possibility of hydrolytic precipitation in the solution. The acid addition was required because of the buildup of nickel from the corrosion of stainless steel (see page 85).

Operation at the next higher power level (4.2 Mw) was interrupted five times before the desired amount of data could be accumulated. Of these interruptions two were the result of apparently excessive frequencies of short reactor periods, one was due to a brief electric power failure, one to malfunction of the fuel feed pump and one was planned. Statistical analysis of the frequency of occurrence of the periods failed to show any

abnormal behavior. Three pairs of high-pressure samples were taken at heat-loss power during this period in addition to the four pairs at 4.2 Mw. A pair of samples was also taken at heat-loss power at the end of the period of 4.2 Mw-operation.

Before attempting to operate the reactor at 5 Mw, the core average temperature was raised from about 260°C to about 270°C by concentrating the solution in the fuel dump tanks. This action was taken to improve the internal recombination of radiolytic gas, thus reducing the possibility of gas-bubble formation at the higher power.

The increase to 5 Mw was made on January 4, 1961. The power was first held at 1.5 Mw for three hours to observe the core-blanket power distribution and then raised to 5 Mw. Operation at this level was interrupted on January 7 because of a continued slow decline in the nuclear average temperature (4°C in 60 hours). However, the temperature behavior on lowering the power indicated that the loss was not due to fuel instability.

While at heat-loss power it was found that increasing leakage through the blanket dump valve had caused the temperature loss by allowing a gradual buildup of fuel in the blanket dump tanks. The air loading on the blanket dump valve was increased (see page 46) to reduce the leakage, and the water inventory in the blanket dump tanks was lowered by about 600 lb to reduce the fuel holdup at that location.

Operation at 5 Mw was resumed on January 9 and continued for 29 hours. At that time, the diaphragm in the east head of the fuel feed pump failed and it was necessary to make the reactor subcritical until the standby head could be put in service. Since the nickel concentration in the fuel was close to the limit for hydrolytic precipitation, it was planned to carry out a series of internal recombination experiments and then charge fresh fuel into the reactor. The reactor had operated for 89 hours at 5 Mw and, although only three pairs of high-pressure samples were obtained at this power, it was felt that these samples, along with the temperature and power data, provided adequate evidence of fuel solution stability.

Internal Recombination Experiments

The east head of the fuel feed pump was isolated with freeze plugs, the west head was put in service and the reactor conditions were adjusted for the internal recombination experiments. The system pressure was lowered to 1000 psia to permit gas-bubble formation at nuclear powers of 2 to 3 Mw, and the fuel feed rate was adjusted to give a nominal letdown rate of 12 lb/min instead of the usual 13 lb/min. A set of five experiments was performed at these conditions with various core average temperatures. For each experiment the power was raised slowly until radiolytic gas bubbles appeared in the high-pressure system.[11] The power was then

[11] Since the bubbles formed in the core, their appearance was detected by a sharp drop in critical temperature; there was no letdown of radiolytic gas to the low-pressure system.

lowered about 500 kw to let the temperature recover and raised again to check the bubble point. In one experiment the power was held at 500 kw below the bubble point for 12 hours to see if there was any effect of relatively short-term power operation on the bubble point. The details and results of these and other recombination experiments are discussed on page 80 et seq.

Fuel Replacement

With the recombination experiments completed, the reactor was made subcritical in preparation for replacing the fuel. The high-pressure systems were allowed to cool to about 180°C (the highest temperature which could be maintained by supplying building steam to the primary heat exchangers) but the circulation was continued and the system pressure was raised to 1400 psia. The old fuel was accumulated in the fuel dump tanks and transferred to the fuel storage tanks. The fuel was transferred with about 200 lb of D_2O, followed by two D_2O transfers, of about 100 lb each, to rinse the dump tanks.

As soon as the old fuel charge had been isolated, the reactor high-pressure systems were heated to 280°C as a pretreatment to fortify the metal oxide film for the next run. While at this temperature, a fresh fuel charge was added to the fuel dump tanks. Pump-up of the fresh fuel to the high-pressure systems was started on January 16, 1961, 80 hours after the reactor was made subcritical with the old charge.

The total amount of material transferred to the fuel storage tanks with the old fuel charge was 463 lb, of which 410 lb was heavy water and the remainder, solute. Samples taken after the transfer indicated that 190 g of uranium was left in the reactor system, out of a circulating inventory of 9.1 kg. The book inventory of uranium at the time was 9.958 kg (83.16% U-235), and it was assumed that the fraction of the book inventory of the other fuel constituents remaining in circulation was the same as that for the uranium (i.e., 1.91%). The new fuel charge contained 7.53 kg of uranium (93.14% U-235) in 204 lb of heavy water. The fuel addition contained only about half of the copper that was required for full-power operation; the remainder was added during the course of the subsequent recombination experiments.

Reactor Operation, Run 24

With the addition of the new fuel charge, the HRT run designation was changed to run 24. The high-pressure systems were pretreated, at 280°C, for 24 hours with oxygenated D_2O and 24 hours with a subcritical concentration of fuel solution. The reactor was made critical at 0405 on January 18, 1961, and the experimental program was resumed.

Internal Recombination Experiments

The first experiments with the fresh fuel charge were two series of internal recombination tests. Since the fuel contained essentially no fission products or soluble corrosion products, an opportunity was afforded

to measure the effective recombination rate without the effects of these variables. Both series of tests were carried out at a system pressure of 1000 psia and a variety of core average temperatures.

The first series, consisting of four experiments, was performed with only about half the normal copper concentration and with a relatively low free acid concentration. Upon completion of these tests, 8.01 moles of copper as $CuSO_4$ and 6.98 moles of H_2SO_4 were added to the system in 106 lb of heavy water. A series of five recombination experiments was then performed in the same core average temperature range as the first set. These experiments, along with those performed in run 23, provided information about the effects on recombination of fission and corrosion products, acid concentration and copper concentration.

The run 24 recombination experiments were started on January 18 and completed on January 22, 1961. The general procedure for these experiments was the same as that used in run 23. The results of all of the internal recombination experiments are discussed on page 80 et seq.

Power Operation

At the conclusion of the recombination experiments, system conditions were adjusted to those at which the reactor had been operating before the fuel replacement: core temperature, 270°C; blanket temperature, 230°C; and pressure, 1400 psig. On January 24, the power was raised in 1-Mw steps (2 hours at each step) to full power. After 7 hours at 5 Mw, the power was lowered because the NAT was decreasing. There was no recovery upon lowering the power; the NAT finally leveled off 7°C below the starting point. No explanation was found for the decrease; uranium had not accumulated in the blanket dump tanks, and the feed cooler temperatures indicated that the feed rate had remained steady. The fuel dump tanks were concentrated to bring the NAT back up, and, on January 26, the power was again raised, this time to 3 Mw. After 24 hours at 3 Mw with no unexplained NAT changes, the power was raised to 4 Mw. In 20 hours at this power, the NAT changed little, then it began to decrease and lost 4°C in 8 hours. The power was lowered to heat loss and the NAT leveled off 7°C below the NAT at the beginning of the 3-Mw operation. A measurement of the fuel feed rate showed that it was low, but not enough to account for all of the NAT decrease. Further attempts to operate at high power were suspended until the cause for the anomalous changes in NAT could be investigated.

Investigation of Reactivity Loss

The loss of reactivity had been accompanied by a decrease in the calculated physical inventory of fuel solute, implying either a real loss of solute from circulation or increased error in some factor used in the inventory calculation.

One suspected error was in the calculation of the solute in the fuel dump tanks. Over one-third of the solute was in these tanks, and the calculation of the amount there depended on an assumption. It was assumed that the ratio of the dump tank concentration to the core concentration

was the same as the ratio of the letdown rate to the feed rate, which was inferred from the pumping rates of the several feed and purge pumps. The actual ratio of letdown to feed at the core loop would be different from that obtained from the pumping rates if there were short-circuiting of solution from the feed stream into the letdown stream in the fuel letdown heat exchanger. On January 29, an experiment was conducted to check this possibility. The dump tank weight was reduced from 550 to 300 lb while the feed pump stroke was adjusted to keep the NAT (and the core concentration) constant. The pumping rate was measured to determine the amount by which it had been reduced.

While the results of this experiment were being analyzed, operations were conducted to recover any solute which might, by some unknown way, have accumulated in the various low-pressure tanks not normally in use. There was no solute recovery from this, nor was there any recovery during 13 hours of subcritical operation.

The dump tank-feed pump experiment indicated that feed solution was leaking into the letdown stream in the letdown heat exchanger at about 1.5 lb/min, a rate sufficient to account for the apparent decrease in reactivity and calculated inventory. Various steps were then taken to confirm this leakage. The feed pump was stopped, and the temperature on the feed line between the letdown heat exchanger and the core loop rose, implying backflow through the line and leakage in the heat exchanger. The experiment of varying the dump tank weight and the feed rate was repeated, and a dump tank sample was taken to compare with the concentration inferred from the core concentration and the assumed letdown-to-feed ratio. Both of these experiments indicated some leakage, but the results were not entirely conclusive. The dump-tank sampling operation, involving transfer of solution through lines normally filled with condensate, could not be relied upon to obtain an absolutely representative sample. The results of the other experiment were compromised because of the behavior of the fuel feed pump. The rate drifted so that there was no fixed relation between controller setting and feed rate. Therefore reliable values for the fuel feed rate during the experiment could not be inferred from the observed controller settings. (There had been trouble with the fuel feed pump collecting air in the intermediate system for some time and efforts to seal points of possible inleakage had not been successful.)

Temperature records for the letdown heat exchanger were used to compute heat balances over a long period before and after the beginning of the reactivity loss. The heat balances indicated increasing leakage. It was decided, therefore, to plan for the replacement of the heat exchanger at the next shutdown. First, however, the fuel stability would be investigated at full power, if the fuel-feed-rate variations could be overcome.

Termination of Run 24

In an effort to remedy the leakage of air into the pump intermediate system, the pulsator housing was replaced on February 2. With the interruption in feed, the reactor went subcritical, but the temperatures were held up with the package boiler. Following this repair, the feed rate

showed less variation, and on February 5 and 6, the NAT and the feed cooler temperatures indicated steady operation for over 24 hours.

The power was raised to 5 Mw on February 6. During the next several hours, the temperatures on the fuel feed cooler indicated that the feed rate was increasing, so the controller was lowered twice to return the rate indication to the original value. The NAT decreased, implying that the feed rate had actually been lowered. Before the situation could be resolved, however, the power was lowered for another reason.

The cell air activity had begun to rise when the power was raised, and after 16 hours it was up by a factor of 20, about twice the previous high in operation at 5 Mw. This could not be accounted for by fissioning of uranium already in the cell from the run 22 leak, so it appeared that there was another leak. The power was, therefore, reduced to heat loss as the first step in a shutdown to locate the leak.

Before the reactor was shut down completely, a test was conducted to help localize the leak. Because of the composition of the activity (predominantly xenon), the dump tanks were suspect. (See page 27.) The dump tank pressure was raised by about 10 psi on February 7 to observe the effect on the leakage. The cell air activity, which had been decreasing since the power reduction, rose slightly. After fresh activity was found in sump samples on February 8, the high-pressure system was diluted, cooled, depressurized and drained. Three rinses with heavy water condensate followed. During one of the rinses, the leak in the letdown heat exchanger was confirmed by freezing the feed line and pumping with the feed pump.

Post-Operation Maintenance

The principal reason for the shutdown was the breach in the primary containment. It was, of course, necessary to locate this leak and correct the condition before power operation could be resumed. There were, however, a number of other malfunctioning components in the cell which did not constitute an operational hazard, but which did limit the quality of certain experimental data that could be obtained from the operating reactor. These included the fuel letdown heat exchanger, the blanket dump valve (PCV-252), the chemical-plant isolation valves (HCV-141 and -142) and both heads of the fuel feed pump. All of these components were replaced while the system was shut down to repair the leak.

The reactor cell was opened to start the actual maintenance work on February 10, and this phase was completed on March 11. Some of the maintenance operations were carried out with the reactor cell dry, working through the dry-maintenance shield, and others were done with the cell flooded. The work was programmed to minimize the amount of flooding water discharged to the waste pond, with the result that there was sometimes a significant time lapse between the start and completion of a particular job. However, it is convenient to report each job under a single heading with reference to the chronological order where necessary.

Location and Repair of Primary System Leak

Since the principal reason for the shutdown was the primary system leak, one of the first operations that was undertaken was the location of the leak. This was done with the cell dry and most of the searching was done using the smearing technique that was developed for locating the feed-pump leak at the end of run 22. However, several supplementary approaches were also used to good advantage.

The activity of liquid samples taken from the fuel and blanket sumps indicated that the leak was probably on the fuel side (west half) of the reactor cell. The nature of the cell air activity prior to the shutdown also provided some information. It was noted that the ratio of xenon to krypton in the cell air was much higher than the normal fission yield. This implied that the leaking fluid had been outside the fissioning region long enough for most of the krypton to be formed and stripped out.[12] Since it seemed reasonable that krypton would be stripped from solution more readily than the iodine, it was tentatively concluded that the leaking material was dump-tank solution--either directly from the dump tanks or from the fuel feed piping.

Still another approach to locating the leak was the use of a high-level (6000 r/hr), beta-gamma radiation probe which could be lowered into the cell for scanning the floor and equipment. The probe was used to look for areas of gross contamination on the cell floor and on structural components that are not normally strong sources of radiation. The probe itself was shielded to make its response directional.

Extensive smearing of the cell floor in the vicinity of the fuel feed pump and the fuel letdown heat exchanger failed to produce any significant indications. A few smears around the blanket feed pump were also negative. The "hottest" smears obtained in the first three days were from around the fuel sump. In the meantime, surveys with the radiation probe had revealed a large source of activity in the direction of the south end of the fuel dump tanks. When the dry maintenance facility (through which all smearing was done) was moved to this location to permit direct viewing of the area, a large, blue-green stain was found on the cell floor between the two fuel dump tanks. A single smear[13] from the stained area read 200 r/hr at 5 in., as opposed to readings of about 10 r/hr, or less, at contact for smears

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[12] Of the radiokryptons having half-lives greater than one hour, 73% is formed either directly or by decay of bromine precursers having half-lives of less than one minute. By comparison, essentially all of the radioxenons with half-lives greater than one hour are formed by iodine precursers with half-lives at least 6.7 hours.

[13] Chemical analysis of the material picked up by the smear showed that it contained uranium.

taken from regions other than the fuel sump. Additional floor smears around the fuel feed valve (HCV-337) revealed that the stain had not come from the only flanged closure in that area.

The exact location of the leak was finally established by looking through a periscope lowered through the dry maintenance facility. A thick deposit of dried salt was found on a 1-in. tee in the fuel feed line (L-107) between the downcomers from the two fuel dump tanks.[14] Figure 4 is a photograph, taken from the top of the cell, showing the salt deposit on the tee. The location of the tee is shown schematically in Fig. 5.

The salt deposit on the tee, and on the floor under it, was flushed away with a solution of hydrogen peroxide (1%) and ammonium oxalate (4%), and rinsed with water. The rinse material, containing 180 grams of uranium, was then jetted from the cell. With the pipe clean, it was possible to see water dripping from the body of the tee when the dump tanks were pressurized. Although it was not possible to see the exact source of the leakage, it appeared to be coming from the body of the tee, rather than from any of the welds. Careful re-examination of the gamma-graphs of all the welds also showed no cause for weld failure. Figure 6 shows the cleaned tee with an arrow pointing to the approximate location of the leak.

It was planned to isolate the leak with freeze plugs rather than attempt to repair it. As may be seen from Fig. 5, the interconnection of the dump tanks through the evaporator legs made it possible to isolate the leak without affecting the homogeneity of either the dump-tank solution or the fuel feed stream. It was also planned to put a clamp around the tee for mechanical support. The only function of the clamp was to prevent complete rupture of the tee while the dump tanks were at high pressure in the event of a dump with the freeze plugs thawed. Since it was necessary to fabricate the freeze jackets and the clamp, the installation of these items was scheduled as the last operation in the maintenance program. The leaking tee was painted with Glyptal to keep the cell flooding water from leaking into the piping during the maintenance period.

The installation of the clamp and the freeze jackets on line 107 was made with the reactor cell flooded. The work was done under conditions of very poor visibility because of increasing turbidity of the water. An attempt was made to improve the visibility by adding flocculating agents [$Al_2(SO_4)$ and NaOH]. This cleared the water but the floc settled in a layer about 2-ft thick in the bottom of the cell. Since the work area was less than 1 ft above the floor, it was necessary to drain some water from

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[14] In the original HRT design this tee had served the transfer line between the dump tanks and storage tanks. However, in a piping revision, which included the installation of a sediment trap in the fuel feed line, the transfer line was moved and a cap was welded over the branch opening of the tee.

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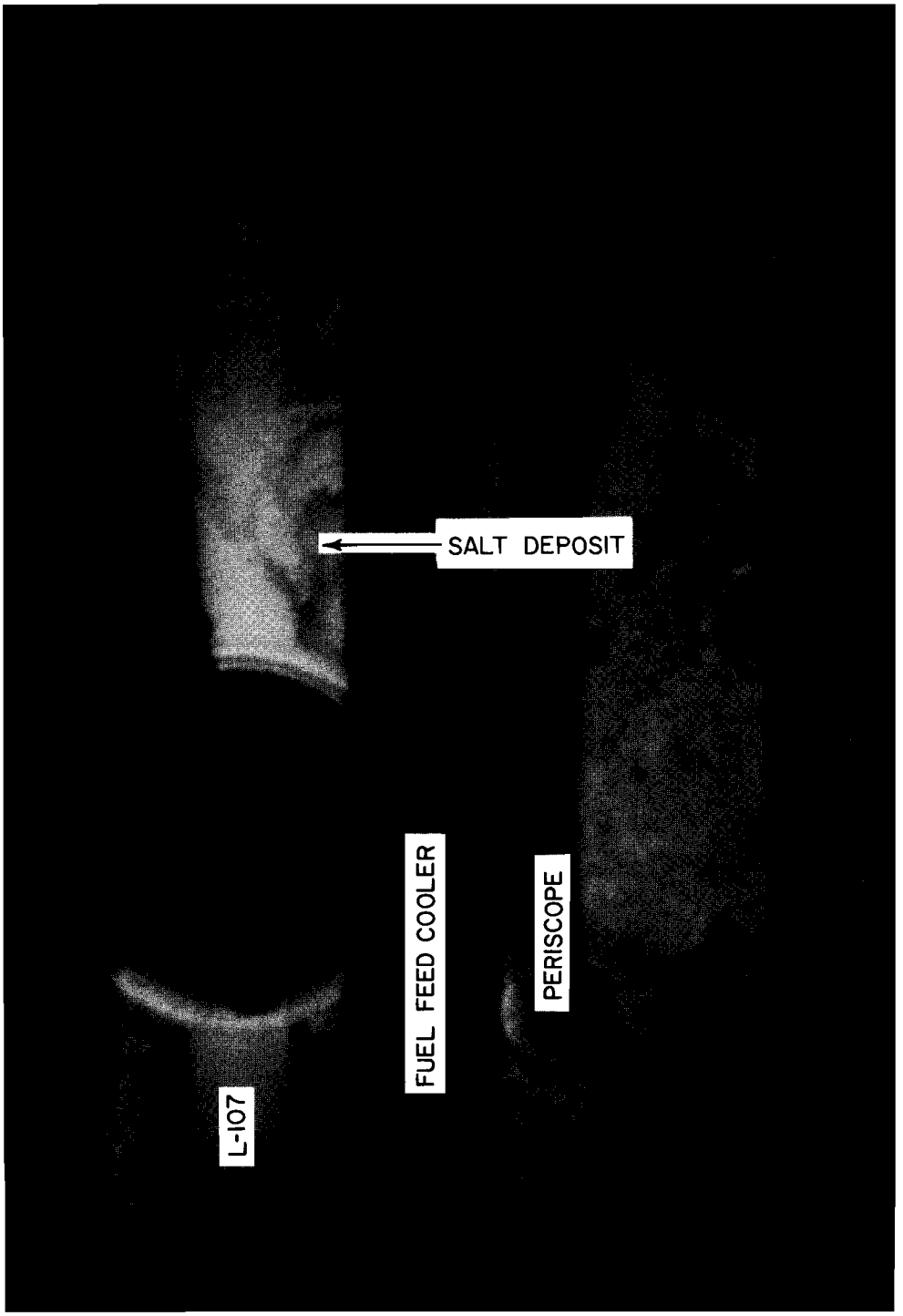

Fig. 4. Leaking Tee in Fuel Feed Line.

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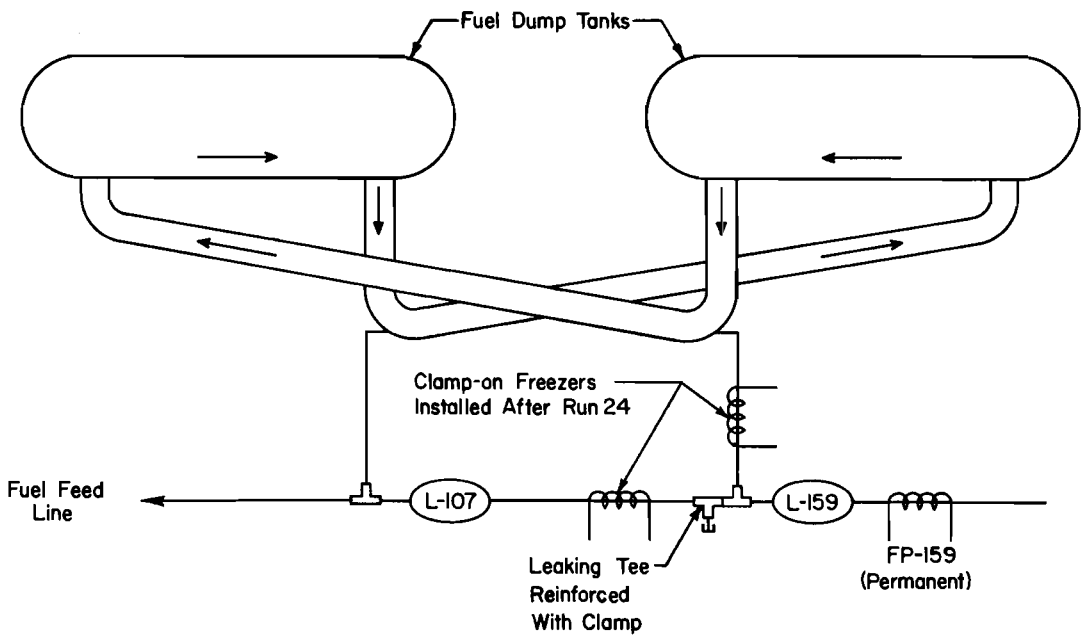

Fig. 5. Schematic Arrangement of Fuel Dump Tanks and Fuel Feed Line.

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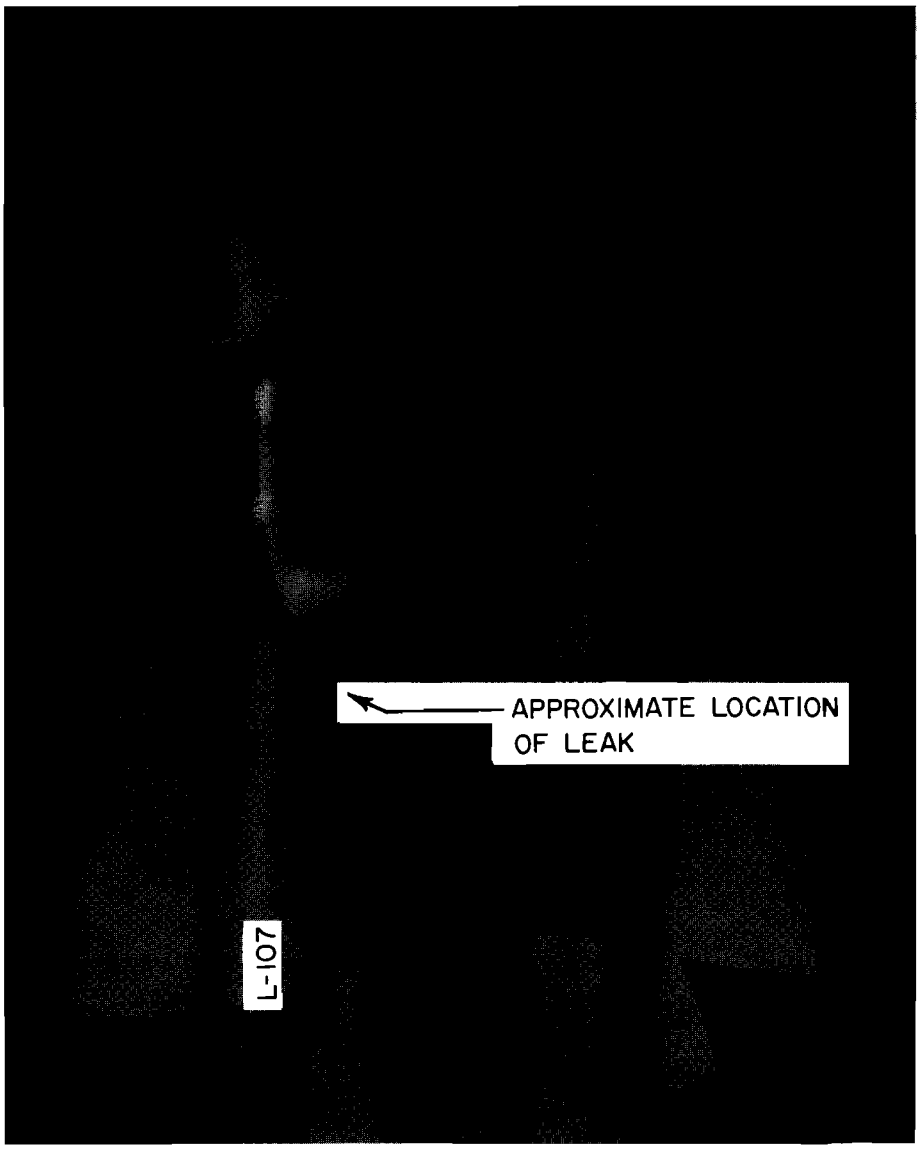

Fig. 6. Leaking Tee in Fuel Feed Line After Removal of Salt Deposit.

the bottom of the cell to remove the floc. The net improvement of visibility in the work area was small.

Since the tee-clamp was the smallest item to be installed, it was put in place first and the two attached bolts tightened. Then the two freeze jackets, each with insulated supply and return coolant lines attached, were installed. The coolant lines and four thermocouple leads were brought out through a specially-fabricated penetration in the top of the cell. Coolant for the freeze jackets was supplied by the chemical plant refrigeration system.

Photographs of the installation were made after the shield water was drained from the cell. These revealed that only one of the two bolts on the tee-clamp was completely made up, but, with the freeze jackets in place, it was impossible to reach the other bolt for further tightening. Subsequent hydrostatic tests showed that the tee was adequately supported. The freeze-jacket installations appeared to be in good order. Figure 7 shows the tee-clamp and part of the adjacent freeze jacket, while Fig. 8 shows the other freeze jacket.

Replacement of Fuel Letdown Heat Exchanger

The second major maintenance operation performed during this shutdown was the replacement of the fuel letdown heat exchanger (LDHX). Reactor operations during run 24 (see page 24) had indicated an interconnection between the feed and letdown sections of the unit and special tests during the early part of the shutdown had confirmed the fact.

The most difficult part of the replacement operation was the removal of the old LDHX. Although the unit has a relatively small total heat transfer surface, it is a very cumbersome piece of equipment. Figure 9 is a sketch of the LDHX showing some of the principal dimensions.

The heat exchanger has eight flanged joints, of which six are process lines with ring-joint flanges and two are cooling-water lines with soft gaskets. It was originally planned to install temporary freezer jackets and to freeze ice plugs in all of the process lines so that the entire removal operation could be performed under water. Although the jackets were installed without much difficulty, it was not possible to establish freeze plugs in all of the lines at the upper end. (One of the freeze jackets was only 1 in. long because of space limitations.) It was then decided to open the upper flanges with the shielding water below the level of the flanges by working through the dry maintenance shield. Since it was necessary to flood over the open flanges for removal of the LDHX, rubber stoppers were inserted in the open ends of the permanent flanges to keep light water out of the process piping. In order to provide access for insertion of the stoppers, the mating flanges on the LDHX were sawed off. Figure 9 indicates the flanges which were cut off. The process lines at the lower end were sealed with ice plugs and these flanges were opened under water.

With all of the flanges opened, the cell water level was raised to provide the required shielding for moving the LDHX. Again, because of the

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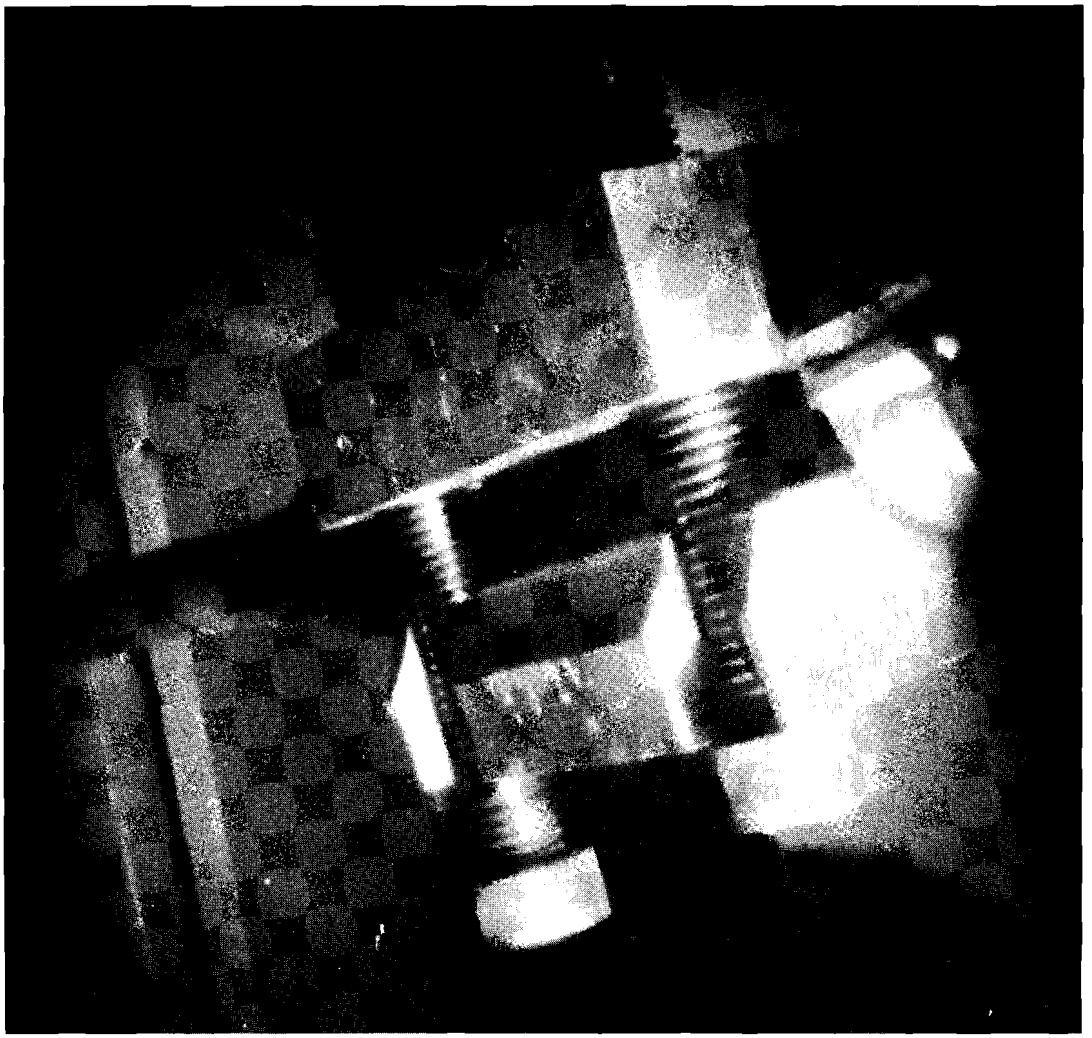

Fig. 7. Photograph of Clamp Around Cracked Tee in Line 107.

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Fig. 8. Freeze Jacket Installed on Downcomer from East Fuel Dump Tank to Fuel Feed Line.

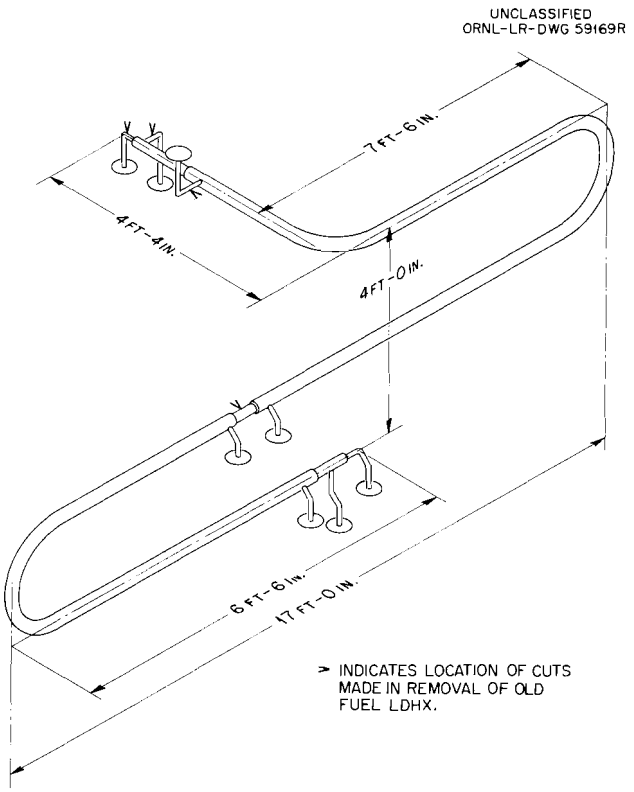

Fig. 9. Schematic of the Fuel Letdown Heat Exchanger.

awkward shape of the unit, it was necessary to cut it into two parts to permit its removal without raising parts of it above the water. After removal, it was stored in the reactor cell because this was more convenient than cutting it into sections small enough so that they could be transported in the available shielded carrier.

The installation of the new LDHX reversed the above procedure except that, since it was uncontaminated, it could be handled above the water, thereby greatly simplifying the operation. In the new installation two thermocouples which had been welded to the old LDHX were not reinstalled. In addition, one thermocouple on the permanent piping was destroyed in installing the temporary freeze jackets.

Other Maintenance Operations

Several, relatively simple, maintenance operations were also performed during the shutdown. The blanket dump valve (PCV-252) and the chemical-plant isolation valves (HCV-141 and -142) were replaced by dry-maintenance techniques, and both heads of the fuel feed pump were replaced under water.

Before the reactor cell was flooded, the east head of the fuel feed pump was temporarily unbolted to permit the activity in the intermediate drive line to be flushed out. While the flanges were open, the refrigeration system failed and was out of service for several hours, allowing the ice plugs in the process lines to thaw. This, along with misoperation of a valve, resulted in the loss, to the cell floor, of about 200 lb of heavy water that had been stored in the fuel condensate tank.[15] The flanges were reclosed and some heavy water was transferred from the blanket condensate tanks to re-establish the ice plugs.

In general, the use of ice plugs and other techniques was very successful in keeping light water out of the process piping during the maintenance period. Samples taken at the end of the period indicated that no more than 14 lb of light water had entered the system.

AUXILIARY SYSTEMS

Reactor Steam System

Changes

Several changes were made in the reactor steam system before run 22.

The containment of activity was improved in the event that a failure in the main heat exchanger allowed fuel solution to enter the reactor

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[15] It had been necessary to store some of the D_2O in the condensate tanks because the blanket storage tanks did not have sufficient capacity for all of the operating inventory. The fuel storage tanks contained the old fuel charge that was taken out of circulation at the end of run 23.

steam. The existing monitors and block valves closed the main steam lines if activity was detected in the steam. Two new lines with valves actuated by the existing activity interlock were installed to vent the steam lines downstream of the block valves to the reactor cell (which is maintained at subatmospheric pressure).[16] Any leakage past the block valves would be conducted back to the cell, minimizing contamination of the air-cooled condenser and other unshielded parts of the system. Another interlock was provided to close the vent valves if the cell pressure exceeded 1 psig.

A pressure-control valve was installed in the condensate discharge line from the air-cooled condenser. This valve allowed the operating pressure of the condenser to be raised sufficiently to provide capacity for subcooling the condensate, to prevent excessive flashing as the condensate discharged into the deaerator. An indicator-controller for the valve was installed on the main control board.

To improve the operation of the feedwater deaerator, an automatic steam-pressure control valve was installed. The steam supply had previously been throttled manually.

The 1/4-in. valves in the heat-exchanger blowdown-sample lines were replaced with 3/8-in. valves. On several occasions the smaller valves had become plugged with solids from the heat-exchanger blowdown.

Operation

The reactor steam system operated smoothly. The only difficulty was the rupture of the deaerator-steam-controller bellows early in run 22 due to freezing. The deaerator steam was controlled manually until the end of run 24, when an improved controller was installed.

Reactor Steam and Cooling Water Systems Chemistry

The reactor steam system was operated in the normal manner 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 and rarely exceeded 0.1 ppm. The rate of addition of hydrazine was very generous at 1.77×10^{-5} parts N_2H_4 per part steam produced. A rate of treatment 30% less than this was found adequate in run 21.[17]

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[16] S. E. Beall et al., HRP Quar. Prog. Rep. Jan. 31, 1960, ORNL-2920, p 16-18.

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

The average concentrations of ammonia and hydrazine in sixteen sets of heat exchanger samples at 1.8 Mw and above showed striking differences between the blanket heat exchanger (BHX) and fuel heat exchanger (FHX). The same chemical treatment per pound of feedwater reached both heat exchangers. However, the BHX was operated at a lower temperature than the FHX in order to maintain the blanket solution at 230°C. There were the usual differences in the radiation dosage in the BHX and FHX resulting from the smaller power fraction, lower power density and the slower rate of circulation of the blanket fuel solution.

The average concentration of BHX hydrazine was 23 ppm in the letdown and 3.4 ppm in the steam. The FHX hydrazine was 1.6 ppm in the letdown and 0.5 ppm in the steam.

The average concentration of BHX ammonia was 90 ppm in the letdown and 75 ppm in the steam. The FHX ammonia was 22 ppm in the letdown and 65 ppm in the steam. The ammonia level was considerably higher than would be found in a once-through steam cycle with the rate of hydrazine treatment used because the steam condensate was recycled. Since the deaerator removed only fractions of the ammonia and hydrazine from the steam condensate, a definite buildup of ammonia was to be expected.

The demineralized cooling water system was maintained at a nominal concentration of 0.1% potassium chromate. It was necessary to replace about 25% of the potassium chromate per week after a steady cooling water leak into the reactor cell developed. The pH varied from 8 to 9.

Instrumentation and Control

During the long shutdown preceding run 22, a number of revisions were made in the reactor control circuits. Some were required because of the plugging of the core tank holes; others were to improve containment or to facilitate operations. No major difficulties with instrumentation and control were encountered during runs 22 through 24. Some of the changes are described in the sections on the steam and off-gas systems and containment. Others are described below.

Pressurizers

After the installation of the core tank patches, which restricted the free access between core and blanket, it became necessary to operate with steam in both pressurizers to accommodate expansion of solution in either region. (In earlier runs, the core pressurizer alone provided adequate relief for both regions.) The pressurizers could not be freely interconnected, however, because the leak which remained around the lower core tank patch was large enough that it was necessary to maintain the pressure in the blanket pressurizer several psi below that in the core pressurizer to reduce transfer through the hole to within the capacity of the blanket purge supply (about 6 lb/min). During the initial tests of the pressurizer level control, a rupture-disc assembly separated the

pressurizers.[18] An orifice, 0.0625 inches in diameter, was drilled in parallel with the rupture discs so that a small flow of steam would help minimize pressure fluctuations between the pressurizers. The control requirements were quite stringent, since if the differential pressure changed by more than about 0.2 psi, both liquid levels could not be kept within the 5-inch ranges of the level-element floats. The solution to this problem was to control the fuel pressurizer level with the fuel letdown valve, and to control the blanket level by regulating the blanket pressurizer heat input relative to the core pressurizer heat input. Backup protection against solution rising into the blanket pressurizer was provided by an interlock which opened the blanket letdown valve (normally closed) on high level. The heater control proved to be quite effective, and the blanket letdown valve opened only a few times during runs 22 through 24.

With steam in both pressurizers, the use of the dump valves as pressure relief valves was no longer necessary, and the loading pressure on the blanket dump valve was increased to reduce leakage past the seat (see page 46). Another measure adopted during operation with one pressurizer was the use of a temperature probe to back up the float-type level indicator on the fuel pressurizer. This was retained, but the routine daily check of its response was discontinued, since undetected failure of the float element was less likely and the consequences of failure were less severe with both pressurizers in service.

Two sets of interlocks which had proved to be unnecessary in operation were eliminated before run 22. The first were pressurizer level interlocks which had stopped the circulating pumps on low level. The other unnecessary interlocks which were removed were those which prevented the pressurizers from being vented during filling of the high-pressure systems (with the start-run switch in "start" position).

Automatic venting of the blanket pressurizer in the event of excessive pressure (600 psi above normal) was reinstated. (While the blanket pressurizer was liquid-filled, only the fuel pressurizer was automatically vented.)

Dump

While the core and blanket were freely connected, only the blanket dump valve was used during a dump (the core dump valve was not allowed to open) to avoid the possible overfilling of the fuel dump tanks. After the core tank was patched, the action of the core dump valve was restored to both the controlled dump and the "last ditch" dump. Interlocks on core-blanket differential pressure were set to close the appropriate dump valve during a controlled dump should excessive pressure differential develop. (This was more likely during operation with the blanket temperature much lower than that in the core.)

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[18] During the reactor startup in November, it was found that the blanket-to-core disc had failed, even though no pressure differences approaching the 100-psig rating of the disc had occurred. Since the rupture discs were a precautionary measure rather than a necessity, they were replaced with a blank, also provided with a 0.0625-inch orifice.

Blanket Temperature

In order to help cool the core wall, the blanket was kept at a lower temperature than the core. This was accomplished by separating the steam systems, which had been connected, and operating the blanket heat exchanger at a lower pressure. The steam withdrawal from the fuel heat exchanger was left on manual control and this was used to adjust the total power. Steam withdrawal from the blanket was controlled automatically to maintain the desired blanket temperature. The signal used for control was proportional to an average temperature, derived from inlet and outlet thermocouples. (A constant blanket average temperature was desirable to minimize solution concentration changes due to expansion and contraction of the blanket solution.)

Low-Pressure System

In addition to the revised condensate transfer system (described on pages 74—77), there were two other changes in instrumentation and control of the low-pressure system. The operation of checking the blanket feed pump rate was facilitated by changes in the circuitry controlling the blanket dilute valve (HCV-435). A switch was provided so that HCV-435 could be operated without actuating the fuel system shutdown-dilution circuit. Clamp-on thermocouples were installed on the fuel feed cooler to provide better response than had been obtained with the electrically insulated couples previously in use. (These thermocouples were used in calculating the fuel feed rate by a heat balance.)

Oxygen and Off-Gas

Oxygen was injected into the core and blanket high-pressure systems at 1.5 and 1.0 standard liters per minute, respectively, throughout the operating portions of runs 22, 23 and 24. The injection system, as such, performed normally except for two minor interruptions, one near the end of run 22 and one at the start of run 23. Both of these were due to instrument malfunctions and were readily corrected. Two mechanical repairs of the oxygen compressor were made during the operating period. However, since the compressor operates only intermittently to recharge supply cylinders, these repairs did not interrupt the oxygen service to the reactor system.

As a result of the re-evaluation of the reactor secondary containment, the operation of the charcoal adsorber beds was modified in run 22 to keep the operating pressure of the beds below atmospheric pressure. Prior to this time, the discharge of the beds had been allowed to flow normally, through a mercury trap and the metering stations, to the stack. Because of the pressure drops in this equipment and in the charcoal beds themselves, the pressure at the inlet to the beds was normally about 5 psi above atmospheric pressure. In run 22 and subsequent runs, the discharge gas from the charcoal beds was routed through a vacuum pump which provided the pressure required to force the gas through the metering stations to the stack. The pressure at the discharge end of the charcoal beds (inlet of the vacuum pump) was automatically controlled, by a

throttling valve, at a value which kept the bed inlet pressure just below atmospheric--14 to 14.5 psia. Since most of the charcoal-bed pressure drop occurs in the first, small-diameter adsorber sections, this arrangement left the major portion of the beds (where most of the fission gas holdup is achieved) at pressures well below atmospheric. Because of the reduced adsorptive capacity of the charcoal at lower pressure, all three beds were normally used in parallel in these runs to insure adequate holdup time for the decay of gaseous fission products.

On December 29, 1960, a release of gaseous activity in the reactor building was traced to the off-gas system. The discharge of the adsorber vacuum pump normally followed two parallel paths to the off-gas metering station. One of these was a very small stream through a hydrogen analyzer and the other was a bypass around the instrument. The bypass was throttled manually to regulate the flow through the hydrogen analyzer. On this occasion the bypass flow was throttled excessively, allowing a pressure buildup in the vacuum pump discharge line. Under these conditions, some of the off-gas, carrying a small amount of activity, leaked out of a loose fitting and was detected by the building monitors. Normally the gas was vented to the stack where it was greatly diluted before being released.

On January 6, 1961, while the reactor was operating at 5 Mw, the temperature in the 6-in. section of charcoal bed "A" rose to 80°C. Although there was no evidence of combustion in the bed, it was isolated to allow the temperature to drop. Within 24 hours after bed "A" was isolated, significant amounts of short-lived gaseous activity began to appear in the discharge from bed "B." Bed "B" was then isolated and bed "A" was put back in service. Bed "B" was also restored to service as soon as the shorter-lived activity had decayed. The general temperature of the charcoal adsorbers was reduced about 10°C (by lowering the cooling water temperature from 26°C to 14°C) to increase the effective adsorption capacity of the beds and improve the safety margin in the off-gas holdup time with all three beds in service.

The deuterium concentration in the reactor off-gas was continuously monitored by a conductivity-type hydrogen analyzer which had been installed late in run 21. The reliability of the hydrogen analyzer had been demonstrated in pre-operational tests and the zero and range of the instrument were checked periodically with standard gas mixtures. During normal operation, the instrument reading was recorded once per shift. With the new low-pressure recombiners in service (see below), the highest deuterium concentration recorded in the off-gas was 3.8% in run 22. The usual concentration in all three runs was 1 to 2%, which was much lower than had been observed before the recombiner replacement.

COMPONENTS

Recombiners

Because of progressively poorer recombination of radiolytic gas in previous runs, both the primary and secondary fuel low-pressure recombiners were replaced during the shutdown which preceded run 22. A number of

changes were incorporated in the replacement units to improve performance and permit a more detailed observation of their operation.[19] In the primary recombiner, the catalyst-support material was changed from alumina pellets to Incoloy ribbon, and the bed was designed for axial rather than radial flow. In the old recombiner, there was evidence that the alumina pellets had been degraded and had settled, leaving an open path through which vapor could bypass the catalyst. The new bed was expected to have greater mechanical durability and the axial flow pattern further reduced the possibility for forming channels. Thermocouples were installed at the inlet and outlet of the recombiner section to permit evaluation of the recombination heat. The secondary recombiner was made larger and was provided with an electrically heated jacket to permit heating to 600°C. Thermocouples were installed in the secondary recombiner bed and in the heating jacket.

Pre-operational Tests

Extensive testing of both the primary and secondary recombiners, under a variety of conditions, was carried out in the light-water testing period from October 6 to 20, 1960. The tests were run with oxygen and normal hydrogen injected, from cylinders, directly into the low-pressure system. The off-gas was routed directly to the off-gas metering station, bypassing the charcoal adsorber beds. This arrangement permitted the detection of unrecombined gas within a few minutes of the time it left the recombiners, permitted complete purging of the system in about two hours, and also eliminated the possibility of setting fire to the charcoal. The presence of low levels of hydrogen was detected with the Thermatron hydrogen analyzer, and, by material balance, the off-gas flow rate indicated the amount of hydrogen present at high concentrations.

Secondary Recombiner.--The secondary recombiner was tested at conditions simulating those to be expected with the reactor in normal operation. Hydrogen was injected at from 2 to 4 standard liters per minute along with oxygen at 2.5 SLPM in excess of the stoichiometric recombination requirement. High D_2O vapor flow rates were maintained in the primary recombiner to reduce the possibility of recombination at that location.

No recombination occurred when the secondary recombiner was preheated to 110°C. However, when the bed was preheated to 200°C, essentially complete recombination took place. At the low hydrogen addition rates, the recombination was accompanied by regularly spaced flashes which apparently originated at the leading edge of the secondary recombiner bed. The flash frequency increased with increasing hydrogen flow up to 3.0 SLPM of hydrogen. The temperature at the upstream end of the recombiner bed rose gradually to about 450°C just before each flash and dropped to 300

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[19] W. R. Gall et al., HRP Prog. Rep. for Period May 1 through October 31, 1959, ORNL-2879, p 35-38.

to 350°C afterwards. The lower minimum temperatures were associated with the lower flash frequencies and, hence, with the lower hydrogen flow rates. The time between flashes decreased from 12-14 min at 2 SLPM of hydrogen to about 6 min at 3 SLPM.

When the hydrogen addition rate was increased to 3.5 SLPM, the flashes stopped and continuous recombination, either catalytic or by burning, took place. The temperature at the upstream end of the recombiner bed leveled at 300°C. These conditions remained unchanged when the hydrogen flow was increased to 4.0 SLPM.

The maximum concentration of hydrogen detected in the off-gas was about 0.5%, which corresponds to a minimum recombination efficiency of 99.5%.

Primary Recombiner.--The first tests of the fuel primary recombiner were carried out at the following simulated reactor operating conditions:

1. dump tank boilup rate of 8 lb/min,
2. heating steam on to iodine adsorber bed, cooling water off,
3. saturated, high-pressure (215 psig) steam on to recombiner heater,
4. fuel dump tank pressure at 17 psia,
5. excess oxygen flow at 2.5 SLPM, and
6. hydrogen flow at 2.0 SLPM.

The secondary recombiner was unheated during these tests to prevent recombination there. Off-gas measurements indicated a complete lack of recombination. The test was repeated with superheated (400°C) steam on to the recombiner heater, also with negative results. Then the vapor boilup rate was progressively lowered to 1.35 lb/min and the hydrogen addition rate was increased to 6 SLPM. Although the last conditions gave a hydrogen concentration in the vapor of 20 times the value expected with the reactor operating, there was no evidence of recombination.

In the next tests, the vapor flow rate through the primary recombiner was reduced to zero and essentially complete ($>$99%) recombination was achieved with 4 SLPM of hydrogen. The hydrogen flow rate was then progressively increased to 38 SLPM with no diluent vapor. This flow rate raised the maximum primary-recombiner-bed temperature to about 500°C where it was held for several hours. The recombination efficiency during this time was greater than 99%. After this treatment, the vapor boilup rate was gradually raised to 0.84 lb/min and then to 2.43 lb/min, still with 38 SLPM of hydrogen flowing. High efficiency recombination continued at the lower vapor rate but, at the higher rate, the efficiency dropped to 80 to 90%. Since this low efficiency prevailed at conditions at least 50 times more favorable for recombination than reactor operating conditions, it was concluded that the primary recombiner would serve no useful function during normal reactor operation. However, since it had been demonstrated that adequate recombination was obtained in the secondary recombiner, normal reactor operation was resumed without further attention to the primary recombiner.

Performance During Operation

Superheated steam was maintained on the primary recombiner and the secondary recombiner was held at at least 250°C throughout runs 22, 23 and 24. Under these conditions, the net recombination efficiency was very good. The maximum deuterium concentration detected in the off-gas was 3.8% and the usual concentration was 1 to 2%.

Diaphragm Pumps

Blanket Feed Pump

The west head of the blanket feed pump was taken out of service on October 12 (during the light water testing period) because the intermediate system would not retain phasing water. While the reactor was shut down on October 18 to replace the multiclone filter, the west head was replaced, without flooding the cell, by working through the dry-maintenance shield. Examination of the head (No. 14) showed that a weld crack had developed where a plug had been welded into a spare connection on the intermediate side of the pump head. Head No. 14 was removed after 4035 hours of service and a new head (No. 24) installed.

Fuel Feed Pump

Run 22 was started with both heads of the fuel feed pump in service. The west head would not remain properly phased, apparently because of leakage of gas into the intermediate system. The west head was taken out of service on November 9 (while pumping continued with the east head) to replace the O-ring seals in the pulsator housing. The new installation still leaked under a vacuum test, and the west head was left in standby.

On November 26, it was necessary to rephase the east head because of gas accumulation in the intermediate system, and on November 29 both heads were rephased and remained in service to the end of the run.

Run 22 operation was ended on December 4 when fresh fission-product activity was detected in the cell. By remote smearing, the source of this activity was traced to the east head of the fuel feed pump. The surface contamination was rinsed from the pump head, using a 4% ammonium oxalate, 3% hydrogen peroxide solution, and the head was removed through the remote-maintenance shield and transferred to the storage pool. Because of the short cooling period, the radiation from the pump was unusually high, 100 r/hr at 4 inches through the shielded carrier (1-1/2 inches lead and 1/2 inch steel). In the storage pool, the head was pressurized to 1600 psig with gas, and the leak located at a weld on the high-pressure side of the discharge check valve.

The head (No. 26) had been in service for 3399 hours in the reactor and for 19,256 hours in previous nonradioactive operation, and was replaced with a new head (No. 25).

Run 23 was started with both fuel feed pump heads in service. The bellows of the phasing valve in the intermediate system was replaced on December 15.

Because of gas accumulation in the intermediate system, it was necessary to rephase the west head on December 22 and 23, and finally on December 24 the head was phased out of service, while pumping continued with the east head only. A vacuum test indicated that air was leaking into the west pulsator housing. As a corrective measure, the O-rings were backed up with sealing compound, which appeared to stop nearly all the leakage, and the head was left in standby.

On December 30, the fuel pumping rate became erratic, and it was discovered that the aluminum air-signal line to the pump stroke controller had cracked, apparently due to vibration fatigue. The broken section was replaced with a new copper line.

On January 10, fresh fission-product activity appeared in the east intermediate system, indicating that the diaphragm had ruptured. The west head was placed back in service in spite of the small inleakage of air suspected at the pulsator housing. The east head was isolated with freeze plugs, and the intermediate piping, which read about 300 mr/hr, was shielded with one inch of lead.

During run 24, a new west pulsator housing was installed in an attempt to stop the leakage of gas into the intermediate system. The reactor was held subcritical and the intermediate line was frozen during the change. It was found that some gas continued to accumulate in the intermediate system even after the new housing was installed.

At the end of run 24, the east head (No. 25) was replaced after 622 hours of service in the reactor and 3308 hours of previous nonradioactive operation. The replacement head (No. 28) was fabricated by assembling test head No. 16 (which had 1728 hours of previous service out-of-pile) with new discharge check valves, since material was not available for a complete new head. The suction check valves were taken from the former reactor fuel feed pump No. 16.

The west fuel-feed-pump head remained in service to the end of run 24. However, during the ensuing shutdown, a low level of fresh fission-product activity was detected in the west intermediate system, indicating a very small leak in the diaphragm. The head (No. 27) was replaced after 1844 hours of service in the reactor and 19,000 hours of previous nonradioactive operation. The replacement head (No. 30) was fabricated from the former blanket feed pump head No. 14, by repairing the weld crack and installing new flanged connections to fit the fuel feed pump.

Fuel-Circulating-Pump Purge Pump

During run 22, the fuel-circulating-pump purge pump stopped pumping, apparently because of check-valve leakage. A fast-suction-stroke cam was installed in the drive unit on November 17 and pumping was restored. After a few days the pumping rate gradually decreased. The pumping rate was brought back to normal by installing a variable-speed drive unit and increasing the pump speed from 31 to 75 strokes per minute.

The pump gave no further trouble until December 30, when it again stopped pumping. Pumping was restored by operating the pump at a very high speed (about 200 strokes per minute) for a few minutes, and then returning the speed to normal (30 strokes per minute). Possibly a bit of solid material was interfering with the operation of the check valves and was flushed away by the rapid operation of the pump.

Samplers

The samplers functioned without major difficulties in runs 22, 23 and 24.

A minor repair was made to the fuel sampler at the start of run 24; the loading tube cam fastening had loosened so that there was insufficient clearance for the sample flask holder assembly to move freely. The cam was retightened with a remote tool.

The volume distribution for the reactor samples taken in each run is shown in Table 1. In run 22, 91% of the samples were of the required 5-ml minimum volume; in run 23, 95%; and in run 24, 98% (compared with 85% in run 21). The improvement in the proportion of successful samples resulted mainly from two new techniques (adopted in run 21); i.e., (1) the adjustment of the isolation chamber pressure to about 2 psig before draining, and (2) the redraining of the isolation chamber using a new sample flask when the volume of the original drain was insufficient.

Valves

Two reactor piping changes before run 22 were facilitated by the accessibility of the reactor valves for remote replacement. The new secondary recombiner (see page 41) was installed by removing HCV-344 (valve K-23) and replacing it with a unit consisting of the new recombiner and a new valve (K-06). The new transfer line (see page 74) was installed by removing HCV-434 (valve K-48) and replacing it with a new valve (K-14) connected to the transfer line. The other end of the line was installed by opening the HCV-938 position (which had been blanked) and connecting the line with valve K-40.

Except for the blanket dump valve and the chemical-plant supply and return valves, the reactor valves functioned without difficulty in runs 22, 23 and 24.

The blanket dump valve was known to be leaking slightly through the seat in run 21, and the leakage continued to increase slowly during the following runs. On January 7, to improve the seating of the valve, the loading pressure was increased from about 34 psig to 50 psig.[20] However,

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[20]The increase in loading pressure nullified the pressure relief function of the valve; however, pressure relief was no longer considered necessary since the blanket pressurizer expansion volume was available.

Table 1. Sample Volumes in Runs 22, 23 and 24

	Volume Range (ml)	Number of Samples				Grand Total
		High-Pressure System		Low-Pressure System		
		Fuel	Blanket	Fuel	Blanket	
Run 22						
	0 - 2	0	0	0	1	
	2 - 5	2	4	0	1	
	5 - 8	1	9	3	0	
	8 - 10	15	16	0	2	
	10 +	21	8	5	0	
	Total	39	37	8	4	88
Run 23						
	0 - 2	0	0	0	0	
	2 - 5	0	3	0	0	
	5 - 8	5	6	1	0	
	8 - 10	26	20	0	1	
	10 +	1	0	1	0	
	Total	32	29	2	1	64
Run 24						
	0 - 2	0	0	0	0	
	2 - 5	0	1	0	0	
	5 - 8	1	3	0	0	
	8 - 10	25	18	0	0	
	10 +	3	1	1	0	
	Total	29	23	1	0	53

the loading-air mercury seal, set to vent the loading air at a pressure of about 35 psig, was inadvertently left open, so that the full increase in loading pressure was not effective until the valve to the mercury seal was closed on January 11. Near the end of run 24, the leakage rate to the blanket dump tank, measured by isolating the tank and observing the increase in weight, was 2.2 lb/min. At the end of run 24 the valve (K-71) was removed after 6,882 hours of service, stored in the reactor cell, and replaced with valve No. 32, which had been used previously in non-radioactive service.

The chemical-plant supply and return valves, HCV-141 and -142, developed leakage through the seats, so that it became increasingly difficult to isolate the chemical plant for periodic draining of the solids removed by the hydroclone.[21] An air loading of 20 psig was applied to the vent side of the valve operators, to provide additional seating force to supplement the normal spring loading. On January 6, with the valves loaded in this manner, the combined leakage was estimated, by observing the rate of pressure rise in the chemical plant high-pressure system, to be 50 ml/min at 1400 psig.

On March 7, HCV-142 (valve L-66) was replaced after 8,163 hours of service with a new valve (L-68), and on March 8, HCV-141 (valve L-65) was replaced after 8,163 hours of service with a new valve (L-67). The valves were replaced using the dry-maintenance technique, but with the reactor flooded to a level below the valves to provide background shielding. The valves removed were stored in the reactor cell.

MISCELLANEOUS

Letdown Heat Exchanger Leakage

The fuel letdown heat exchanger is a concentric-tube exchanger (Fig. 9) in which the fuel letdown stream is cooled by giving up heat to the fuel feed stream, the fuel pressurizer purge stream and a stream of cooling water. The letdown stream is cooled from core temperature, say 260°C, to 80°C at the letdown valve. The feed stream is heated from about 55°C, at the feed pump, to within about 30°C of the core temperature. The feed stream flows through the central pipe, a 0.875-inch OD, 0.148-inch wall tube, at about 1 ft/sec. The letdown stream, in the annulus between the central tube and a concentric 1-inch schedule-80 pipe, moves at about 5 ft/sec. Construction of the exchanger is entirely of type 347 stainless steel.

The letdown heat exchanger was known to be particularly susceptible to corrosion because there are portions in which the normal operating temperatures are in the range most conducive to corrosion of stainless

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[21] O. O. Yarbro, Summary of HRT Chemical Plant Runs 22 Through 25, ORNL-TM-101, to be issued.

steel by the fuel solution: from 200 to 225°C (see page 88). Therefore, when the apparent solute inventory began to decrease during the early part of run 24, leakage from the feed to the letdown stream was quickly suspected.

The discussion of how letdown heat exchanger leakage affects the apparent solute inventory makes use of the following symbols:

f = fuel feed rate, measured at the feed pump,

L = fuel letdown rate, at the letdown valve,

B = bypass rate from feed stream to letdown stream,

C_C = concentration in the core, and

C_{DT} = concentration in the dump tank.

These variables are related by a steady-state material balance which equates the rates at which solute enters and leaves the fuel high-pressure loop in the feed and letdown streams.

$$(f - B)\, C_{DT} = (L - B)\, C_C \quad .$$

Normally, the foregoing relation, with B equal to zero, is used to calculate C_{DT} from C_C, f and L. (L is the sum of the measured pump-up rates less the leakage to the blanket dump tanks.) C_{DT} is then used to compute the inventory. Since L is about twice f, calculations ignoring B will underestimate C_{DT}, and solute inventory, if B is actually not zero.

If it is assumed that the solute inventory in the reactor is a known constant, C_{DT} can be obtained from the dump-tank weight and the difference between the total inventory and that in the high-pressure systems. The above equation can then be used to compute B. Values for the bypass rate, computed in this way, are shown in Fig. 10. High-pressure inventories of copper and uranium were computed using sample results; in addition, the high-pressure inventories were computed using the critical concentrations inferred from the temperature and power distribution.

In a special experiment on January 29, the fuel-dump-tank weight was lowered and the feed rate was adjusted as necessary to hold the core temperature constant (constant C_C). This ensured that the dump-tank inventory, i.e., the product of the dump-tank weight, M, and concentration, C_{DT}, remained constant. Thus for two different dump-tank weights:

$$M_1 \left(\frac{L_1 - B}{f_1 - B} \right) C_C = M_2 \left(\frac{L_2 - B}{f_2 - B} \right) C_C \quad ,$$

which was solved for B.

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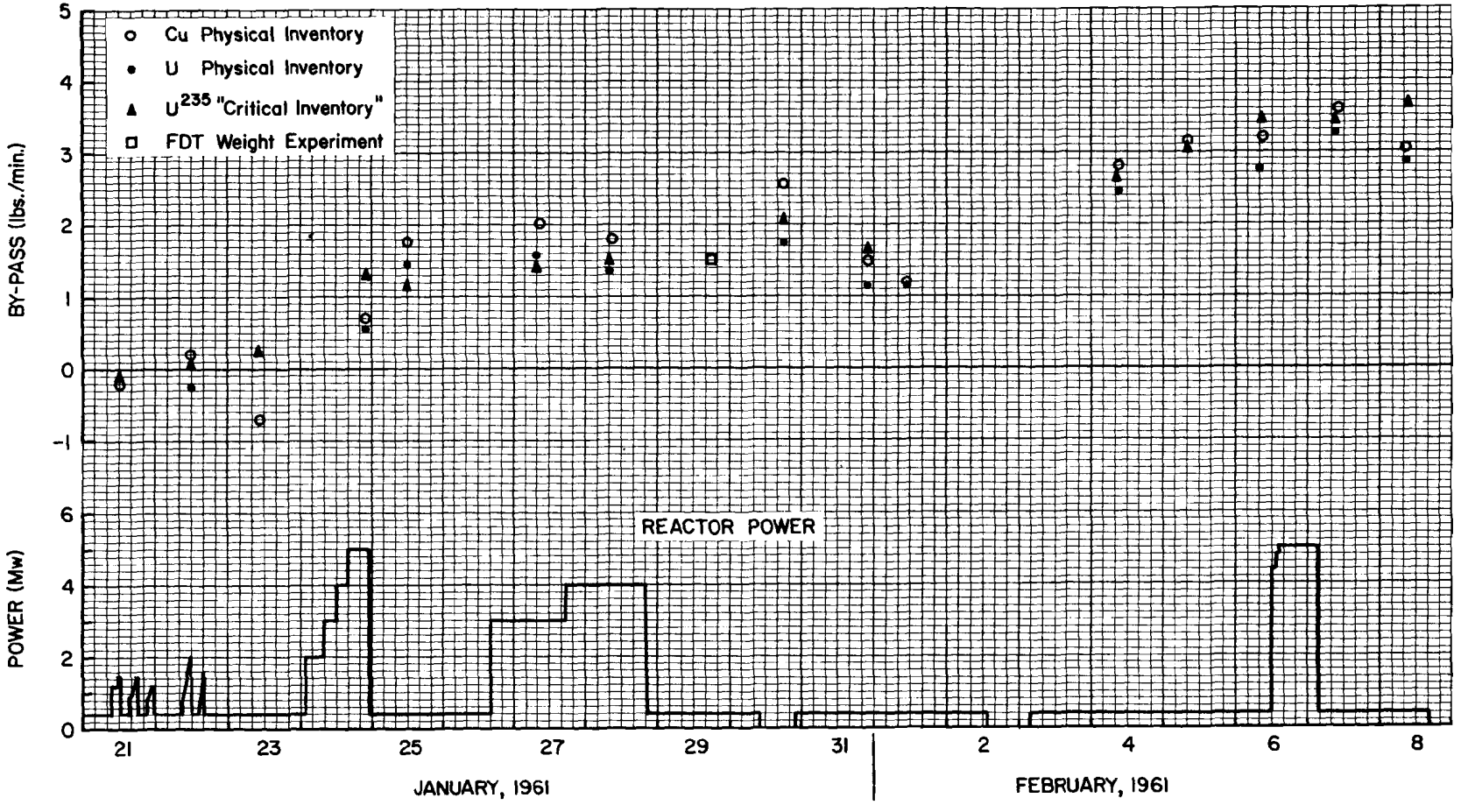

Fig. 10. Fuel Letdown Heat Exchanger Bypass Rate During Run 24.

Another way of calculating B, in principle at least, is by a heat balance around the letdown heat exchanger. Actually, thermocouple error and unmeasured heat losses prevent an evaluation from a single heat balance. These effects were largely eliminated by comparing heat balances taken before and after bypass leakage began. By this method, the calculated leakage at thirteen times on February 5 and 6 ranged from 1.4 to 2.6 lb/min, with an average of 2.1 lb/min. On February 6 and 7, twelve determinations of bypass rate gave values from -0.4 to 5.3 lb/min, with an average of 2.4 lb/min.

It appears from Fig. 10 that the leak grew progressively larger throughout most of run 24.

At the end of run 24, during the rinsing operation, the bypass in the letdown heat exchanger was confirmed by freezing the feed line at the upper end of the exchanger and pumping with the feed pump. Because of the extremely high activity of the exchanger, no examination was made after its removal to determine the nature and location of the leak.

Calculation of Nuclear Average Temperature

In a reactor without control rods, such as the HRT, the critical temperature is a particularly useful tool for following the reactor behavior. Any change in reactivity is reflected as a change in critical temperature and, as such, can be easily observed. However, the critical temperature cannot be measured directly because of the highly nonisothermal character of the system. Therefore, it is necessary to relate the critical temperature, or an approximation of it, to those system temperatures which can be measured: namely, the inlet and outlet temperatures of the core and blanket regions.

A weighted average of directly observable temperatures in the HRT, called the nuclear average temperature (NAT), has been used extensively to study the behavior of the system. The NAT may be defined as the temperature that would be approached by the critical reactor under completely isothermal conditions if there were no changes in concentration. Thus, under isothermal conditions, the NAT and the critical temperature are identical. If the NAT is correctly computed, it will vary with reactivity in the same way as the critical temperature.

The NAT may be defined in terms of the directly observable temperatures as follows:

$$NAT = f\,\bar{T}_c + (1 - f)\,\bar{T}_b \tag{1}$$

$$\bar{T}_c = \theta_c\,T_{co} + (1 - \theta_c)\,T_{ci} \tag{2}$$

$$\bar{T}_b = \theta_b\,T_{bo} + (1 - \theta_b)\,T_{bi} \tag{3}$$

where:

NAT = nuclear average temperature,

$\bar{T}$ = regional average temperature,

T = observed temperature,

f = core-weighting factor,

θ = regional outlet weighting factor,

and subscripts

c = core region,

b = blanket region,

o = outlet, and

i = inlet.

The core-weighting factor, f, is the ratio of the temperature coefficient of reactivity of the core region to that of the reactor. The regional weighting factors, θ, depend upon the flow pattern in the region. The value of θ is 0.5 for straight-through slug flow, and it approaches 1.0 as mixing in the region increases. The value of θ may exceed 1.0 if there is recirculation of fluid from near the outlet of a region back into the active region.

In earlier runs with upward flow in the core, the values of the factors used in computing the NAT were developed from operating experience. The values used were $f = 0.5$, $\theta_c = 0.7$ and $\theta_b = 0.7$. However, experiments on the core hydraulic mockup, prior to run 22, indicated that the value of θ_c would be greater than 1.0 with downward flow through the core because of considerable recirculation of the core fluid in the new flow pattern. Therefore, some special experiments were performed in run 22 to re-evaluate the factors.

The core-weighting factor, f, was evaluated by changing the blanket temperature and observing the change induced in the core temperature. The changes were made with the reactor at heat-loss power, and the purge rates were adjusted to minimize any concentration changes and the associated reactivity effects. The value obtained for f was 0.6, which was in agreement with theoretical calculations for the core-to-blanket concentration ratio which existed. This value was used throughout the three runs.

A value for θ_c was obtained by varying the core-solution circulation rate with the reactor operating at 1.2 Mw. The neutron level, blanket temperature, and core and blanket concentrations were held as nearly

constant as possible. With no changes in concentration, there should have been no reactivity change and, hence, no change in NAT. Then, with the blanket temperature constant, the core nuclear average temperature must have been unchanged. Under these conditions θ_c could be evaluated from the changes induced in T_{ci} and T_{co} by the varying flow rate. From Equation (2), for $\Delta \overline{T}_c = 0$:

$$\theta_c = \frac{1}{1 - \Delta T_{co}/\Delta T_{ci}} \tag{4}$$

The results of the experiments gave values of θ_c which varied from 1.1 to 1.4, compared with a calculated value of 1.3 based on information obtained from the hydraulic mockup.[22] All in all, a value of 1.3 appeared to be the best estimate of θ_c and was adopted for subsequent NAT calculations.

Since there had been no changes in the blanket flow, the value of θ_b was not redetermined. The experiments did show, however, that the values $\theta_c = 1.3$ and $\theta_b = 0.7$ were consistent with the system behavior, and this was further supported in subsequent operation of the reactor.

In runs 22, 23 and 24, the above factors provided a useful temperature index for following the system behavior. That is, a change in NAT, as computed using these factors, indicated a change in system conditions and, conversely, a stable NAT indicated steady conditions. It should be pointed out, however, that there are very few reactor operations that can be carried out without affecting the NAT. For example, changing the power level does not, as such, change the system reactivity, but the change in the system temperature distribution that accompanies the power change does affect the concentration and, hence, the NAT. In addition, the changes in fission-product poison level that follow a power change affect the NAT. Thus a change in NAT does not necessarily imply instability. It is necessary to compare the observed change in NAT with that expected from the operation before a judgment of the reactor stability can be made.

In addition to hourly, manual calculations which are the basis for the NAT plots in Figs. 1, 2 and 3, the NAT was continuously displayed and recorded on the control panel. For this purpose, the outputs of four thermocouples (one each at the core and blanket inlet and outlet) were fed to a small computer which applied the appropriate weighting factors and transmitted the result to the recording instrument. The manual calculations were based on temperature readings obtained from a multipoint

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[22] C. G. Lawson, personal communication.

precision indicator. The readings (in each case, all that were available) of three thermocouples were averaged for the core inlet and for the core outlet temperatures, and single thermocouples were read for the blanket inlet and outlet temperatures.

Both methods of obtaining the NAT were subject to some error. In the automatic record, the limitation was the precision of the computer-recorder chain. In the manual procedure, the variations were due partly to the fact that the individual temperatures were constantly changing somewhat--a finite amount of time was required to obtain eight thermocouple readings--and partly to the temperature variations in the reference junction of the precision indicator.[23] In both cases the NAT variations up to $\pm$ 0.3°C were observed, even under otherwise steady conditions.

Fuel Distribution Between Core and Blanket

Since the new core patches were not leak-tight, it was necessary to continue purging the blanket with a small stream of D_2O--as had been the practice in previous runs[24]--in order to limit the uranium concentration and thus the power generation in the blanket. However, with the new patches, the relatively high purge rates used in the previous runs would have resulted in a uranium concentration low enough ($<$ 1 g U/kg) to cause hydrolytic precipitation of some uranium in high-temperature areas. Because of this possibility, it was decided that the blanket purge rate should be selected and controlled to prevent the blanket concentration from going below the 1 g/kg level.

Evaluation of Core-Blanket Mixing

Since the blanket fuel concentration was to be controlled by the operators, a knowledge of the degree of mixing between the core and blanket was essential to competent operation of the reactor. The mixing may be conveniently determined from a steady-state solute material balance around the blanket high-pressure system:

$$RC_c + f_b C_{bdt} = C_b \left[L_b + R + (P_b - L_b) \right] \qquad (1)$$

where

C = concentration,

f = feed-pump pumping rate,

L = letdown rate, either deliberate or by leakage,

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[23] The reference junction was controlled electrically to $\pm$ 0.25°C.

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

P = total pumping rate into the high-pressure loop,

R = backmixing rate from core to blanket,

and subscripts:

b = blanket

c = core

bdt = blanket dump tank.

The pumping rate of the blanket feed pump, f_b, is treated separately from the other blanket diaphragm pumps which contribute to P_b, because it may pump fuel-bearing solution while the purge pumps pump only condensate. The quantity, L_b, represents leakage to the blanket dump tanks since there was no deliberate letdown from the blanket high-pressure system. The net flow rate from the blanket to the core high-pressure system is given by $(P_b - L_b)$. The quantity, R, is not intended to describe the mechanism by which mixing occurs; it is merely the average flow rate of core solution into the blanket which would be required to produce the observed effects.

All of the quantities, except R, in equation (1) can be evaluated directly, but the equation can be simplified by combining it with a second material balance. At steady state, a solute material balance around the blanket dump tanks gives:

$$C_{bdt} f_b = C_b L_b \quad . \tag{2}$$

Equations (1) and (2) may be combined and the result solved for R:

$$R = \frac{P_b - L_b}{\left(C_b/C_c\right)^{-1} - 1} \tag{3}$$

Thus, the backmixing rate can be evaluated from the blanket-to-core concentration ratio, the total blanket purge rate and the leak rate to the blanket dump tanks.

The value obtained for R from equation (3) varies with the general mixing conditions. Both it and the instantaneous backmixing depend upon the size and configuration of the hole through which mixing occurs, the net flow rate through the hole and the pressure drop across it. However, the instantaneous rate may vary widely, on a short-time cycle, without affecting the average value of R.

Results of Operation

In general, the concentration ratios used to evaluate R are obtained directly from analytical results. In runs 22 and 23, the averages of the ratios of U, Cu and Ni were used, and in run 24 where the Ni concentration ratio was uncertain because of the low Ni concentration in the fuel, only the U and Cu results were averaged. Figure 11 shows the relation between the core-to-total power ratio and the blanket-to-core concentration ratio. The concentration ratios for the points were obtained from sample results in runs 22, 23 and 24, and the power ratios were determined from the flow rates and temperature rises of the primary circulating streams. The solid curve is the calculated relation using the GNU code. The agreement between the calculated and observed relation is reasonably good; both analytical error and errors in the power determination contribute to the scatter in the empirical results. With Fig. 11 the concentration ratio may be determined at times other than sample times. This approach is not sufficiently precise for absolute determination of the concentration ratio, but it is particularly useful for showing short-term changes in the concentration ratio.

Once the concentration ratio has been determined, it is necessary only to evaluate the net blanket purge, $(P_b - L)$, to obtain the backmixing rate. The observed effect of net blanket purge rate on the concentration ratios in runs 22, 23 and 24 is shown graphically in Fig. 12. The results cover a relatively narrow range of purge rates because the blanket concentration was controlled at slightly above 1 g U/kg D_2O. The results are also somewhat cloudy because of uncertainty in the leak rate to the blanket dump tanks. This leakage was low at the start of run 22 and apparently increased with time. A blanket dump tank sample taken November 19, 1960 showed the leakage at that time to be less than 0.5 lb/min. Therefore, the run 22 data up to that time were plotted with no allowance for blanket letdown. No run 22 data were used after November 19. The leakage to the blanket dump tanks was measured late in run 23 and several times in run 24 with an average result of 1.9 lb/min. This correction was applied to all of the run 24 data and to the data for the latter part of run 23. The concentration-ratio data for the early part of run 23 are not plotted on Fig. 12 because of uncertainty in the leak rate. The effect of applying a correction for leakage to the blanket dump tanks to a given data point is to move that point to the left in Fig. 12. Therefore, the run 22 points could be moved only to the left and the run 23 and 24 points, which may have been overcorrected (but probably not undercorrected), could be moved only to the right. Thus, the change, from early run 22 to runs 23 and 24, in the effect of purge rate on concentration ratio is, if anything, underestimated.

A short straight line[25] was drawn through the central mass of data for run 22, and a parallel line was drawn through the run 23 and 24 data.

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[25] The actual relation is probably a curve since the concentration ratio must go to 1.0 when the net blanket purge rate goes to zero, but a straight line is a reasonable approximation for the region in question.

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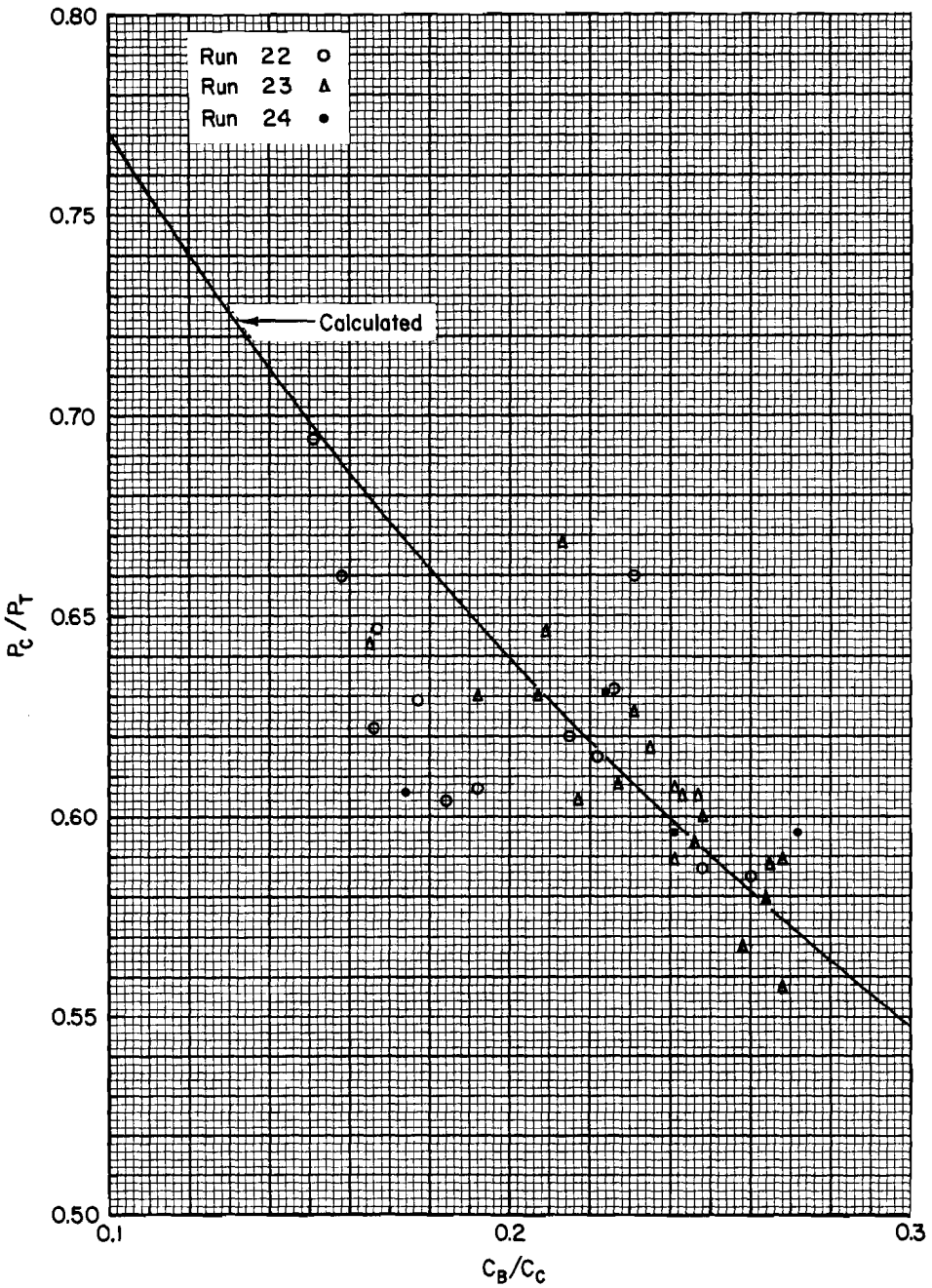

Fig. 11. Effect of Concentration Ratio on Power Distribution.

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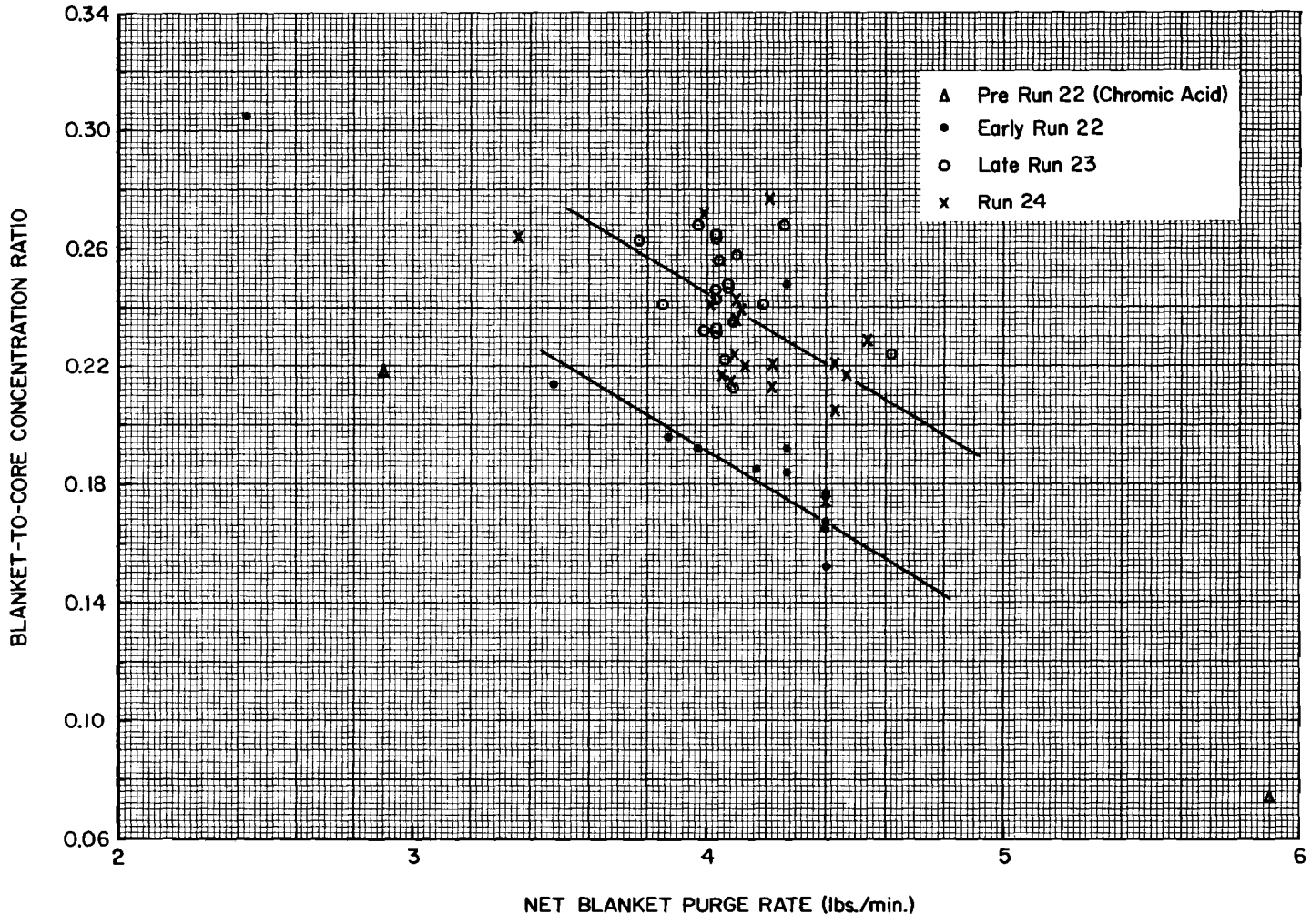

Fig. 12. Effect of Net Blanket Purge Rate on Concentration Ratio.

The concentration ratios at a net blanket purge of 4.0 lb/min were then used to calculate the effective, mean backmixing rates. The mixing rates calculated at this condition for run 22 and for runs 23 and 24 were 0.9 and 1.3 lb/min, respectively. Thus a change in mixing from the start of run 22 is apparent, but there is no significant difference between runs 23 and 24. Since leakage to the blanket dump tanks has already been accounted for and changes in the main loop pressure drops, which determine the ΔP across the patches are unlikely, it must be concluded that the configuration of one or both of the patches changed.

The time at which the condition of the patches changed cannot be fixed with certainty, but it appears that most of the change occurred during run 22. Between November 15 and 28, it was observed that the blanket-to-core concentration ratio increased (see Fig. 1) while the blanket pumping rate was relatively constant.[26] The leakage to the blanket dump tanks was not measured during this period, but an increase in leakage of 1.7 lb/min would have been required to cause the observed change in concentration ratio if there were no change in backmixing. An increase of this magnitude in the leakage would be reflected as a decrease in the calculated inventories. No such effect was observed at the time,[27] so it appears likely that the change in concentration ratio was due to increased backmixing.

Critical Concentrations

Predictions

Before the beginning of run 22, calculations were made to predict the critical concentrations (and other nuclear properties of the HRT) to be expected when the reactor was operated with the blanket temperature much lower than the core.[28] GNU, a one-dimensional, multigroup diffusion code for the IBM-7090, was used to compute critical concentrations. The calculations used the most up-to-date cross-section data, and took into account the stable neutron poisons in the HRT fuel solution and the enrichment of uranium (84%) and heavy water (97.5%) which existed in the HRT at the beginning of run 22.

The GNU results were used to construct working curves which could be conveniently used in planning and analyzing the operation of the HRT. Figure 13 shows the predicted core critical concentration as a function of core temperature for several blanket/core concentration ratios. (The ratios are for concentrations on a molal, or mass, basis rather than on a molar, or volumetric, basis; the two are different if the blanket temperature is not the same as that in the core.) During runs 22 through 24 the blanket

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[26] The increase in blanket concentration was reversed and the desired concentration restored by increasing the blanket purge rate.

[27] The behavior of the calculated physical inventories indicates that most of the increase in blanket leakage probably occurred after about 500 hours of operation in run 23 (see page 88).

[28] D. R. Vondy and M. L. Tobias, Nuclear Properties of the HRT-- A Collection of Steady-State Calculations, ORNL CF-61-8-42 (Aug. 4, 1961).

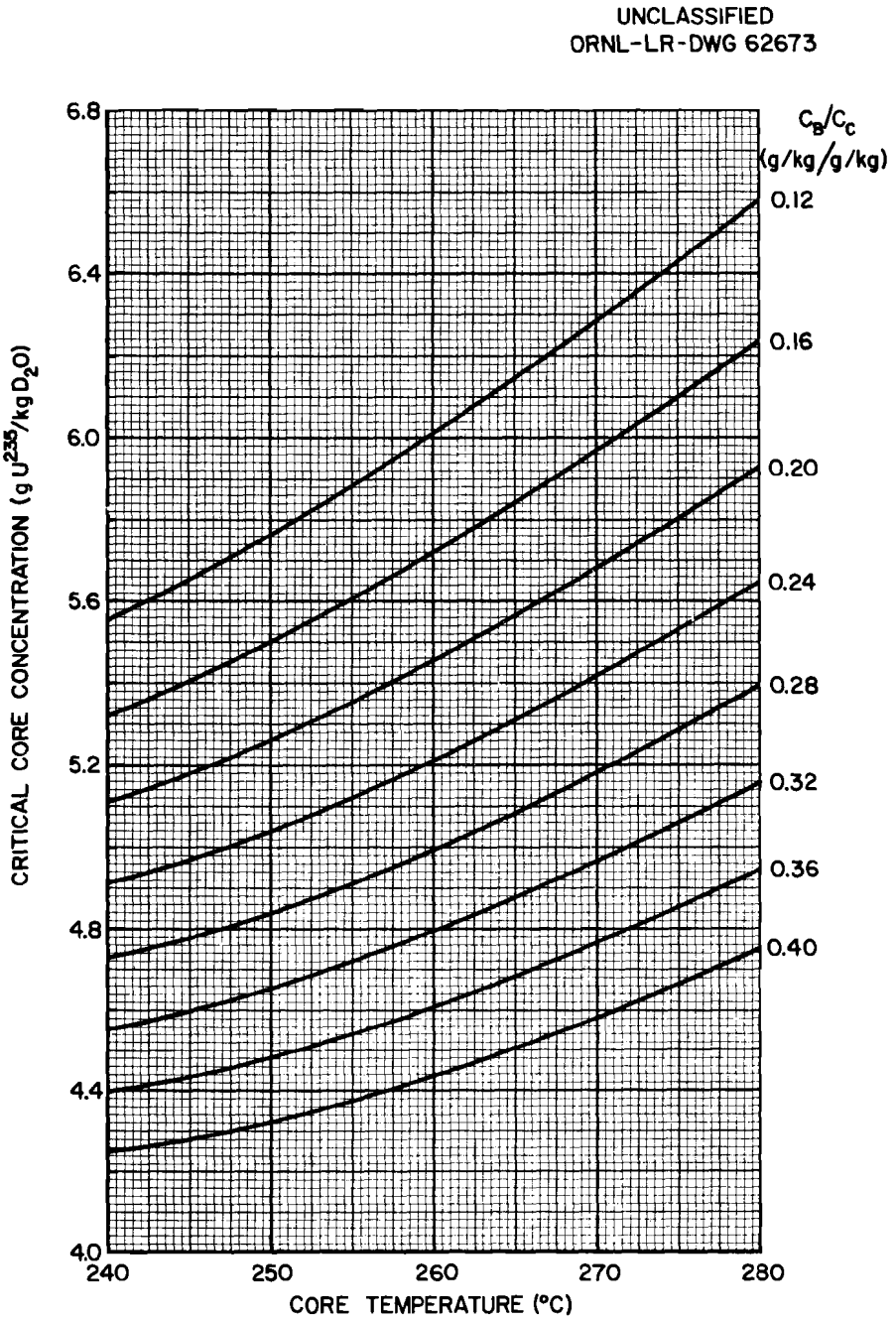

Fig. 13. Critical Concentrations Predicted by GNU Calculations for Run 22 Conditions.

temperature was held very near 230°C; the predicted effect of deviations from this blanket temperature, shown in Fig. 14, was used to adjust the concentrations from Fig. 13.

Comparisons

During runs 22 through 24, as in all HRT nuclear operation, the predicted critical concentration at the time of each pair of high-pressure system samples was compared with the observed concentration. These comparisons served several purposes: Occasionally, abnormal error in sampling or analysis was revealed by an unusually large difference between the sample results and the predicted concentration. The presence of noncirculating uranium in the active region of the reactor was detected, in earlier runs, by a correlation between power level and the difference between the solution concentrations and the concentrations required for criticality. The data from runs 22 through 24 were examined for such a correlation. Finally, the accuracy of the criticality calculations was checked by comparison of the predicted critical concentrations with all of the observed concentrations.

Results of Runs 22 and 23

Power-Dependent Uranium Effects.--Figure 15 shows runs 22 and 23 data in the type of plot which would reveal the effect of power-dependent, noncirculating uranium, if such existed. Shown in this figure is the ratio of the predicted core critical concentration to the core concentration indicated by sampling. The critical concentration was obtained from the GNU results, using the uranium, sulfate, copper and nickel analyses to compute the blanket-to-core concentration ratio. The core and blanket average temperatures were computed from the inlet and outlet temperatures, using weighting factors of 1.3 on the core outlet and 0.7 on the blanket outlet (see pages 51-54). The "observed" U-235 concentrations were computed from the reported concentrations of total uranium in the samples, corrected for analytical bias,[29] and the enrichment of the uranium in solution.[30] Figure 15 shows no correlation of the ratio with power. It thus appears that there was very little, if any, deposition of uranium on the core tank as the power was raised. (Fifty grams of uranium depositing on the core tank wall would increase the ratio by 0.2.)

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[29] During run 22, the uranium analyses of eighteen control samples with concentrations near those in the core averaged 1.0065 of the actual concentrations, with a standard deviation of 0.0124. In run 23, the ratio of analysis to actual concentration in thirteen control samples was 1.0116 ± 0.0101.

[30] The enrichment used in the conversion was 83.46 weight percent U-235, based on run 21 sample results and calculated changes due to burnup. Three samples taken during runs 22 and 23 assayed 82.67, 82.63 and 82.70 weight percent U-235. If the average of these were used for the conversion from total uranium to U-235, all the points on Figs. 14 and 15 would be moved up by 0.009.

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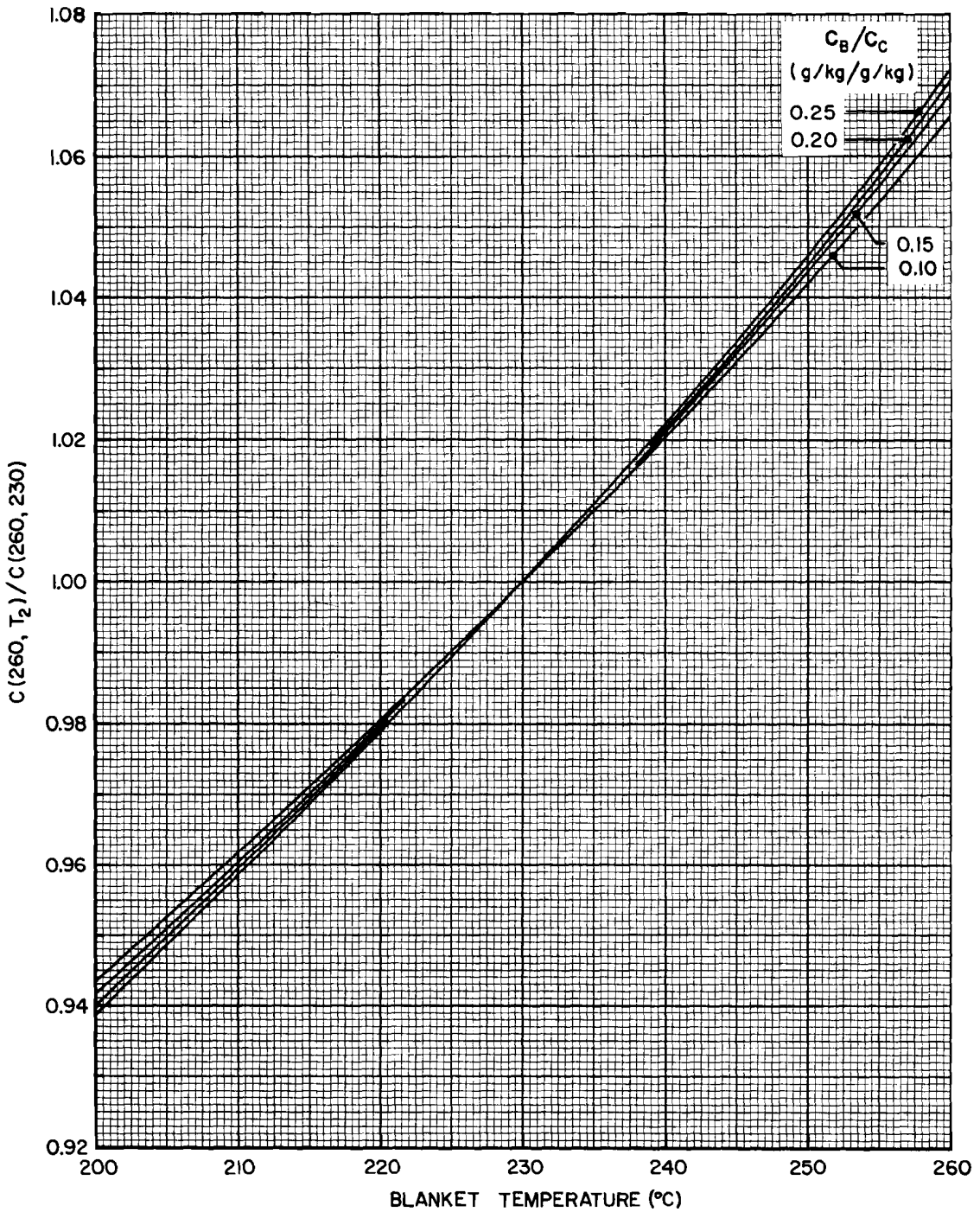

Fig. 14. Effect of Blanket Temperature on Critical Concentration Predicted by GNU Calculations.

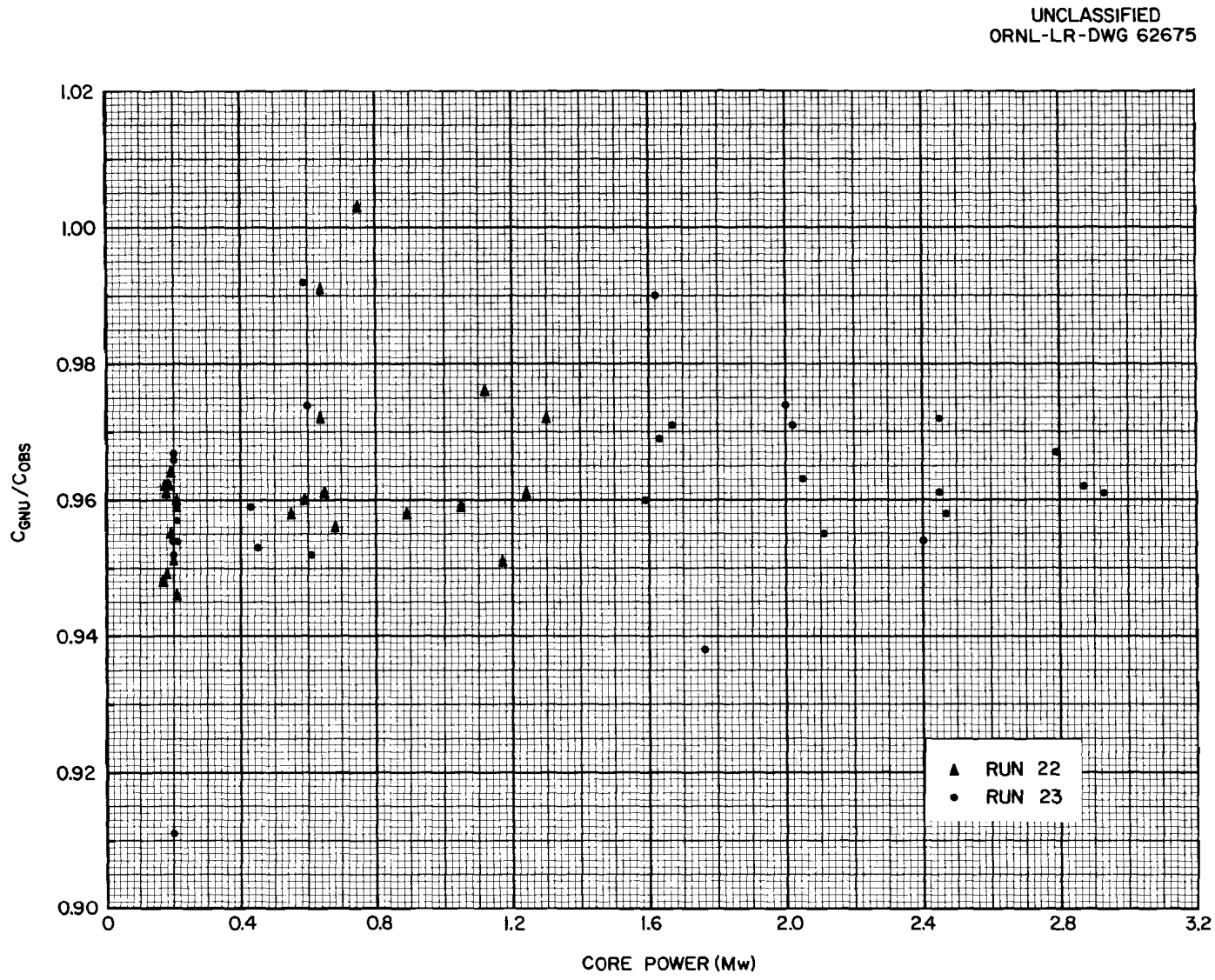

Fig. 15. Ratio of Critical Concentrations Predicted by GNU Calculations to Those Observed in Runs 22 and 23 vs Core Power Level.

Xenon Effects.--As xenon-135 concentrations in the reactor change following changes in the reactor power, the concentration required for criticality at a given temperature also must change. No allowance was made for xenon poisoning in arriving at the critical concentration from the reactor temperature and the GNU curves, so the ratio of the predicted critical concentration to the observed concentration should decrease as the xenon builds up. The expected effect is not large, however. J. Hirota, using iodine and xenon holdup factors which were based on early data,[31] predicted that equilibrium xenon at 5 Mw should cause a decrease in reactor nuclear average temperature of 3.3°C, if the uranium concentrations were unchanged. This is equivalent to an increase of about 2% in the critical concentration at a given temperature. Later data,[32] on the composition of xenon in the off-gas, indicated that the xenon effect should be less than half that calculated by Hirota. The comparison of predicted critical concentrations with observed concentrations in runs 22 and 23 was analyzed for possible effects of xenon as shown in Fig. 16. The variable against which the predicted/observed concentration ratio is plotted is the xenon poisoning expressed as a fraction of that which should exist after long operation at 5 Mw. This type of plot is different from a plot of the ratio against power because many of the samples with high xenon were taken at low power, several hours after a reduction from high power but while the xenon concentrations were still high. The xenon poisoning was estimated from the power history of the reactor using Hirota's calculated transients. (The shapes of the calculated transients were assumed to be correct, even though the absolute poisoning was too high.) The scatter in the data is such that no significant correlation can be seen in Fig. 16. The results are not inconsistent with the poisoning expected on the basis of the isotopic ratio measurements.[32]

Accuracy of Predictions.--It is apparent from Figs. 15 and 16 that the predicted critical concentrations are, on the whole, lower than the observed concentrations, thus indicating error in the predictions. There is no correlation between the error and either the core temperature (which ranged from 242 to 278°C) or the blanket/core concentration ratio (which ranged from 0.14 to 0.27 during runs 22 and 23). Lumping all of the run 22 and 23 points together, the ratio of predicted to observed concentrations was found to be 0.962 ± 0.014 (mean and standard deviation).[33]

Results of Run 24

When the fuel was replaced at the end of run 23, the enrichment was changed from about 83 to 93%, and the poisoning due to dissolved nickel and fission products was practically eliminated. As a result, it is necessary to treat separately the criticality data from run 24.

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[31] J. Hirota, Internal correspondence.

[32] W. D. Burch and O. O. Yarbro, HRP Quar. Prog. Rep. Jan. 31, 1960, ORNL-2920, p 9.

[33] The mean would be higher by 0.009 if the sample assays were used instead of the book values for the enrichment in arriving at the "observed" concentrations. See footnote 30.

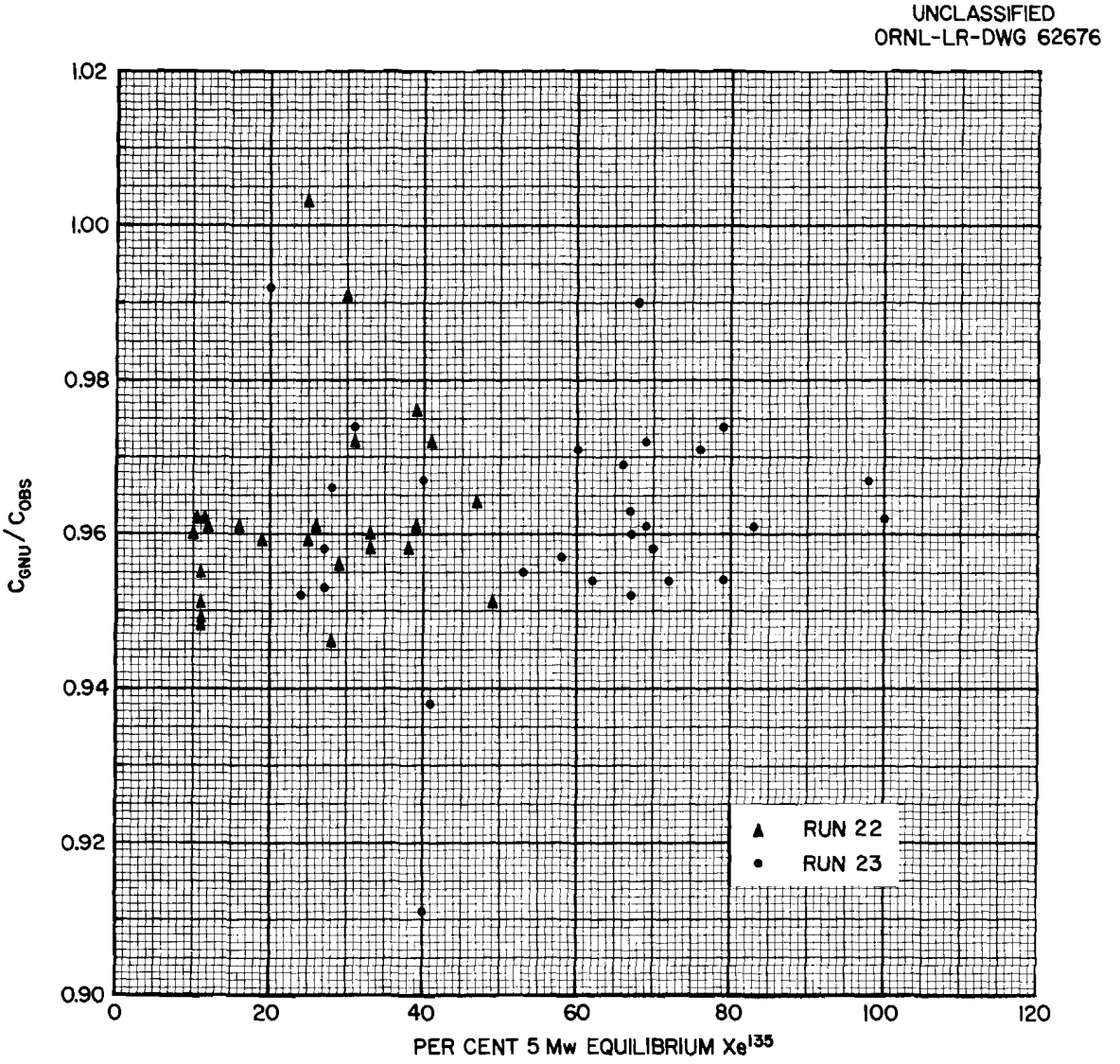

Fig. 16. Ratio of Critical Concentrations Predicted by GNU Calculations to Those Observed in Runs 22 and 23 vs Calculated Xenon Poisoning.

There was no detectable effect of noncirculating uranium in run 24, and again the data were not sufficiently accurate to give a value for the xenon poisoning.

Figure 17 serves to show all of the run 24 data. The predicted critical concentrations were obtained from Figs. 12 and 13, with no correction for the reduced poisoning in run 24. The "observed" concentrations were obtained from the sample results, corrected for analytical bias, and the enrichment.[34] There was no significant correlation of the ratio with any variable. Excluding the highest point,[35] the GNU curves give concentrations which are 0.997 ± 0.009 of those observed in run 24. (The mean would be 0.999 if the sample assays were used.)

Effects of Fuel Replacement

Comparison of run 24 criticality data with that of runs 22 and 23 shows that the critical concentration of U-235 decreased by about 3% as a result of the fuel replacement. Net reduction in the poisoning due to changes in nickel, copper, sulfur and uranium isotopes accounts for a change of 1.0% in the critical concentration. The remainder of the change, about 2%, is attributed to the removal of the fission products which had accumulated in the old charge.

Power Fluctuations

When the nuclear operation of the HRT was resumed in run 22, after the reversal of the core flow, two major differences in behavior from earlier operation were immediately apparent. The first was the behavior of the core inlet and outlet temperatures as the power was changed; as the inlet and outlet temperatures diverged, both decreased instead of diverging about a mean as had been normal before. This was expected as a result of tests in the hydraulic mockup, which had predicted that "short-circuiting" and recirculation of the fuel in the core would cause some regions to be much hotter than the core outlet. The other difference was the magnitude of the power fluctuations; they were about twice the size of those observed when the reactor operated with upward flow. Although the increased

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[34] In run 24 the ratio of analysis to actual uranium concentration in nine core control samples was 1.0042 ± 0.0126. The enrichment used was 92.89 weight percent U-235. Two samples taken during run 24 assayed 92.70 weight percent U-235. If this enrichment were used, all the ratios in Fig. 17 would be increased by 0.002.

[35] There is much independent (albeit circumstantial) evidence that the blanket concentration reported for this point was much lower than the true blanket concentration.

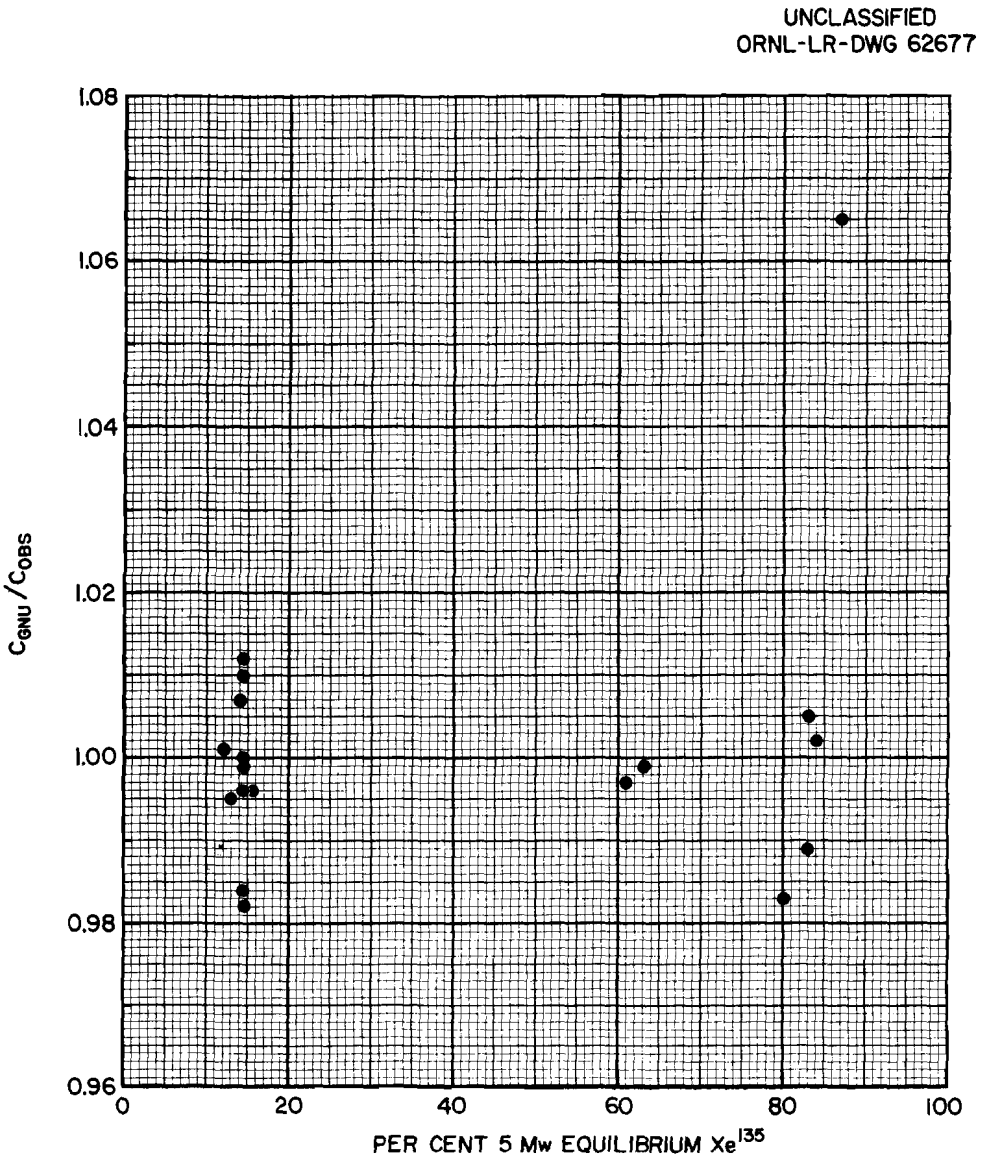

Fig. 17. Ratio of Critical Concentrations Predicted by GNU Calculations to Those Observed in Run 24 vs Calculated Xenon Poisoning.

turbulence in the core was strongly suspected as the cause of the larger fluctuations, caution was used in raising the power until fuel instability or other potentially harmful conditions could be ruled out as a cause.

Typical behavior of the fission power, or neutron level, with downward flow through the core is shown in Fig. 18, which is a photograph of a recorder chart taken in run 22. Many of the peaks are larger than what had been recognized as pips in earlier operation.[36] Unlike pips, the largest fluctuations appeared to be merely part of the general "noise," differing from the smaller fluctuations only in degree, not in cause. This impression was strengthened when the size-frequency distribution of the power peaks was determined. As shown in Fig. 19, the largest peaks appear to be part of the same statistical distribution as the smaller fluctuations. This was not true of pips.[37]

Investigation

The cause of the power fluctuations became the subject of extensive investigation. In run 22, various changes in operating conditions were tried to determine their effect on the fluctuations. Changes included: operating at various core temperatures and pressures, reducing core circulation rate, reducing blanket circulation rate, interruption of fuel and blanket feed pumps, manual control of letdown valve position, interruption of feedwater pumps, and manual control of the steam withdrawal valves. Some of the changes were necessarily of short duration. For others, namely the pressure, temperature and circulation rate changes, the reactor was operated for at least four hours at a given condition while data were collected. None of the changes had a noticeable effect on the fluctuations. To facilitate the analysis of the data, procedures were developed which made use of the large automatic digital computers in Oak Ridge. Through the efforts of R. K. Adams, provisions were made to record the neutron level in digital form on punched paper tape. (A voltage signal was transmitted to a digitizer and tape punch at X-10.) The tape was edited for mistakes by an ORACLE code, then was converted to punched IBM cards, which were used in analysis codes devised for the IBM-7090 by M. L. Tobias and D. R. Vondy.[38] Meanwhile, experiments were conducted by C. G. Lawson[39] in which the HRT hydraulic mockup, with provisions for salt injection and concentration detection, was coupled with

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[36] M. W. Rosenthal, S. Jaye and M. Tobias, Power Excursions in the HRT, ORNL 2798 (Feb. 15, 1960).

[37] Ibid., p 13.

[38] P. R. Kasten et al., HRP Prog. Rep. Dec. 1, 1960 to May 31, 1961, ORNL-3167, p 28-36.

[39] Ibid., p 14-21.

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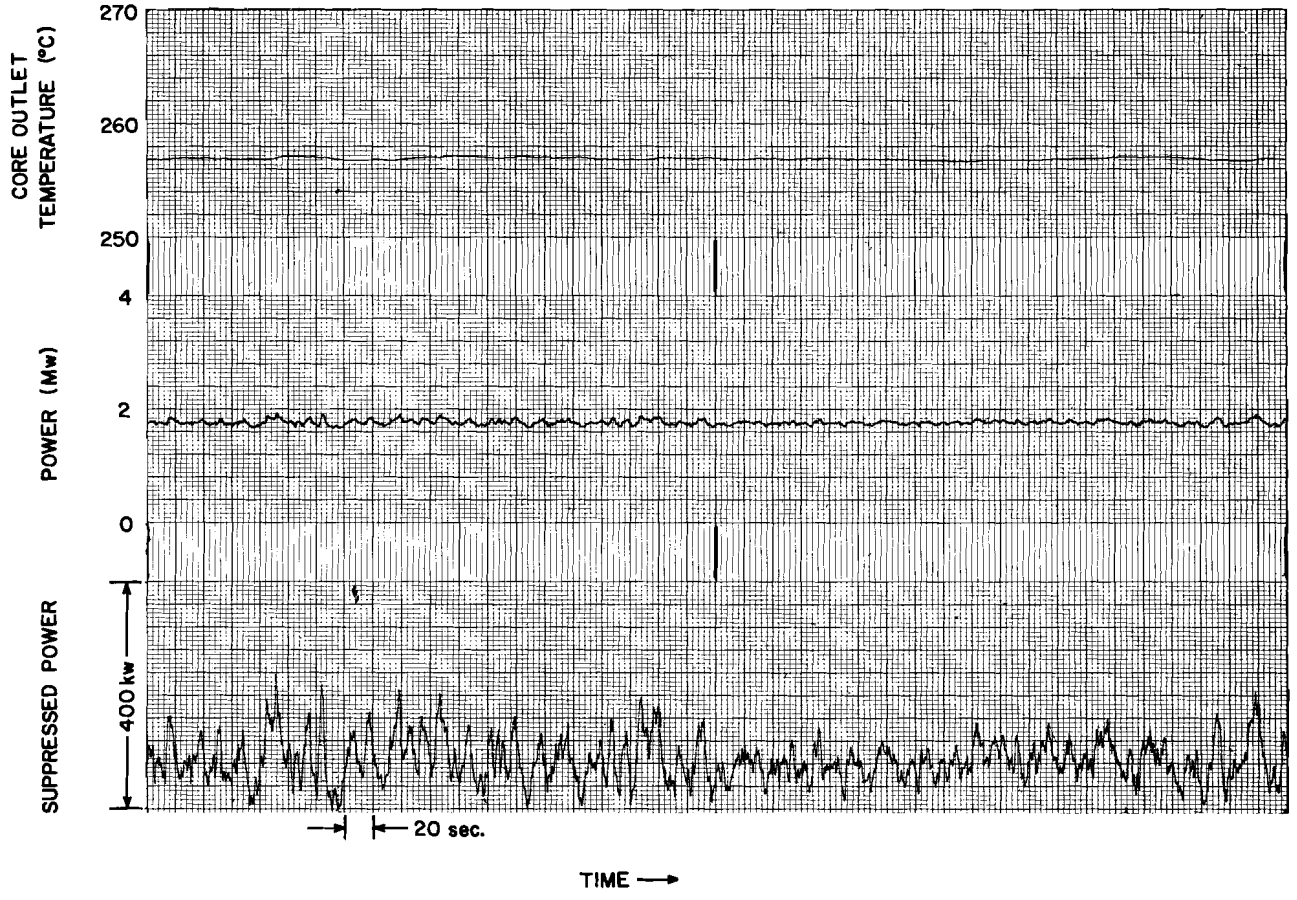

Fig. 18. Behavior of Neutron Level (Power) and Core Outlet Temperature at 1.8 Mw in Run 22.

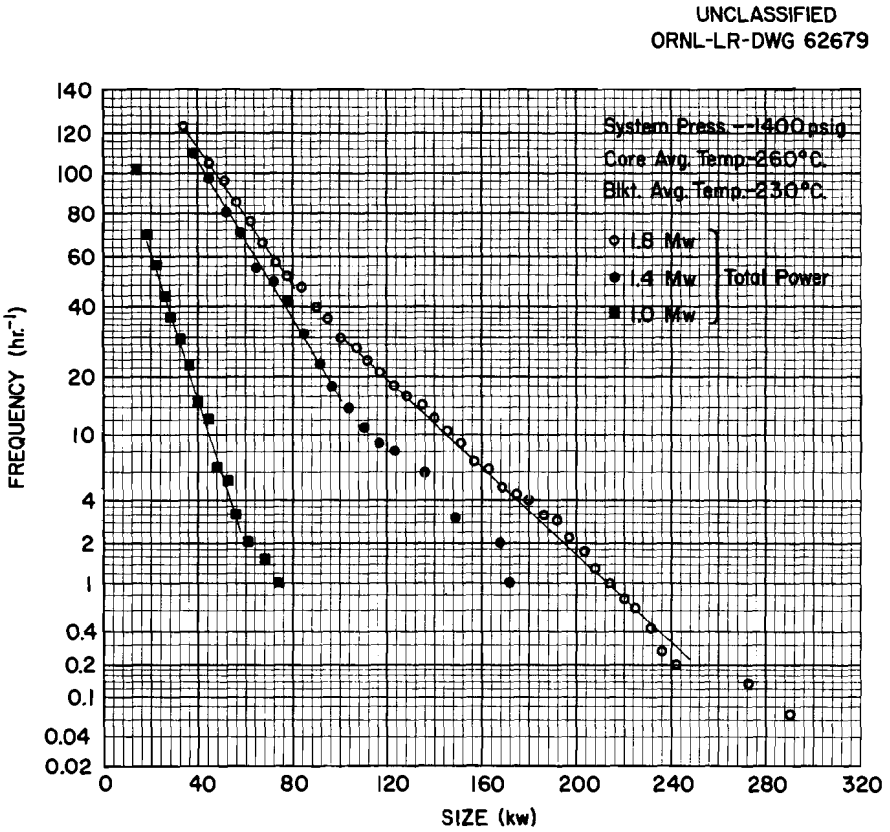

Fig. 19. Size-Frequency Distribution of Positive Power Fluctuations in Run 22.

an analog circuit simulating the neutron kinetic equations. These experiments produced simulated power traces which were compared with the actual power data from the HRT. Results of all these investigations point to the small fluctuations in the flow pattern in the core, with consequent fluctuations in the core average temperature, as being the cause of the neutron power fluctuations.

Operating Restrictions

The conclusion that the fluctuations in neutron level did not indicate potentially harmful conditions in the core had already been tentatively reached by the end of run 22. The fluctuations remained, however, a problem in the operation of the reactor; they would tend to conceal actual pips, or disturbances symptomatic of fuel instability, if such should occur. Some way was desired to recognize the symptoms quickly so that prolonged operation under conditions of fuel instability could be avoided. The approach which was adopted was to scrutinize the size-frequency distribution of the fluctuations for abnormal deviations. If such occurred, the power was to be lowered and the disturbances in question examined carefully to judge if the abnormalities were actually pips. The analysis of the fluctuations to provide a guide for operations had to be done manually, since the machine computations involved undesirable lag time in the data reduction.

In practice, the fluctuations in reactor period were more convenient to analyze than the power fluctuations, mainly because the mean of the inverse period was always zero, while the mean power was subject to slow drifts and had to be calculated for each section of chart which was analyzed. Furthermore, the size of inverse period peaks is more closely related to the size of reactivity disturbances than is the size of the power peaks. The size of the peaks in inverse period was found to follow a statistical distribution as did the power peaks. Figure 20 shows data at six power levels in run 23.[40] The effect of core power (proportional to the temperature rise of fuel in the core) is evident in Fig. 21, which is a crossplot of Fig. 20.

The frequencies plotted in Fig. 20 are average values measured over intervals of 6 to 22 hours. It was found that distributions based on two hours of data varied considerably from one two-hour interval to the next, even though, externally, there had been no change in operating conditions. The bands shown on the points at the highest powers in Fig. 21 are the 95%

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[40] The periods shown on this and Fig. 21, and the limits referred to in the discussion which follows, apply to periods indicated on RR-163, a Brown recorder. Because this recorder had a 4-second time constant, the shorter periods were not faithfully recorded. (Peaks which were recorded as 10-second periods on RR-163 were shown as 8-second periods on a Sanborn recorder which has a time constant on the order of milliseconds.)

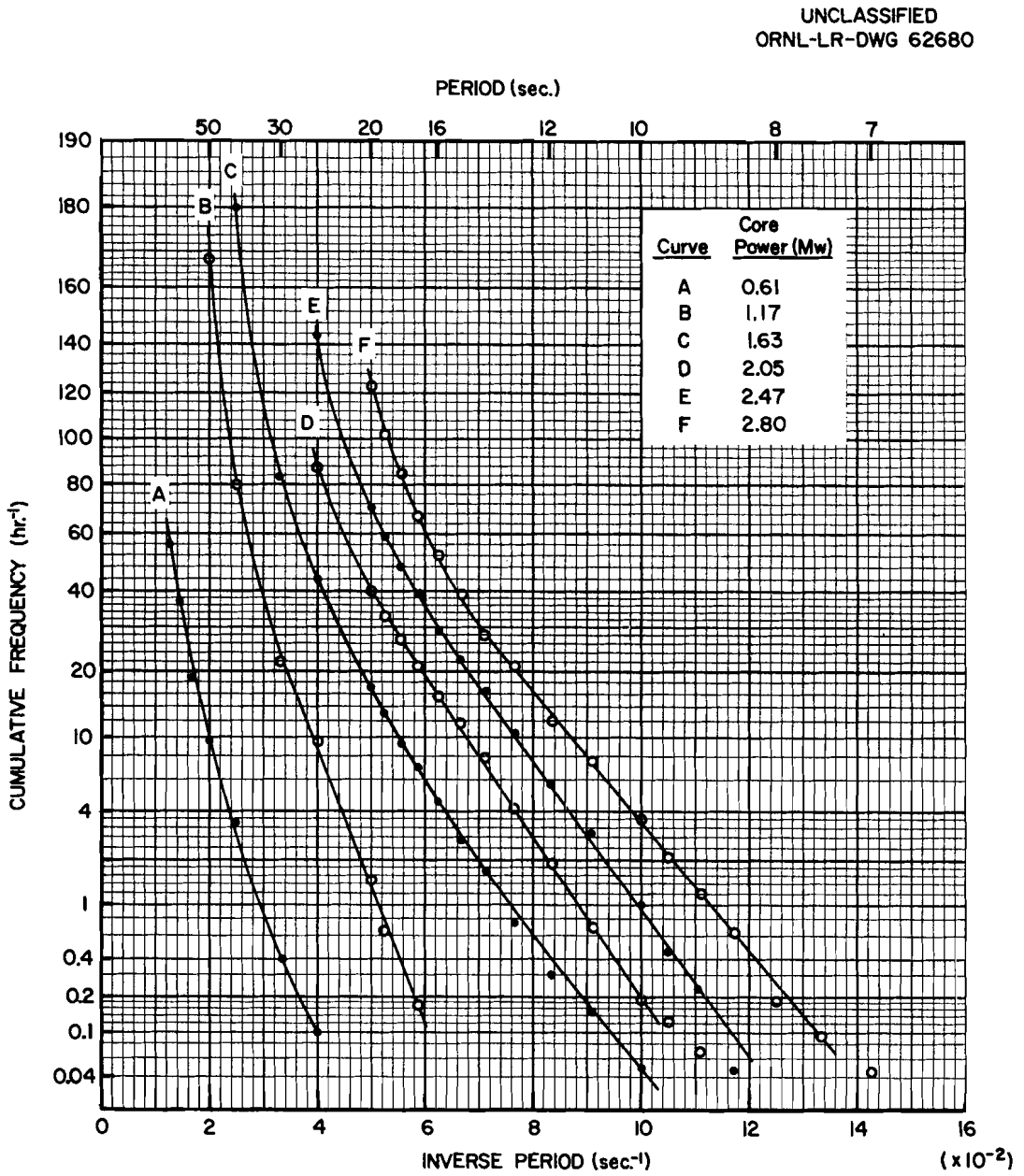

Fig. 20. Size-Frequency Distribution of Positive Peaks in Inverse Period in Run 23.

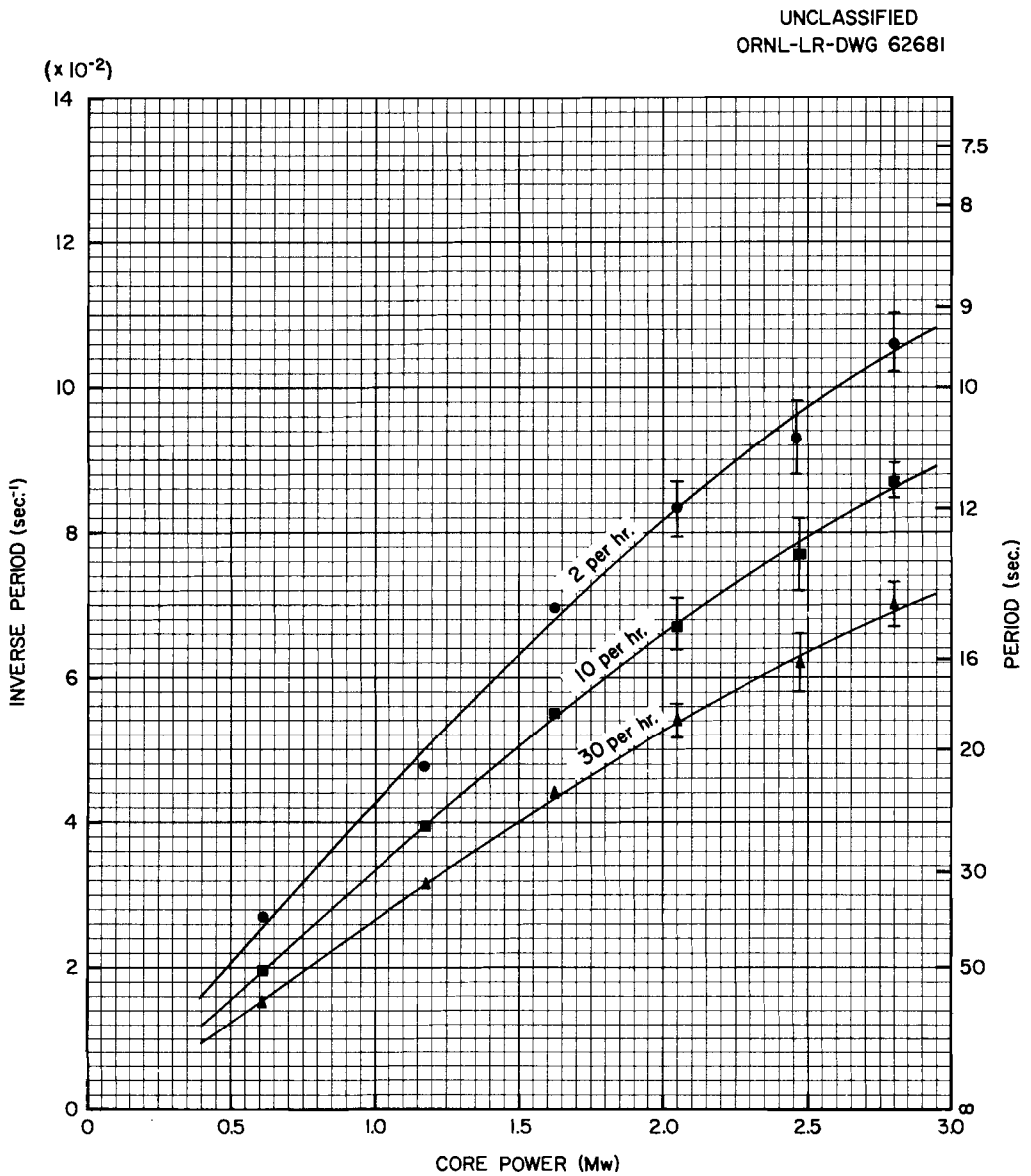

Fig. 21. Effect of Core Power on Size of Peaks in Inverse Period in Run 23.

confidence limits for two-hour intervals. In establishing the operating guides for detecting pips among the normal fluctuations, it was necessary to take into account this nonstationary nature of the statistical distribution.

The operating limitation followed in runs 23 and 24 was a specified maximum number of inverse period peaks greater than a certain value in any interval of specified length. If the specified frequency was exceeded, the power was to be lowered. In specifying the limits, a low frequency of occurrence was desirable, because counting of many events was impractical for operating personnel. Furthermore, a short time period was required since operation with separated uranium for extended periods was undesirable. A two-hour interval and a two-per-hour average frequency were chosen as a reasonable compromise between these requirements and the better statistics obtained with longer intervals and more frequent occurrences. The size limit on the inverse period peaks was based on the line representing the mean size at two per hour on Fig. 21 plus 1.5 times the 95% confidence spread. To illustrate, at a core power of 2.8 Mw (total about 5 Mw) the occurrence of peaks in the inverse period greater than 12.4 sec^{-1} (periods shorter than 8 seconds) more than four times in any two-hour interval was cause for lowering the power to heat loss.

Several times during runs 22 and 23 the operating limit on inverse period peaks was exceeded and the power was lowered. In every such case, analysis of the chart records showed a normal size-frequency distribution, but one which had shifted more than expected toward larger, more frequent fluctuations. These power reductions were the only way in which the power fluctuations hampered the operation of the reactor. The heat capacities of the fuel solution and the heat exchanger attenuated the fluctuations in fission rate, so that there were no detectable fluctuations in the reactor steam pressure or the reactor thermal power output.

Condensate Transfer

Change in Transfer Route

A continuous transfer of condensate from the fuel to the blanket low-pressure system is required to supply D_2O for the blanket purge. Before run 22, the condensate was transferred through the low-pressure system transfer line provided in the original reactor installation. This line enters the blanket feed line between the blanket dump tank and the blanket feed pump as shown in Fig. 22. Since make-up to the blanket dump tank is required to replace the condensate pumped by the two blanket purge pumps, the net flow in the feed line was toward the blanket dump tank. Thus any fuel solution let down to the dump tank from sampling or valve leakage remained in the dump tank and had to be returned to the high-pressure system periodically by interrupting the transfer and emptying the dump tank.

During the shutdown before run 22, a new line was installed to permit the condensate transfer to enter the top of the dump tank (as shown

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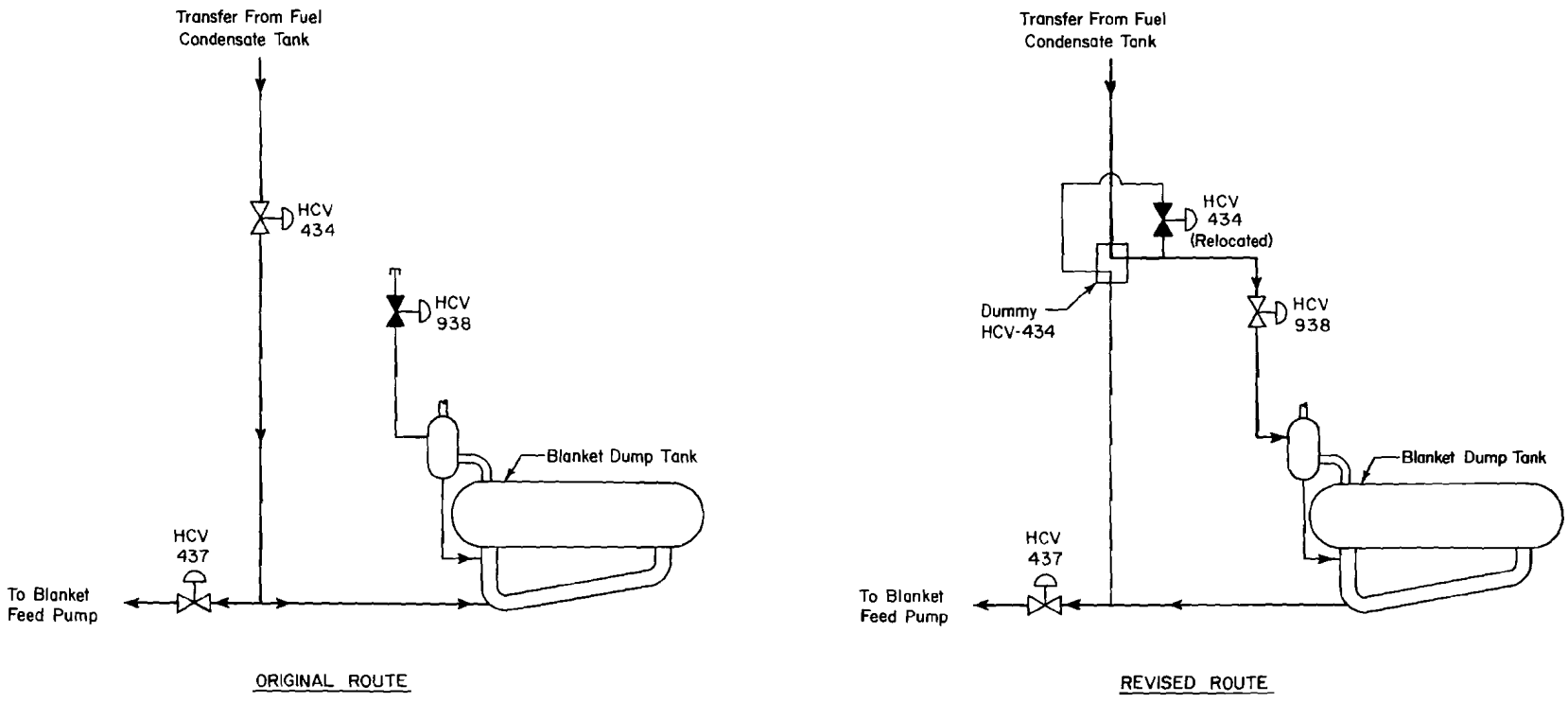

Fig. 22. Simplified Flow Diagram of Transfer System.

by the revised route in Fig. 22). One end of the line was connected to a dummy valve body with the required flow passages, which was installed in place of HCV-434.[41] The other end of the line was connected to the previously unused blanket chemical processing line at HCV-938. HCV-434 was reinstalled in a new position, but connected so that the function of the original transfer line was unchanged.

With the new transfer route in service, the blanket feed pump was supplied solely from the blanket dump tank, so that any fuel let down to the dump tank was continuously returned to the high-pressure system.

Control of Transfer

Two principal objectives in the control of the reactor low-pressure system are (1) maintaining the desired fuel-dump-tank weight (since this controls the fuel concentration in the high-pressure system), and (2) maintaining the transfer rate as required to supply the total blanket purge.

By run 21, in pursuit of these objectives, the following control system had evolved:

The fuel-dump-tank weight was controlled automatically by throttling one of the valves (HCV-333) in the transfer line. This determined the withdrawal of condensate from the fuel condensate tank; a second controller maintained the condensate weight fixed by throttling the valve (HCV-336) admitting condensate from the fuel recombiner condenser. To provide the driving force required for the condensate transfer, a pressure differential was maintained between the fuel and blanket low-pressure systems by periodically expelling gas from the blanket system (using steam pressure) and then closing the off-gas valves at the cold traps and lowering the steam pressure to produce a partial vacuum.

This control system had several shortcomings; the inclusion of the condensate weight controller in the fuel-dump-tank weight loop resulted in an unnecessary lag in the dump-tank weight control, the dump-tank weight had to be controlled manually whenever the transfer was interrupted or the fuel condensate weight was changed, and the transfer had to be interrupted periodically to renew the blanket vacuum.

To correct these shortcomings, an improved control system was devised and installed for run 22. The fuel-dump-tank weight was controlled directly by throttling the flow of condensate from the recombiner condenser to the fuel condensate tank via HCV-336. A controller was installed to maintain the pressure differential required between the fuel and blanket low-pressure systems, by throttling the flow of off-gas from the fuel cold traps. The fuel-condensate-tank weight was controlled automatically by varying the differential pressure controller setpoint to control the transfer rate. The maximum differential pressure was limited to prevent gas-binding of the fuel purge pumps.

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[41] S. E. Beall et al., HRP Quar. Prog. Rep. Jan. 31, 1960, ORNL-2920, p 16-17.

The new control system worked well. However, during run 22 it appeared that the changes in fuel-dump-tank pressure resulting from the action of the differential pressure controller were affecting the pumping rate of the fuel feed pump. On November 23, to eliminate this possible source of fluctuation in the feed rate, the differential pressure controller was replaced by a fuel-dump-tank pressure controller, operating in the same manner, by throttling the off-gas discharge. The fuel-dump-tank pressure was set (usually at 18 psia) to provide ample differential pressure (usually 4 psi) for driving the transfer. The fuel-condensate-tank weight was controlled automatically by throttling transfer valve HCV-333. The revised control system worked well, and it was used for the remainder of run 22 and succeeding runs.

Containment

Improvements in Secondary Containment

As part of a Laboratory-wide re-evaluation of the containment of radioactive processes, the HRT was surveyed, and, before run 22, a number of improvements were made in the secondary containment and the ventilation of reactor areas. These changes are summarized as follows:

1. In the reactor steam system, provision was made to vent back to the reactor cell any leakage through the steam block valves.

2. In the off-gas system, provision was made to operate the adsorber beds at subatmospheric pressure.

3. The adsorber-bed pit, the waste system, and the sampler cavities were vented to the stack through a silver-plated-mesh iodine trap. Provision was made to automatically vent the discharge of the trap to the reactor cell in the event of high radioactivity in the discharge gases.

4. A new stack fan and filter system was installed.

5. Special ventilation was provided for the chemical-plant and reactor control levels, the steam valve pit, and the waste-evaporator enclosure.

6. A radiation block valve was installed in the reactor cell sump discharge line.

Reactor Steam and Off-Gas Systems.--The changes in the reactor steam and off-gas systems have been described in preceding sections of this report.

Waste System and Sampler Vent.--Charcoal-bed iodine traps were formerly used in the vent lines from the main waste tank and from the sampler cavities. During the re-evaluation it was decided that charcoal beds were unacceptable for this application because of the possibility that they might become ignited, thus releasing all their accumulated activity. The two charcoal beds were replaced by a single silver-mesh

bed to serve as an iodine trap for both systems. To prevent a major discharge of gaseous activity in the event of a catastrophic failure of a sampler, for example, in which a large quantity of fuel solution was released in the sampler cavity, a block-and-vent system was installed in the discharge from the bed. In the event of a discharge of highly radioactive gas from the bed, the line to the stack would be blocked and a vent line to the reactor cell opened so that the activity would be confined to the reactor cell. An interlock was provided to prevent this action if the reactor cell pressure were higher than atmospheric pressure.

Secondary containment of the off-gas adsorber beds was also provided by this system. The off-gas adsorber beds are submerged in water in a pit, covered with concrete blocks and earth, but not sealed from the atmosphere. Sealing of the pit was impractical; however, a vent line was installed from the overflow line of the pit via the waste vent system to the iodine trap. A valve in the vent line was throttled to maintain a purge rate of 15 to 20 cfm in the air space above the water, so that any gaseous activity released in the adsorber pit would be carried into the waste system.

Stack Fan.--A higher-capacity dual-unit stack fan was installed to provide additional ventilation capacity (see below). The standby unit started automatically on failure of the operating unit. A new iodine trap and particulate filter were installed downstream of the original filter as additional protection against a hazardous release of activity.

Ventilation.--The secondary containment, in three areas in which there are numerous shield penetrations, was improved by sealing off the areas from the remainder of the building and providing a ventilating system to maintain them at a slight negative pressure. These were the reactor control levels, which were connected to the stack-fan system, and the chemical-plant control levels and the steam valve pit, which were each provided with independent exhaust fans discharging through iodine-traps and filters.

In addition, the secondary containment of the waste evaporator was completed by sealing the openings in the enclosure and providing an exhaust duct to the stack-fan system.

Cell-Sump Discharge.--A radiation monitor and block valve were installed on the cell-sump discharge line to minimize the possibility that highly radioactive solution (such as spilled fuel) could be inadvertently discharged from the reactor cell. The monitor was set so that the block valve remained open for the discharge of low-level waste (such as cell flood water) to the waste system. The hand valve in this line was normally kept closed during reactor operation.

Activity Releases

Both runs 22 and 24 were ended when fresh fission-product activity was detected in the reactor cell sumps, indicating that there were leaks in the primary containment.

On December 4, at approximately 2100, the cell air activity, as indicated by the monitors in the west instrument cubicle, started increasing (from <10 mr/hr). One hour later and after the monitors were calibrated, the indicated activity level peaked at about 40 mr/hr. A sample of the fuel shield sump revealed that fresh fission products were present. The quantity of solution in the sump was so small, however, that it was necessary to add water to the sump before a sample could be withdrawn. The blanket sump contained a considerable amount of water (approximately 1500 gal) because of a cooling water leak. Activity could not be detected in this water with portable instruments, but was found to be present in very dilute quantities when more sensitive detection methods were used. The reactor, which had been operating at 1.8 Mw, was made subcritical at 2230; dilution of the high-pressure system was started at midnight (ending run 22). The activity level of the monitors decreased after these steps were taken. The total gaseous activity in the cell reached a peak of 1 to 2 curies, with the gases being predominantly Xe-133 and Xe-135.

The leaking component was found to be the east diaphragm head of the fuel feed pump (see page 44). Decontamination of the pump and the immediate area freed 19 g of uranium, which was flushed to the waste system.

Airborne fission-product activity was detected in the reactor cell throughout runs 22 and 23. Since the container was sealed and the activity level never rose above 5 curies for the entire cell, this condition was not considered a hazard for continued operation. However, on February 8, the gaseous activity level increased to 10 to 20 curies and fresh fission-product activity was detected in the cell sumps, so run 24 was ended. The leak was located in a 1-in. tee below the fuel dump tank (see pages 26-32). A rinse of the area of the leak freed 180 g of uranium.

Each time the leaks occurred in the primary system, the reactor was shut down in a normal manner with no release of activity to the atmosphere.

Cell Leakage

Throughout runs 22, 23 and 24, the cell was maintained at half atmosphere, with leakage being an acceptable 3 to 5 liters/min. This leakage was corrected for a normal nitrogen purge of 3 liters/min and a demineralized cooling water leak of about 0.5 liters/min.

Flange Leakage

During the runs 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.

Radiation Levels

Radiation levels outside the reactor shield were negligible during operation. Typical values of radiation levels measured inside the shield are given in Table 2.

Internal Recombination Experiments

A significant problem in the operation of an aqueous, homogeneous reactor is the recombination of the gases formed by the radiolytic decomposition of the water. In the HRT essentially all[42] of this recombination is accomplished in the high-pressure systems with Cu^{++} (added as $CuSO_4$) as the recombination catalyst. It is highly desirable, in the HRT, that the internal recombination be sufficiently rapid so that the concentration of the gases in solution does not exceed the solubility limits and form bubbles. In order to limit the operation of the reactor to those conditions where radiolytic gas bubbles do not form, it is necessary to know the effectiveness of the catalyst under reactor conditions.

Recombination experiments were performed in runs 23 and 24 to try to determine the effects of acid level and fission and corrosion products on the effective recombination rate. The experiments in run 23 were carried out with fuel solution which contained significant concentrations of fission and corrosion products. In run 24, two sets of experiments were performed with relatively clean fuel but at different levels of acid and copper. Each set consisted of four or five experiments covering a range of core average temperatures. The experimental procedure that was followed has been described earlier (see pages 22-23).

In evaluating the experiments it was assumed that the recombination reaction is first-order with respect to deuterium concentration and that all of the recombination was due to catalysis by copper. It was further assumed that the solution rate constant was proportional to the copper concentration and that the copper specific rate constant followed the Arrhenius equation. The quantities evaluated were the two constants in the Arrhenius equation. These were expressed in terms of a specific rate constant at 250°C and the activation energy. Under the above assumptions, any of the effects sought would appear as differences in these quantities. Since there were two quantities to be evaluated, a single experiment could not give definitive results. However, any two experiments give a value for each of the terms. Thus, with five experiments, ten values are obtained for each constant, and, with four experiments, six values are obtained.

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[42] Since internal recombination requires a finite concentration of the reactants, a small amount of gas, proportional to the concentration in solution and the fuel letdown rate, escapes high-pressure recombination and must be handled by the low-pressure recombiners.

Table 2. Radiation Levels Inside Reactor Cell

Run No. 22

Date	Time	Reactor Status	Gamma Dose Rate (r/hr) RM-2	RM-3	RM-4	RM-5	RM-6
Nov. 6, '60	2140	Before power. Fuel in HP system.	24	36	15	13	17
Nov. 13, '60	2130	After operating 150 hr at HL power.	2,600	5,500	3,700	265	4,400
Nov. 16, '60	2130	Operating at 1 Mw	6,300	13,500	8,400	670	14,400
Nov. 27, '60	2130	Operating at 1.8 Mw.	12,000	24,500	15,000	1,250	29,000
Dec. 4, '60	2130	Operating at 1.7 Mw.	6,900	17,800	16,300	1,160	30,500
Dec. 5, '60	2130	22 hr after shutdown. Fuel in FDT.	460	1,200	2,300	240	4,200
Dec. 6, '60	2130	46 hr after shutdown. Fuel in FDT.	95	720	1,150	300	2,170
Dec. 10, '60	2130	142 hr after shutdown. Fuel in FST.	200	510	580	46	1,200

Date	Time	Power (Mw)	Gamma Dose Rate / Reactor Power (r/hr / kw) RM-2	RM-3	RM-4	RM-5	RM-6
Nov. 13, '60	2130	0.33	7.88	16.67	11.21	0.80	13.33
Nov. 14, '60	2130	1.0	6.20	14.40	8.00	0.56	11.00
Nov. 23, '60	2130	0.94	6.60	13.83	---	0.60	14.36
Nov. 27, '60	2130	1.86	6.45	13.17	8.06	0.67	15.59
Nov. 30, '60	2130	0.80	6.25	15.00	11.63	0.80	21.25
Dec. 4, '60	2150	1.73	3.99	10.29	9.42	0.67	17.63

Location of radiation monitors:

RM-2 At blanket circulating pump.
RM-3 Above reactor thermal shield.
RM-4 Above fuel dump tanks.
RM-5 Above blanket dump tanks.
RM-6 Above fuel feed pump (exposed to direct radiation from the reactor through opening in thermal shield).

Table 2. Radiation Levels Inside Reactor Cell (Con't.)

Run Nos. 23 and 24

	Date	Time	Reactor Status	Gamma Dose Rate (r/hr) RM-2	RM-3	RM-4	RM-5	RM-6
Run 23	Dec. 15,'60	2130	Before power. Fuel in FDT. D_2O in HP system.	180	500	1,000	90	700
	Dec. 17,'60	2130	Operating at 1 Mw.	6,050	13,800	7,750	890	13,400
	Dec. 18,'60	2130	Operating at 1.8 Mw.	10,800	24,000	13,000	1,300	21,000
	Jan. 6,'61	2130	Operating at 5 Mw.	30,000	61,000	39,000	3,800	77,000
	Jan. 15,'61	2130	56 hr after sub-critical. Fuel in FST. D_2O circulating in HP system.	530	2,200	2,000	85	3,900
Run 24	Jan. 17,'61	2130	Before power. Fuel in HP system	1,600	1,000	2,200	260	2,950
	Jan. 20,'61	2130	Operating at heat-loss power.	3,300	7,100	5,400	360	12,000
	Jan. 26,'61	2130	Operating at 2.8 Mw.	17,000	36,000	18,000	1,600	36,000
	Feb. 6,'61	2130	Operating at 4.8 Mw.	29,000	59,000	33,000	3,100	55,000
	Feb. 8,'61	2130	4 hr after sub-critical. Diluting to FDT.	2,600	4,700	3,900	270	28,000
	Feb. 10,'61	2130	52 hr after shutdown. Fuel in BST.	550	1,200	1,400	280	2,500
	Feb. 12,'61	2130	100 hr after shutdown.	430	800	880	270	2,170

	Date	Time	Power (Mw)	Gamma Dose Rate / Reactor Power (r/hr / kw) RM-2	RM-3	RM-4	RM-5	RM-6
Run 23	Dec. 17,'60	2130	1.0	6.05	13.80	7.75	0.89	13.40
	Dec. 18,'60	2130	1.8	6.00	13.33	7.22	0.72	11.67
	Dec. 21,'60	2130	2.62	5.53	12.60	7.25	0.76	16.03
	Jan. 1,'61	2130	4.25	6.12	12.59	7.76	0.75	14.94
	Jan. 6,'61	2130	4.98	6.02	12.25	7.83	0.76	15.46
Run 24	Jan. 26,'61	2130	2.81	6.05	12.81	6.41	0.57	12.81
	Jan. 27,'61	2140	3.81	6.25	13.12	7.24	0.64	13.65
	Feb. 6,'61	2130	4.77	6.08	12.37	6.92	0.65	11.53

Calculation of the experimental results was based on a flow model which considered the temperature distribution in the external piping of the core loop as well as in the core itself. The core temperature distribution was that predicted from experiments on a full-scale hydraulic mockup of the vessel. Pressure variations around the loop resulting from friction losses were also considered. Preliminary calculations indicated that, with downward flow through the core, radiolytic gas bubbles would appear in the core vessel itself where they would be detectable by their effect in lowering the reactor critical temperature. Letdown of radiolytic gas to the low-pressure system could not be expected and was not observed.

The results of the recombination experiments performed in runs 23 and 24 are presented in Table 3. The average concentrations of acid and copper are given for each group of experiments. For the effective specific rate constants, both the average of the six or ten independent values and the range covered by the individual values are given. These numbers express the absolute reaction rate, in gram-moles per second of D_2 reacted per liter of 250°C solution, at unit molar[43] D_2 concentration and unit molar Cu concentration. The average apparent activation energy for the reaction is given in kilocalories per gram-mole of D_2.

The results of the last two sets of experiments indicate a lower effective copper activity at the higher concentrations of acid and copper. If it is assumed that this effect is due only to the acid level and the resultant acid dependence is linearly extrapolated to the acid level of the old fuel experiments, an enhancement of the catalytic activity by the fission and/or corrosion products is indicated. There was no detectable effect on the catalytic activity due to short-term (up to 12 hours) operation at power levels just below the bubble point.

FUEL CHEMISTRY AND CORROSION

Changes in Fuel During Operation

Mixing Experiment

In the mixing experiment at the start of run 22, the reactor was operated for one week with dilute chromic acid in light water (see page 12). The chromic acid served both to pretreat stainless steel parts of the system and to provide a chemical tracer for measurement of the rate of mixing between core and blanket. The chromic acid was stable in

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[43] In this context, molar is defined as gram-moles of solute per liter of solution at the temperature in question.

Table 3. Results of Recombination Experiments in Runs 23 and 24

Fuel Charge	No. of Expts.	Cu Concentration (g-moles/kg-D_2O)	Acid Concentration (g-moles/kg-D_2O)	Specific Rate Constant at 250°C $\left(\frac{sec^{-1}}{g\text{-mole Cu/l}}\right)$		Activation Energy (K-cal/g-mole)
		Average	Average	Average	Range	Average
Old	5	0.0138	0.0288	2.04	1.99-2.07	23.8
Fresh	4	0.0077	0.0195	2.25	2.16-2.32	22.1
Fresh	5	0.0133	0.0239	2.00	1.93-2.27[a]	22.8

[a]One value, out of ten, was above 2.02.

the absence of radiations due to fission. The core concentration was about 2000 ppm Cr and at the end of the operation, the core solution also contained about 70 ppm U, 30 ppm Ag and 20 ppm Cu. There was no trace of Ni. The solution was removed and the system was rinsed before the heavy water fuel solution was removed from the storage tanks to begin nuclear operation.

Acid Additions in Runs 22 and 23

During the course of runs 22 and 23, sulfuric acid was added to the fuel on three separate occasions. Some of this acid was required to replace that consumed by the corrosion of stainless steel which resulted in the addition of metal cations to the solution. The remainder was added to provide a greater margin of safety against phase instability due to heavy-liquid phase separation or hydrolytic precipitation, both of which are favored by low acid levels. Improved flow in the core vessel was expected to prevent generalized overheating, but there was some concern that phase separations could result from localized overheating in crevices around the patches and in the deepest pits of the Zircaloy core wall. It was therefore decided that the acid concentration should be raised as high as possible without inducing serious corrosion of the stainless steel. The allowable maximum free acid concentration, set as a result of mockup experiments,[44] was 0.035 molal. This limit was reached after three additions of acid: 7.0 moles on November 8, 4.8 moles on November 18, and 4.8 moles on December 26, 1960. The last addition was made before the reactor power was raised above 3.4 Mw.

Removal of Fuel After Run 23

With each addition of acid, there was a detectable increase in the generalized stainless-steel corrosion rate. Early in January 1961, the core (0.09 molal $SO_4^=$) and the blanket (0.02 molal $SO_4^=$) solutions, because of the buildup of nickel, reached concentrations equivalent to saturation at 300°C.[45] In order to remain below the 5-component saturation limits at 300°C, it would be necessary to raise the total sulfate ion inventory approximately 2.5 moles for each mole increase of nickel inventory. But further acid addition was undesirable from the standpoint of increased corrosion. (Any sizeable addition would have pushed the concentration above the 0.035-molal limit which had been set.) Therefore the decision was made to change the fuel solution.

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[44] J. E. Jones, Jr., HRT Mockup Runs CS-25 and CS-26, ORNL CF-61-4-96 (Apr. 26, 1961).

[45] W. L. Marshall and J. S. Gill, "Aqueous Systems at High Temperature, III. Investigations on the System UO_3-CuO-NiO-SO_3-H_2O at 300°C," Journal of Inorganic and Nuclear Chemistry (In Press, 1961).

The operations of concentrating and storing the old fuel are described on page 23. After these operations, samples from the core and blanket high-pressure systems and the core and blanket dump tanks showed that 0.19 kg of uranium remained in circulation. It was assumed that the same fraction of the other solution components also remained.

Operation in Run 24

The addition of new fuel was made in two parts. The first contained all of the uranium, but only half of the ultimate copper. After recombination experiments at low copper concentrations, the remainder of the copper, with additional acid, was added to bring the copper-to-U-235 ratio back to the level it had been in run 23.

The new fuel was safely below the saturation limits at 300°C, at a sulfuric acid concentration in the core solution of only 0.025 molal. This was possible because the nickel, which had reached about 0.015 molal in the core, had been removed, and the critical concentration of uranium had been lowered by about 12%. As explained on page 66, the critical concentration of U-235 decreased because of poison removal, but the main effect on total uranium was the increase of enrichment from 83 to 93%. (The low enrichment was caused partially by burnup of U-235, mostly by the isotopic-dilution experiment in run 18.)

The increase in the enrichment of the fuel between runs 23 and 24 is shown in Table 4, which presents the results of isotopic assay of samples taken during runs 22, 23 and 24. The changes in composition were caused by the addition of uranium containing 7014 g U-235, 402 g U-238, 77 g U-234 and 38 g U-236. From the change in the U-235/U-238 ratio, it was calculated that the added uranium mixed with 0.39 kg of total uranium, 0.20 kg more than was left in solution when the old fuel was stored. Presumably the difference was due to exchange between the dissolved uranium and the "permanently deposited" uranium.[46] (The difference between book and physical inventories, attributed to uranium retained in scale or other deposits, was 0.9 kg at the time of the addition.)

Reactor Power and Fuel Stability

There was no evidence of power-dependent fuel instability at any time during runs 22, 23 or 24. There were no reactivity disturbances attributable to fuel instability (see pages 66-71) nor did the comparison of critical and circulating U-235 concentrations show any effect of non-circulating uranium (page 59 et seq. and Fig. 15). The physical inventories likewise showed no indications of fuel instability.

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[46] H. B. Piper, S. R. Buxton and P. N. Haubenreich, A Study of Uranium Isotopic Ratio Information from the HRT, ORNL CF-60-10-102 (Oct. 27, 1960).

Table 4. Isotopic Composition of Uranium in Samples Taken During Runs 22, 23 and 24

Date	Sample	Assay (atom percent)			
		U-234	U-235	U-236	U-238
Nov. 8, 1960	RFH-22-141	0.93	82.83	1.30	14.94
Jan. 13, 1961	RFH-23-352	0.95	82.79	1.46	14.80
Jan. 13, 1961	RFH-23-353	0.93	82.85	1.45	14.76
Jan. 17, 1961	RBH-24-21	0.98	92.75	0.54	5.72
Jan. 18, 1961	RFH-24-25	1.02	92.77	0.55	5.67

The physical inventories of fuel solution constituents for runs 22 and 23 are plotted against operating time in Fig. 23; similar data for run 24 are given in Fig. 24. The reactor power level at the time of each core and blanket sample pair is also shown. The run 22 and 23 inventories do not include solution in the blanket dump tanks, but the run 24 inventories do. Some solute was held in these tanks because of leakage through the blanket dump valve. Because of the continuous purge through the blanket dump tanks, the amount of solute in them reached a steady state, which depended on dump tank weight and dump valve leak rate. Leakage to the blanket dump tanks also affected the calculated amount of solute in the fuel dump tanks, because it reduced the letdown-to-feed ratio in the fuel system. In run 24 the blanket dump valve leak rate was measured and taken into account. In the earlier runs it was not. The accumulation in blanket dump tanks was first recognized as appreciable when the tanks were pumped out at 559 hours in run 23 (see Fig. 23) and the NAT rose 4°C, indicating the recovery of about 200 g of uranium. The inventories in the two days preceding the pump-out were the lowest in the run, indicating that the holdup was increasing. To minimize the holdup, the blanket dump valve loading pressure was increased to the maximum, and the blanket dump tank weight, which had been running around 860 lb, was reduced to 250 lb, by storing some D_2O in the blanket storage tanks.

Figure 25 is a plot of circulating uranium inventory in run 23 against core power. The two low inventories at 2.79 and 2.87 Mw are the two in which there was probably the greatest error because of the blanket dump tank holdup. This plot shows no correlation of inventory with power, which would indicate power-dependent losses. Run 24 data is less conclusive, because of error resulting from the letdown heat exchanger bypass leakage, but there is not evidence of fuel instability.

The effects of core pressure and temperature on fuel stability were not investigated. With the exception of the last three inventories in run 23 and the first seven inventories in run 24, which were taken at 1000 psig during recombination experiments, all of the inventories were at a core pressure of 1400 psig. The core temperature at the time of high-power inventories ranged only from 258 to 269°C.

Corrosion of Stainless Steel

The generalized corrosion rate of stainless steel, based on the area of the high-pressure systems, is shown in Figs. 23 and 24 for runs 22, 23 and 24. The rates were calculated from the rate of change of the ratio nickel/copper in the physical inventories. In runs 22 and 23, the corrosion rate increased from 0.44 to 0.74 mpy as the acid concentration was increased. In run 24 the calculated rate was 0.45 mpy.

Although the fuel letdown heat exchanger was not examined to verify the nature of the leak between the feed and letdown streams, it is highly probable that there was penetration by localized corrosion. Such a failure was not unexpected, because of experience with the HRT mockup.[47]

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[47] R. E. Wacker and J. C. Griess, Summary of Corrosion Data for HRT Mockup Operation Period Ending November 5, 1956, ORNL CF-57-5-71 (May 22, 1957).

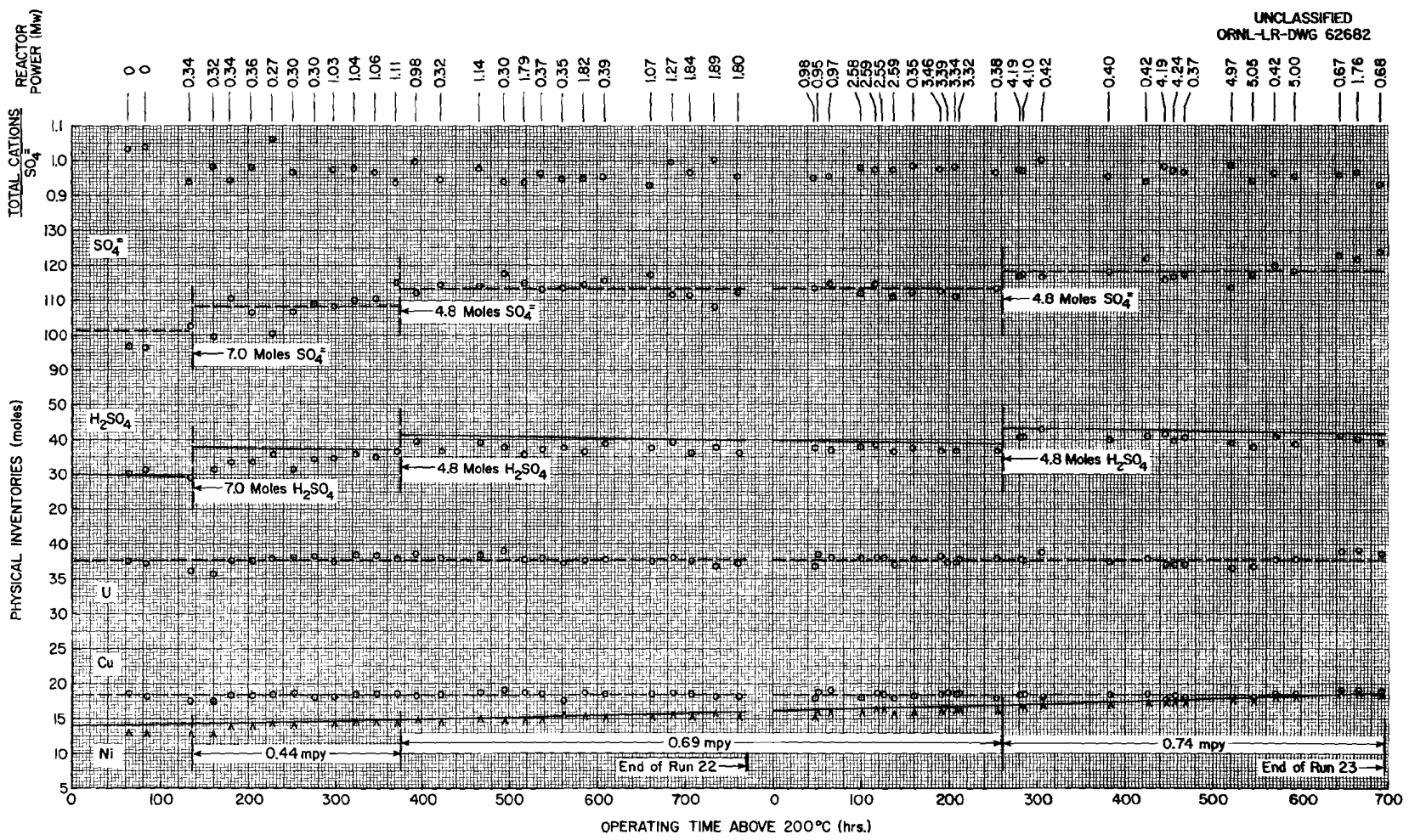

Fig. 23. HRT Physical Inventories in Runs 22 and 23.

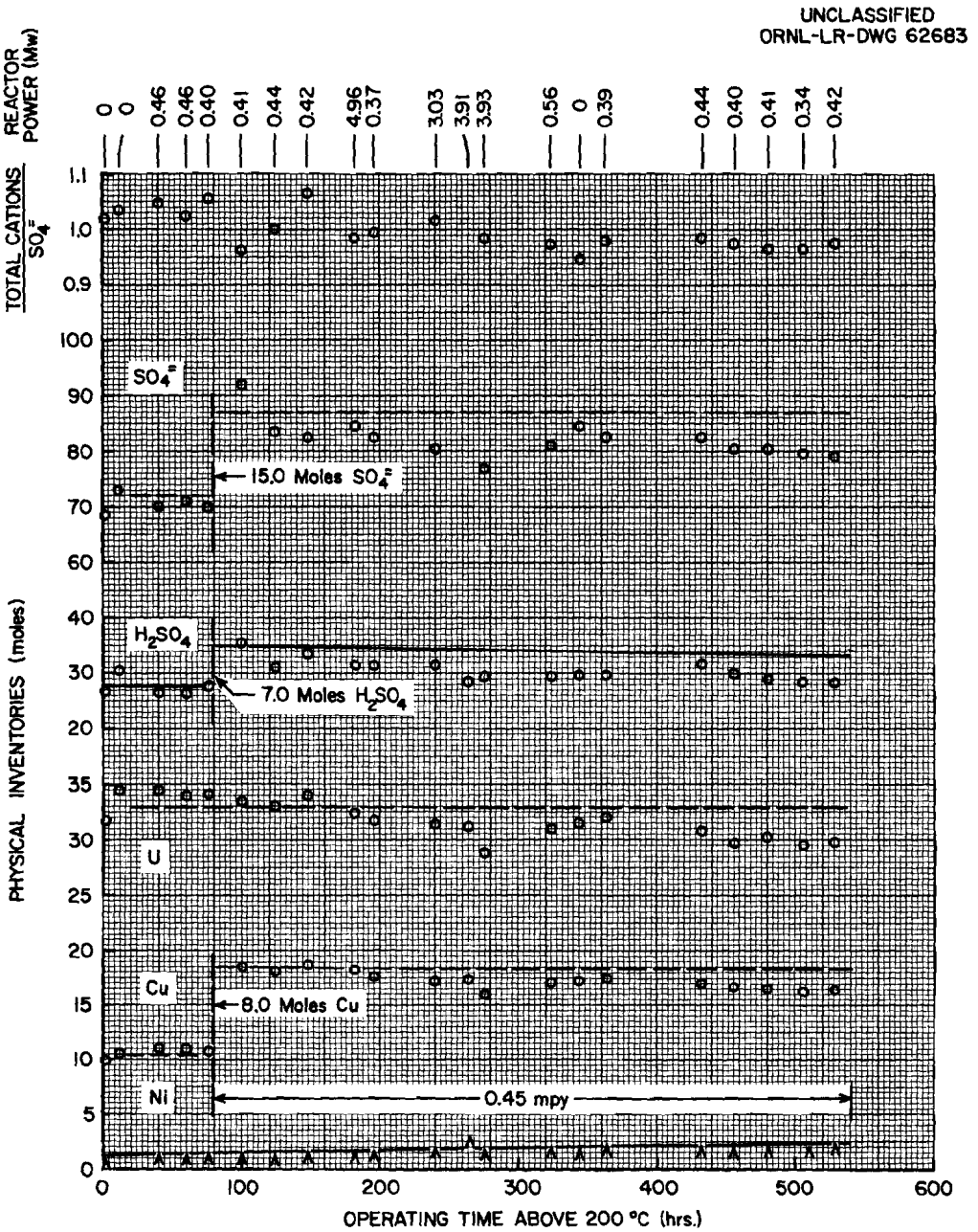

Fig. 24. HRT Physical Inventories in Run 24.

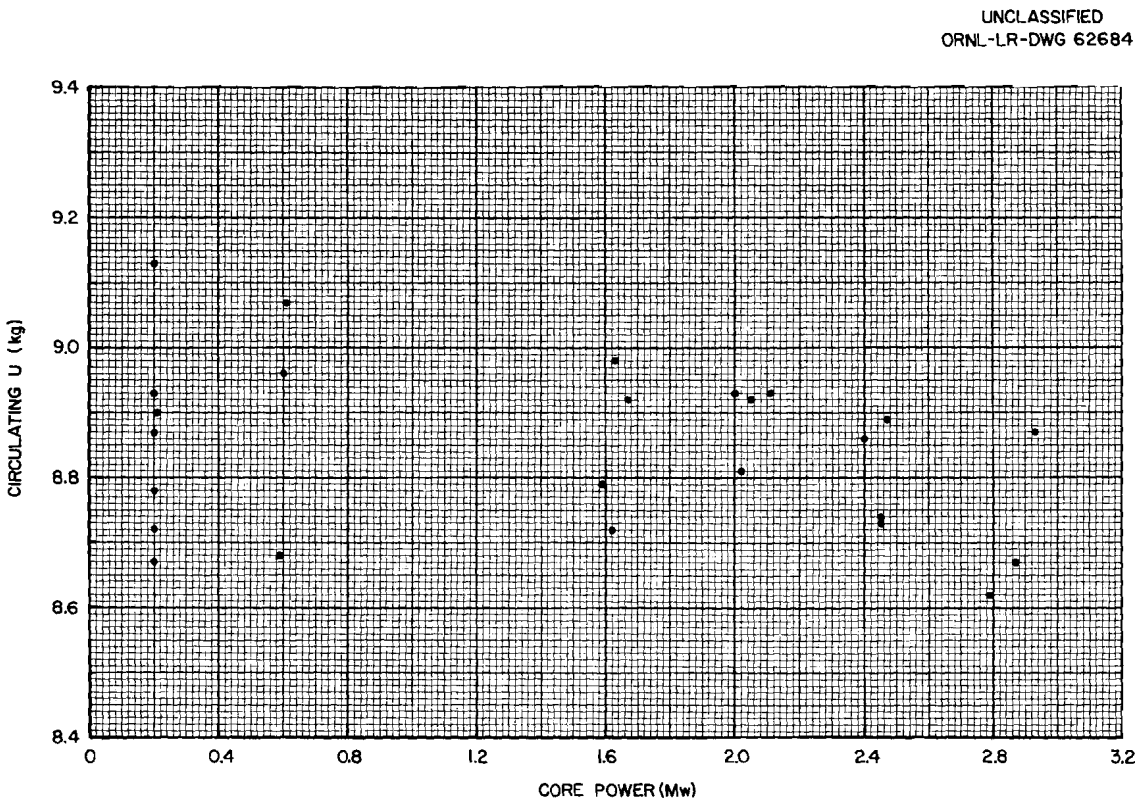

Fig. 25. Plot of Circulating Uranium Inventory vs Core Power in Run 23.

In the regions of the mockup letdown heat exchanger, which were at 200 to 225°C, severe pitting attack was observed with rates as high as 75 mpy. The susceptibility of 347 stainless steel to attack by uranyl sulfate solutions in this temperature range has been well established in loop studies.[48] Advanced reactor designs, for long life, have substituted titanium for stainless steel where such temperatures are encountered.

Precipitation in Fuel Samples

On January 3 and 4, 1961, it was observed that small amounts of finely divided precipitate appeared in the reactor solution samples after they had been stored in the analytical hot cells for some hours. The phenomenon occurred in the samples at 4.19, 4.24 and 0.37 Mw after 440 hours of operating time in run 23. In contrast to the similar event in run 21,[49] the precipitate appeared in the core as well as the blanket samples and consisted primarily of copper, rather than copper and iron, according to spectrographic analysis.

The solids consisted of two phases,[50] which could not be identified by X-ray analysis. One was colorless with a refractive index less than 1.7 and the other was red with a refractive index greater than 1.7. Both phases were isotropic and transparent. The particles ranged from 2 to 5 microns in diameter.

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[48] J. C. Griess et al., HRP Quar. Prog. Rep. July 31, 1957, ORNL-2379, pp 71-72.

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

[50] C. F. Weaver, Internal correspondence.

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