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ABSTRACT

This report presents a detailed study of HRT Run 14, which was abruptly terminated by the formation of a hole in the Zircaloy-2 core tank. In this run the reactor was operated at high temperature and pressure for ten days from March 25 to April 4, 1958. Nuclear power operation extended from March 29 until April 4, when fuel solution leaked into the heavy-water blanket. The maximum power level attained was 6.4 Mw.

The reactor equipment performance was satisfactory, except for the core vessel, but the nuclear operation was characterized by a power-dependent fuel solution instability, indicated by the critical temperature behavior, chemical studies, and power surges which appeared at higher power levels.

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PRELIMINARY OPERATIONS

Maintenance and Inspection

Run 13, ⁽¹⁾ in which the HRE-2 was first operated at power, was terminated because of failure of Fiberglas insulation on the power wiring in the reactor cell. With the reactor shut down, the cell was flooded with water to provide shielding and about 1000 ft. of new magnesia-insulated power wiring was installed.

Thirty-one days after the end of power operation a radiation survey of the submerged reactor equipment was made. Around most of the piping, the gamma intensity was 15-30 r/hr. The only exceptions were the fuel feed line sediment pot, where the radiation was 1100 r/hr, and the line connecting the fuel pressurizer with the circulating system, which read 500 r/hr. These high levels were thought to be an indication of deposited solids, most of which were formed during the period of rapid corrosion in Run 13. ⁽¹⁾

In order to explore more thoroughly the effects of the Run 13 corrosion, it was decided to open the core inspection port and to examine the interior with periscopes. A small mound of solids was seen in the pressurizer line and more solids were piled in the core around the periphery of the upper diffuser screen. Samples of the solids from the core and from the pressurizer line were obtained. ⁽²⁾ The thickness of the pipe wall at the point of solids deposition was measured with an ultrasonic device and was found to be unchanged.

While the cell was flooded, the fuel circulating pump was removed and corrosion specimens in its discharge pipe were taken out for inspection. ⁽²⁾ The pump, with new specimens, was then replaced under water. The fuel let-down valve, which had begun to leak during Run 13, was replaced, also under water, without undue difficulty.

Isolation of Original Fuel Charge

After maintenance and inspection work was completed, the cell was drained and the fuel was returned from the storage tanks where it had been held during the shutdown period. The fuel solution at this time contained about 0.6 kg of nickel, a result of stainless-steel corrosion during Run 13. The high concentrations of nickel raised questions of fuel stability. Furthermore, the normal scatter in measurements of this much nickel would

⁽¹⁾J. R. Engel et al., "Summary of HRE-2 Run 13 (Initial Power Operation)," ORNL CF-58-10-115 (Oct. 29, 1958).

⁽²⁾G. H. Jenks, A. R. Olsen, and W. C. Yee, "Examinations of Specimens and Scales Taken From the HRT Following Runs 13 and 14," ORNL CF-58-9-37 (Sept. 11, 1958).

prevent the accurate determination of the corrosion rate from nickel build-up rate if the corrosion occurred at a moderately low rate. Because of these considerations it was decided to begin Run 14 with a fresh charge of fuel. The old charge was to be stored in the fuel storage tanks until it could be removed for chemical processing. Thus on March 23, the fuel was boiled down to a small heel in the dump tanks and this was transferred to the storage tanks.

On March 24, with the high-pressure systems completely filled with D₂O, the pressure was raised to 2500 psia with the purge pumps to test the strength of the system. At the conclusion of the hydrostatic test, the pressurizer heaters were turned on, pressurizing the systems with steam. Circulation was begun in core and blanket at 2245 on March 24. An oxygen flow of 2 liters/min was started into the core at the same time.

Shortly after the start of circulation in the high-pressure systems, the fuel feed pump was switched to pump from the dump tanks. The purpose of this was to mix in any uranium which might have remained in the dump tanks after the transfer, permitting determination of the amount left behind from a high-pressure sample. After over three hours of mixing, a fuel high-pressure sample was isolated at 0249 on March 25. After confirmation that a satisfactory sample had been obtained, at 0547 the fuel feed pump was switched back to condensate supply to purge the uranium from the core into the dump tanks.

Heating of the high-pressure systems had started at 0220 on March 25. The heatup continued, first with plant steam then with steam from the package boiler, until that evening at 1720. At this time, heat addition was stopped in order to replace the soot blower steam valve on the boiler and to stop a steam leak. The system temperature which had reached 230°C fell, in the absence of heating, to 177°C before plant steam was admitted to the heat exchangers at 2220 on March 25. Another attempt to put the package boiler in service was made at 0008 on March 26. This time the feedwater bypass valve failed to function properly and had to be repaired. Satisfactory operation of the boiler was achieved at 0245 and the system temperature was raised to 270°C by 0600 on March 26.

In the period from 0600 on March 25, to 2049 on March 26, five samples, taken from the fuel dump tanks indicated that the solution in the dump tanks still contained about 340 g of uranium. The solute was then concentrated in about 60 lb of solution and was transferred to the fuel storage tanks. After this transfer was completed at 0033 on March 27, three more samples were taken from the fuel dump tanks. These showed that of the original charge of 6397 g of uranium, only 62 g remained in solution in the dump tanks. This small amount was left to be mixed with the new charge. (It later became evident that several hundred grams had been left in the reactor, but did not show in the samples, presumably because it was not soluble.)

Operation with Subcritical Uranium Concentrations

On March 27, a new charge of fuel was injected. Addition of fuel was through line 162 to the fuel dump tanks, beginning at 1315 and ending at 1722. Heavy water was used to flush the lines after the addition. Quantities added in this charge were:

UO_2SO_4	13.64 kg
CuSO_4	3.63 kg
H_2SO_4	1.89 kg
D_2O	60.61 kg

Following the addition of the fuel charge, the dump tank weight was adjusted to 600 lb and at 1852 the feed pump was switched to pump fuel solution. The dump tank weight and the pumping rate (5 lb/min) were chosen so that the core concentration rose only to the critical concentration for 250°C. Because the reactor temperature was being held at 270°C with heating steam, the reactor was well subcritical. This condition was maintained for about 36 hours, until 0700 on March 29. During this period a number of samples were taken to provide a baseline for analysis of chemical behavior during power operation.

POWER OPERATION

Special Supervision and Analysis of Operations

All aspects of the reactor operation were studied closely and continuously during Run 14. Three technical persons were assigned to each shift to plot and to evaluate the data being taken by the regular operating personnel. Also, in addition to the regular shift crew, (1) there was on duty at all times one of three Operations Senior Supervisors. This arrangement was put into effect during the period of subcritical operation and continued until the end of the run.

Some of the operating variables which were followed are plotted in Figs. 3 through 8. (The plots are discussed on pages 15 through 17.) In reading the account which follows, it may be convenient to refer to these figures to get a better understanding of operations.

Initial Operation at 1 Mw

During the subcritical operation, when the system was being heated with steam from the package boiler, condensate levels in the heat exchangers were kept below the upper tubes to expose the necessary heat transfer area. At 0700 on March 29, as the first step in making the reactor critical, the trapping out of condensate was stopped to allow the levels to rise. As the condensate rose, covering more and more tubes, the heat input became less than the heat losses and the reactor temperature began a slow decline.

The slow approach to the critical temperature was halted at 0835 to permit replacement of a faulty process radiation monitor (RE-560A on steam and cooling water lines). While the fuel temperature was being held at 259°C, a fuel high-pressure sample was isolated. The cool-down was resumed at 0955 and criticality was achieved at 1020 at a core temperature of 255°C. The neutron level rose slowly on about a 30-sec period, then leveled off. At this point external heating was stopped by shutting off the package boiler. The reactor power then rose to match system heat losses (about 200 kw).

At this low power, the blanket temperature began to fall below the core temperature, because the heat generated in the blanket was insufficient to match heat losses from that part of the system. The fuel and blanket heat exchangers were interconnected so that steam was free to pass from the fuel to the blanket exchanger, thus transferring heat from the core, where most of the power was being generated. In order to minimize the temperature difference between the blanket and the fuel solution, all of the condensate was drained from the blanket exchanger to give more condensing surface.

(1) The shift organization included 3 engineers and 4 technicians.

At 1112, addition of feedwater to the fuel heat exchanger was started in preparation for raising the power by withdrawing steam. At 1138 the power was raised to about 1 Mw by sending steam to the air-cooled condenser. As the condenser warmed up, it became necessary to start the condenser fan to sustain the heat removal at the 1-Mw level.

During the period in which the fuel temperature was being reduced to approach criticality, the fuel dump tank weight was allowed to decrease to make up for the additional solution required in the high-pressure system. After a stable power level was reached, the fuel dump tank weight was raised back to 600 lb by transferring D₂O vapor to the fuel system from the blanket system through a hole that existed in the rupture-disc assembly. This, along with an increase in the blanket temperature, caused the average core temperature to drop from 255°C to 250°C.

After a short period of operation at 1 Mw, the increasing blanket temperature made it apparent that the heat generation in the blanket was now greater than the heat losses there. Since the blanket heat exchanger had been drained, it was not possible to withdraw steam to remove the excess heat. Instead, the small cooling requirement was met by starting the blanket feed pump to remove heat via the letdown stream. In this way the blanket temperature was stabilized.

An effort was then made to fill the blanket heat exchanger by collecting steam condensate. However, it was found that the heat exchanger blowdown valves leaked and it was necessary to close both valves HV-378A⁽²⁾ and HCV-540 to get any increase in level. HV-378A was closed at 1605 on March 29, and by 2115 the heat exchanger level had risen to 8% on the level recorder. At that time HCV-540 was opened to trap condensate from the blanket steam drum to the deserator surge tank. This was accompanied by an increase in power level, indicating that either the steam trap or HV-378B was leaking. The heat exchanger level also began to decrease slowly. At 2317 the fuel and blanket heat exchangers were interconnected through the shell sampling lines to let hot water flow into the blanket heat exchanger. This brought about a gradual increase in the blanket level. The level was allowed to increase in this way until 0555 on March 20, when it had reached 26%. At that time HCV-538 was closed to reduce the blanket steam drum pressure in an effort to increase the water flow rate into the shell. The level continued to increase at about the same rate as previously and reached 45% at 1115. HCV-538 was then reopened and the level rose to 57%. At that time the feedwater valve was opened and the level raised to 70% with the feedwater pump. A stable operating level was reached at 1400 on March 30.

⁽²⁾For locations of Valves and Lines, see:

F. C. Zapp, "ERT Process Flowsheets, Rev. IV," ORNL CP-55-5-156 (Sept. 16, 1957).

First Rise to 2 Mw and Return

The reactivity available to the reactor system appeared to remain essentially constant throughout 28 hours of operation at 1 Mw. Therefore, the power was slowly raised, beginning at 1545 on March 30, and leveled out at 2 Mw at 1650.

The nuclear average temperature of the reactor was about 255°C at the start of the power increase. This began to drop at about 1600, even before the 2-Mw power level was reached, indicating a loss of reactivity and suggesting a loss of uranium. The decline in average temperature continued for about 3 hours and then appeared to level out at about 244°C, a drop of 10°C. A fuel high-pressure sample was isolated at 1753, after about 6°C drop in the nuclear average temperature, in an attempt to verify the apparent loss of uranium from the high-pressure system. At 1954, the power level was reduced to 1 Mw and the nuclear average temperature began to increase almost immediately. After 2 hours of operation at 1 Mw, this temperature had risen to within 1°C of the value before the increase in power.

Investigation of Temperature Effects at 1 Mw

After the reactor nuclear temperature had recovered from the 2-Mw operation, a series of tests was started to try to determine the effect of core temperature on the stability of the fuel solution at the 1-Mw power level. On March 31, about 132 lb of D₂O was transferred from the blanket dump tanks to the fuel dump tanks to reduce the uranium concentration and lower the reactor temperature by about 10°C. The transfer was completed at 0255. No decrease in temperature was noted beyond that expected from the dilution effect. At 0700, collection of condensate was started to restore the fuel dump tank weight to its original value to see if the initial temperature of 255°C could be reproduced. The resultant temperature was, if anything, very slightly higher when steady state was attained.

Starting at 1146, the fuel solution was concentrated further to raise the core outlet temperature to 270°C. A stable operating condition at this temperature was finally achieved at about 1500 and held for approximately one hour, with no apparent loss of reactivity.

At 1218, while the change in system temperature was being made, the cooling water flow to the fuel letdown heat exchanger was reduced to raise the letdown stream temperature. Up to this point, the temperature had been 30-40°C and a decision was made to operate with the temperature at 80-100°C to reduce the likelihood of precipitating uranyl peroxide in the letdown stream. It was believed that a holdup of precipitated peroxide in the dump tanks might account for some of the apparent inventory losses during conditions of instability. (It was intended that future experiments would be conducted to determine the lowest temperature which could be tolerated without losses by precipitation. However, due to the abrupt termination of the run, these tests were not carried out.)

Second Rise to 2 Mw and Return

Following the establishment of apparent fuel solution stability with the core outlet temperature at 270°C, the fuel dump tank weight was again increased to reduce the reactor temperature to about 255°C. The dump tank weight change was made at 1610 on March 31, and by 1800, the core average temperature had leveled out at 258°C and the nuclear average at 256°C. A fuel high-pressure sample was isolated at 1804 in preparation for increasing the power to 2 Mw to see if the previously observed temperature instability at this power level could be reproduced.

Since there had been some question about the convergence and divergence of the core inlet and outlet temperatures with power changes, it was planned to make this change by first reducing the power and then rapidly increasing it to 2 Mw. Therefore, at 1816, the main reactor steam stop valves, HCV-537 and -538, were closed for 10 minutes. In this period the reactor power level dropped to about 100 kw and the core inlet and outlet temperatures changed by approximately equal amounts, converging to 258°C. At 1826, the main stop valves were reopened with the secondary steam throttling valves, HCV-536A and B closed. The steam drum pressure dropped very sharply as the steam from the drums expanded to fill the outside piping. The fuel circulating pump was stopped by the steam pressure-time derivative interlock and was restarted within 5 seconds. The power level rose rapidly as the outer steam throttling valves were opened to a predetermined setting which would withdraw steam corresponding to 2 Mw of power. There was a brief pause in the power increase caused by a momentary upset in the fuel pressurizer level which was recovered in less than a minute. The reactor was stable at 2 Mw within 5 minutes of the time the power increase was started. (The temperature behavior is described in detail on page 67.)

The reactor nuclear temperature remained steady for nearly one hour after the power increase and then started to decline rather rapidly. This temperature had dropped 7°C by 2100, 1-1/2 hours after the start of the decline. At 2105 the power level was reduced to 1.5 Mw which checked the temperature decline and effected an increase of about 2°C. The power was reduced to 1.3 Mw at 2316 and 1.0 Mw at 2345. A slow but steady increase in temperature ensued and, by 0230 on April 1, the original nuclear average temperature of 255°C and the core average of 258°C were attained.

Rise to 2 Mw and Operation at this Power

After the recovery of system temperature, an attempt was made to determine the exact power level at which reactivity loss, indicated by decrease in nuclear average temperature, would begin to appear. The experiment was started with core average temperature at 258°C. In the period from 0303 to 0325 on April 1, the reactor power level was raised from 1.0 to 1.3 Mw with the nuclear average temperature at 255.5°C. At this power, the nuclear average temperature decreased about 1°C in 1-1/2 hours. Therefore, at 0504, the power level was reduced to 1.2 Mw and the nuclear average temperature appeared to stabilize at 253.5°C.

At 0738, after a short period of operation in "shutdown" to measure the fuel feed and purge rates, the fuel dump tank solution was concentrated to raise the core average temperature to approximately 265°C. The power level was reduced to 1.1 Mw at this time and to 1.0 Mw at 0900. The dump tank weight change was completed at 0855, and at 1130, although the nuclear average temperature was at 265.5°C and still rising slightly, the power level was increased to 1.3 Mw. This temperature rose to 266°C and then dropped back to 265.5°C at 1300. At that time the power was increased to 1.5 Mw and the nuclear average temperature dropped about 0.5°C. At 1410, the power was raised to 1.7 Mw and, in the next 2 hours, the nuclear average temperature dropped 2°C to 263°C and leveled out.

At 1751 on April 1, the reactor was switched to "shutdown" to measure the fuel feed pump flow from the condensate tank. This dilution of the high-pressure system caused a drop of nearly 4°C in the nuclear average temperature which began to come back slowly. At 2015, the power level was increased to 2.0 Mw. After the system had stabilized from the pump flow measurement and the power change, the nuclear average temperature was 261°C, approximately 2°C below the equilibrium temperature for the 1.7-Mw power level. The average core temperature was 261.5°C at this time, compared with 265.5°C at 1.7 Mw.

The reactor was held at 2 Mw with the system temperature remaining essentially constant until 1035 on April 2. Starting at that time, the fuel dump tank solution was gradually diluted with 70 lb of condensate to reduce the core average temperature below 255°C. The object of this move was to reproduce the temperature and power at which instability had appeared on March 31. The system reached equilibrium at about 1300 with the nuclear average temperature at 254°C and the core average temperature at 251°C. The observed decrease of 7-1/2°C in nuclear temperature was approximately what would be expected for the dilution that was made. Since the temperature did not continue to decrease beyond this, it was tentatively concluded that no instability had been encountered. At 1413, after one hour of steady operation at the lower temperature, the dump tank weight was reduced to bring the core average and blanket temperatures up to 265°C. The fuel feed pump rate was also increased from 5 lb/min to about 8 lb/min to help bring up the temperature.

Power Increase to 3.5 Mw

After the nuclear average temperature reached 265°C, the reactor power was increased, in steps, from 2 Mw to 3.5 Mw. The first increase, from 2.0 to 2.25 Mw was made at 1745 on April 2. It was then raised to 2.5 Mw at 1907, to 2.75 Mw at 2005 and to 3 Mw at 2122, all with no apparent change in the nuclear average temperature.

In the period from 2150 to 2230, the high-pressure system pressure was increased from 1500 psig to 1700 psig. At 2220, while the pressure was still increasing, the reactor power level was raised from 3 to 3.25 Mw. A slight decrease in the nuclear average temperature became evident. This was, at

first, attributed to the power increase, so the power level was cut back to 2.5 Mw. It was then recognized that the increase in pressure had caused a reduction in fuel feed rate which in turn resulted in lower core concentration and lower critical temperature. Action was then taken to restore the fuel feed rate to 8 lb/min, the value that existed at 1500 psig system pressure. After several adjustments in the fuel feed and purge rates, the nuclear average temperature reached 267°C at 0400 on April 3. It was then reduced to 266°C at 0530 by dilution. In the period from 0555 to 0610, the reactor power level was increased from 2.5 to 3.0 Mw. The nuclear average temperature remained constant, so the power was increased to 3.3 Mw at 0710 and to 3.5 Mw at 0818.

Operation at 3.5 Mw

The reactor system operated stably at 3.5 Mw until 1430 on April 3. At that time an addition of sulfuric acid was made to the fuel dump tanks. The following materials and amounts were added:

H ₂ SO ₄	257 g
H ₂ O	198 g
D ₂ O	10,294 g

Immediately following the acid addition, the nuclear average temperature, which had been steady for about 9 hours, began to rise. In 3 hours it rose 3°C, then leveled out.

At 1830 the system pressure was increased from 1700 psig to 1800 psig. This necessitated an adjustment in the fuel feed rate which had dropped from 8 lb/min to 6.25 lb/min, causing a slight decrease in system temperature.

At 2118 an attempt was made to send reactor steam to the turbine which, up to this time, had been operating with plant steam. The controller which regulates the reactor steam pressure to the turbine throttle failed to function properly, causing several reactor power surges and erratic behavior of the turbine. This resulted in a complete loss of AC power in the building. At this point all steam withdrawal from the reactor was stopped and the feed pump was switched to "shutdown," reducing the power level to below 1 Mw. Building power was then restored from the outside service line and all auxiliary AC equipment was put back in service. After the reactor system was returned to stable operation, the power was raised, reaching 2 Mw at 2245 and 3.5 Mw at 2330.

Since it was planned to raise the reactor to the full 5-Mw power level on April 4, it was decided to raise the nuclear average temperature to 275°C for better recombination-catalyst activity. Accordingly, the fuel dump tank solution was concentrated, starting at midnight, to achieve this temperature.

Starting at 0110 on April 4, the reactor power level was reduced to 3 Mw in preparation for another attempt to operate the turbine with reactor steam. At 0218, the changeover from building steam was started by manually controlling the steam throttling valves. A smooth transition resulted from this procedure. With the turbine on reactor steam, all other steam withdrawal was stopped and it was observed that an indicated reactor power level of 3.2 Mw was required to produce 200 Kw of electricity at a 0.86 power factor. The total reactor power level was restored to 3.5 Mw at 0338 with most of the steam going to the turbine and the remainder to the air-cooled condenser. At 0530, another attempt was made to control, automatically, the reactor steam pressure to the turbine throttle. However, accurate control could not be accomplished and wide fluctuations in power level occurred. Therefore, the attempt was abandoned and manual control was retained.

Stepwise Increase to Maximum Power

Starting at 0900 on April 4, the reactor level was raised in three steps from 3.5 to 5.0 Mw. The power reached 4.0 Mw at 0911, 4.5 Mw at 1040, and 5.0 Mw at 1210.⁽³⁾ In contrast to the equal divergence observed earlier, the core outlet temperature did not change as the power was raised, while the inlet quickly dropped about 5°C at each step increase.

Core Tank Failure and Subsequent Shutdown

The reactor was running very smoothly when at 1232 the indicated fuel pressurizer level abruptly dropped offscale. An interlock on the pressurizer level switched the fuel feed pump supply from the dump tanks to the condensate tank, thus beginning a dilution of the core. When the reactor operator saw the level drop, his first thought was that the letdown valve was stuck open and he immediately increased the feed pump stroke in an effort to halt the drop and bring the level back on scale. Although the pressure and the fuel circulating pump power did not behave as if the level had fallen all the way to the connection with the circulating stream, the level remained out of sight. A hasty check of the fuel feed pump showed that it was indeed delivering a high flow. It was then noticed that the blanket dump tank weight was increasing rapidly and that temperatures indicated that fuel letdown had stopped and that letdown through the blanket letdown line had started. From these indications it was surmised that the level instrument was indicating falsely and that the fuel pressurizer was actually full and overflowing to the blanket through the open rupture-disc line.⁽⁴⁾

⁽³⁾ These are all nominal powers, indicated by a fuel heat exchanger AT recorder. The best estimate of the maximum power is 6.4 Mw.

⁽⁴⁾ Pressurizer temperatures did not drop, as they would have, had the conjectured situation actually existed. This fact was not noted immediately, however.

Proceeding on the assumption that the level element had failed, the operators put the fuel letdown valve on manual control and opened it to reduce the supposed high level in the pressurizer. When the fuel letdown valve was opened, however, the blanket pressurizer level began to fall and, even though this caused the blanket letdown valve to close, it became necessary to operate the blanket feed pump to maintain level in the blanket pressurizer. At about this time it was noticed that, coincidental with the loss of fuel pressurizer level, the blanket inlet and outlet temperatures had begun to diverge, indicating an increasing power generation in that region and suggesting that uranium was entering the blanket (see Fig. 9). Thus it became apparent that a connection which permitted interchange of liquid must have suddenly opened between the fuel and blanket systems.

The reactor power level had been reduced after the efforts to bring the fuel pressurizer level on scale proved unsuccessful. The turbine was shifted from reactor steam to plant steam and the reactor power was reduced further by stopping the steam flow to the air-cooled condenser. With steam withdrawal stopped, dilution lowered the critical temperature faster than heat losses reduced the actual temperature, so the reactor went subcritical. At 1317 the package boiler was put in service to retard the rate of cooling and insure that the reactor would not go critical again.

The presence of uranium in the blanket was confirmed by a sample taken at 1406. This fact was fairly obvious immediately upon withdrawing the sample since the activity of the sample was very high, and chemical analysis soon proved that it contained uranium.

Starting at 1500, a final check was made to clearly demonstrate the existence of an interconnection between the fuel and blanket liquid systems. Both letdown valves were closed manually, both circulating pumps were stopped and both sets of pressurizer heaters were turned off. It was then possible to bring the fuel pressurizer level into the range of its level elements without raising the blanket level out of the range of that element.⁽⁵⁾ The interconnecting hole was apparently fairly large, as the fuel and blanket levels stayed close together when both were raised with either the fuel or blanket feed pump.

The shutdown was carried to completion by continuing to flush the high-pressure systems with condensate for 12 hours, cooling down and draining to the dump tanks. Measurements were then conducted to determine the site and

⁽⁵⁾Earlier attempts to do this had been unsuccessful because of the difference in pressure drops that existed between the core and blanket and their respective pressurizers. With circulation there were pressure drops along the lines of liquid flow in fuel and blanket regions from the hole to the pressurizer connections. Also, the pressurizer heaters were operating in such a way to cause an additional differential pressure between the two pressurizers.

magnitude of the leak in the core tank. (6) On the basis of previously established calibration of liquid level vs. system holdup, the break appeared to be in the vicinity of the welded peripheral joint between the 30- and 90-deg cone entrance sections near the bottom of the Zircaloy core vessel (see Fig. 1). While pumping from the core to the blanket at 25 lb/min, the water levels in the core and blanket were found to differ by only $3/8$ in., with pneumatic level probes being used for the measurements. A head loss of approximately $3/4$ in. was measured at a flow rate of 65 lb/min. On the basis of these measurements, the hole area was estimated to be 1.5 to 2.0 in.².

Over the next few months, several attempts were made to view the opening in the core tank. However, it was not until September 24, that the photograph shown in Fig. 2 was made with the use of a remotely-manipulated mirror and periscope system. In this photograph the hole appears to be nearly round and between 1 and 2 in. in diameter. Its location was at the intersection of the 30- and 90-deg cone as had been predicted by the early measurements.

(6)R. Van Winkle and R. H. Guyson, "A Review of the Data and Observations Taken for the Purpose of Locating and Estimating the Size of the Hole in the Core," ORNL CF-58-6-112 (June 13, 1958).

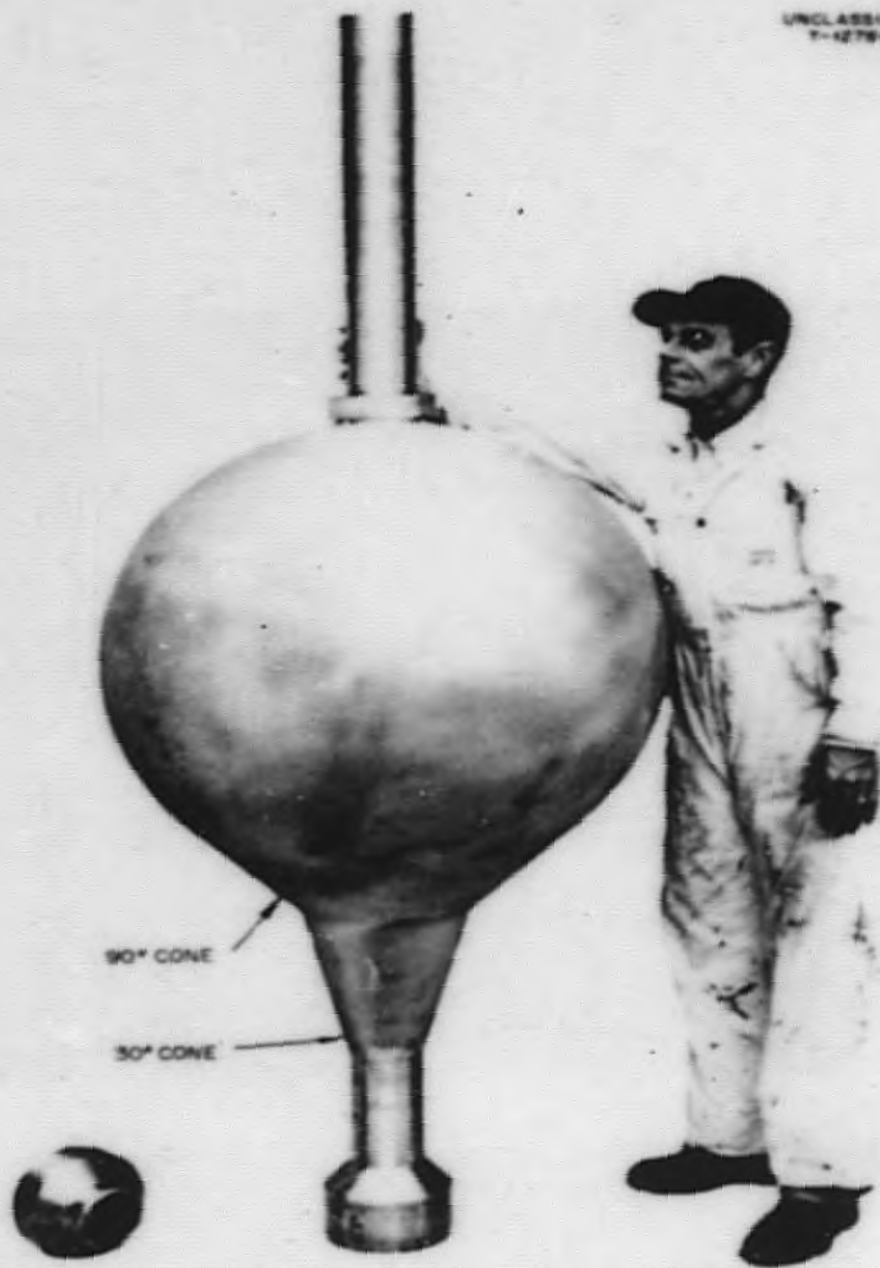


Fig. 1

COMPLETED CORE TANK SHOWING JOINT BETWEEN THE 30- and 90-deg
CONE ENTRANCE SECTIONS.

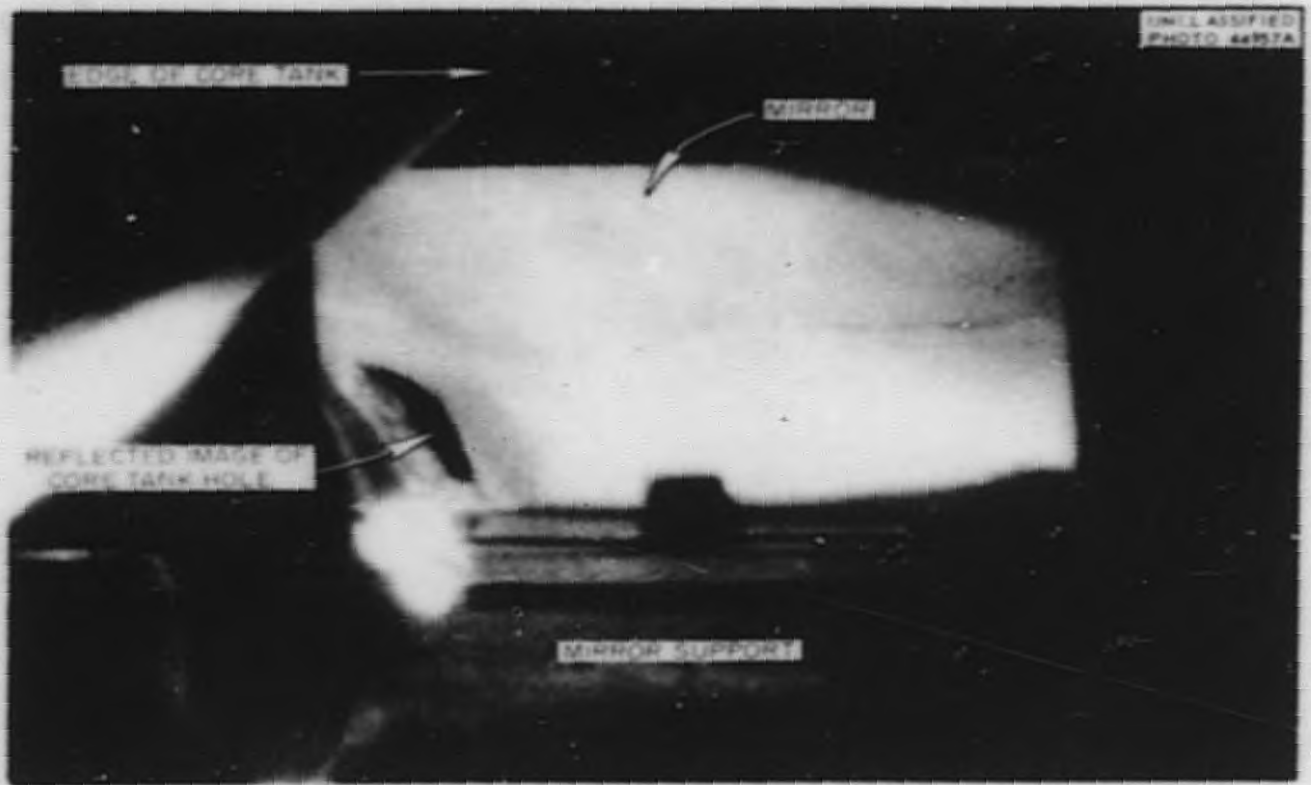


Fig. 2. Photograph of Hole in HRT Core Tank as seen in Viewing Mirror.
(View from Blanket Side)

GRAPHS OF OPERATING CONDITIONS

Continuous plots of many of the reactor operating variables were maintained throughout Run 14. A number of these curves are presented here for the period of power operation. The curves were reproduced directly from the original plots and the complete representation for the seven days of power operation covers two sheets for each curve.

Reactor Power

The reactor power indicated by the line on Figs. 3 and 4 was obtained directly from the chart record of TdR-6530. This instrument records the difference in E.M.F. of thermocouples on the inlet and exit connections to the fuel heat exchanger, adjusted by a constant factor which takes into account the solution flow rate and its heat capacity, to indicate core power directly in Mw. Thus variations in flow rate or heat capacity would introduce error in this measurement. Error may also exist if the scale factor of the instrument is incorrect. In any event, the instrument indicates only the core power and does not take into account power generated elsewhere in the reactor system.

The total reactor powers measured by overall system heat balances are shown as points on the graph. Only a small part of the differences between indicated core power from TdR-6530 and the heat balance power can be attributed to power generation outside the core. It appears that the nuclear power was actually as much as 30% greater than that indicated by TdR-6530 (see discussion on page 36).

Core Pressure

Core pressure, also shown in Figs. 3 and 4, is among the steadiest of the reactor system parameters and is set primarily by the requirement that boiling in the high-pressure system be avoided. To provide a wide margin for error, the pressure was raised from 1500 psig to 1700 and later to 1800 psig as the system temperatures were raised during the run. Raising the pressure lowers the output of the feed and purge pumps, thus affecting the core concentration. Therefore, adjustments in the pumping rates had to be made with each pressure change to minimize undesired changes in other parameters.

Core and Blanket Temperatures

The core inlet and outlet and the blanket outlet temperatures are plotted together in Figs. 3 and 4 because of their close interrelationship. The core temperature on this plot was taken from one recorder (TR-6560) while the blanket temperature was taken from another (TR-6510). The core temperature was also recorded on the multipoint instrument with the blanket temperature, but the TR-6560 record was plotted because it was more convenient and because it had been adopted as the reference instrument by which the system was controlled.

Factors Affecting Concentration

The factors which can be varied to control the concentration of fuel in the core are the fuel dump tank weight and the rates of fuel feed and purge. These variables are plotted in Figs. 5 and 6.

Changes in the fuel dump tank weight may be effected by diverting some condensate to the condensate tank or by dropping some of the condensate from that tank. Such changes are the normal means of controlling core concentration. If the condensate weight is held steady while the core temperature changes for any reason, the change in high-pressure system density will show up as a change in dump tank weight. Perturbations in dump tank weight result when pumping rates are measured by pumping from the condensate tank. In such operations, the condensate weight is allowed to decrease while the dump tank weight goes up. Generally the dump tank weight is restored to normal soon after pumping rate measurements are made.

The points shown on the plots of feed and purge rates are rates measured by pumping from the condensate tank.

"Nuclear Average" Temperature

The critical concentration of U-235 in the core is a function of the "nuclear average" temperature of the reactor. Figs. 7 and 8 show an approximation of the "nuclear average" temperature equal to one-third the sum of the core inlet, core outlet and blanket outlet temperatures. (The significance of the "nuclear average" and the reason for this particular approximation are discussed on pages 73 through 74.)

Concentration Ratios

The core uranium concentration can be determined in three ways: by a material balance using the book inventory, from the critical temperature and by sample analysis. During the course of Run 14, core concentrations estimated by the three different methods were compared to detect changes in the uranium in circulation and as a check on the analytical results. Figs. 7 and 8 compare concentrations as the ratios of critical and analytical concentrations to the concentration calculated from a material balance. (The points plotted in these figures differ from the points originally calculated because of a revised value of book inventory during Run 14.) For further discussion, see page 72.

Charcoal Temperatures

Since Run 14 was the first operation to generate very large quantities of fission product gases, the performance of the charcoal adsorber beds was followed closely. The temperature in the 1/2-in. section of one of the beds in service is shown in Figs. 7 and 8, with temperature in the idle bed plotted for comparison.

Core Tank Failure

Because of the careful examination that was made of the system conditions at the time of the appearance of the core hole, a special plot was prepared, on an expanded time scale, to show the reactor temperatures and power levels for this period. This information is presented in Fig. 9.

-9T-

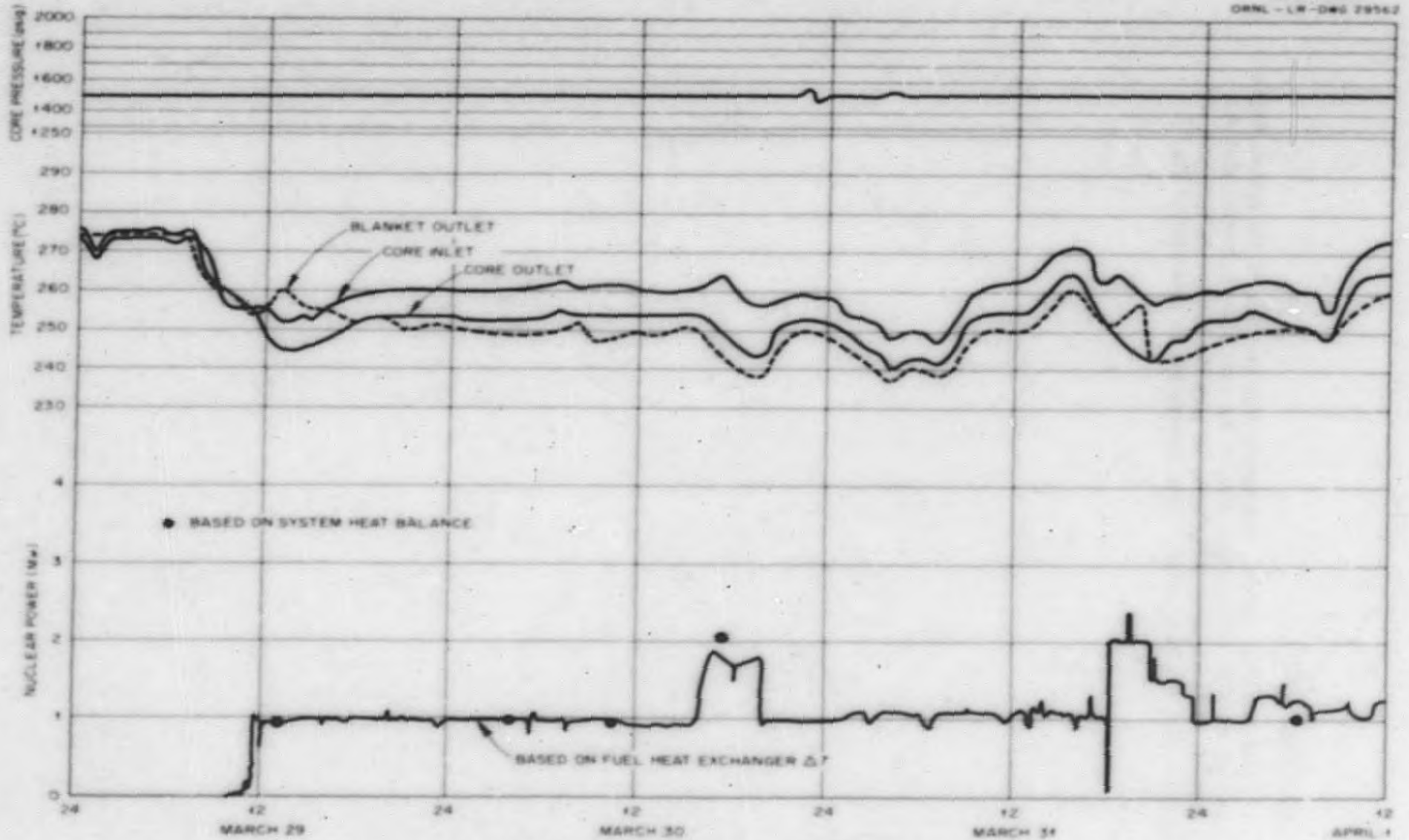


Fig. 3. Operating Parameters for HRT - Run 14.

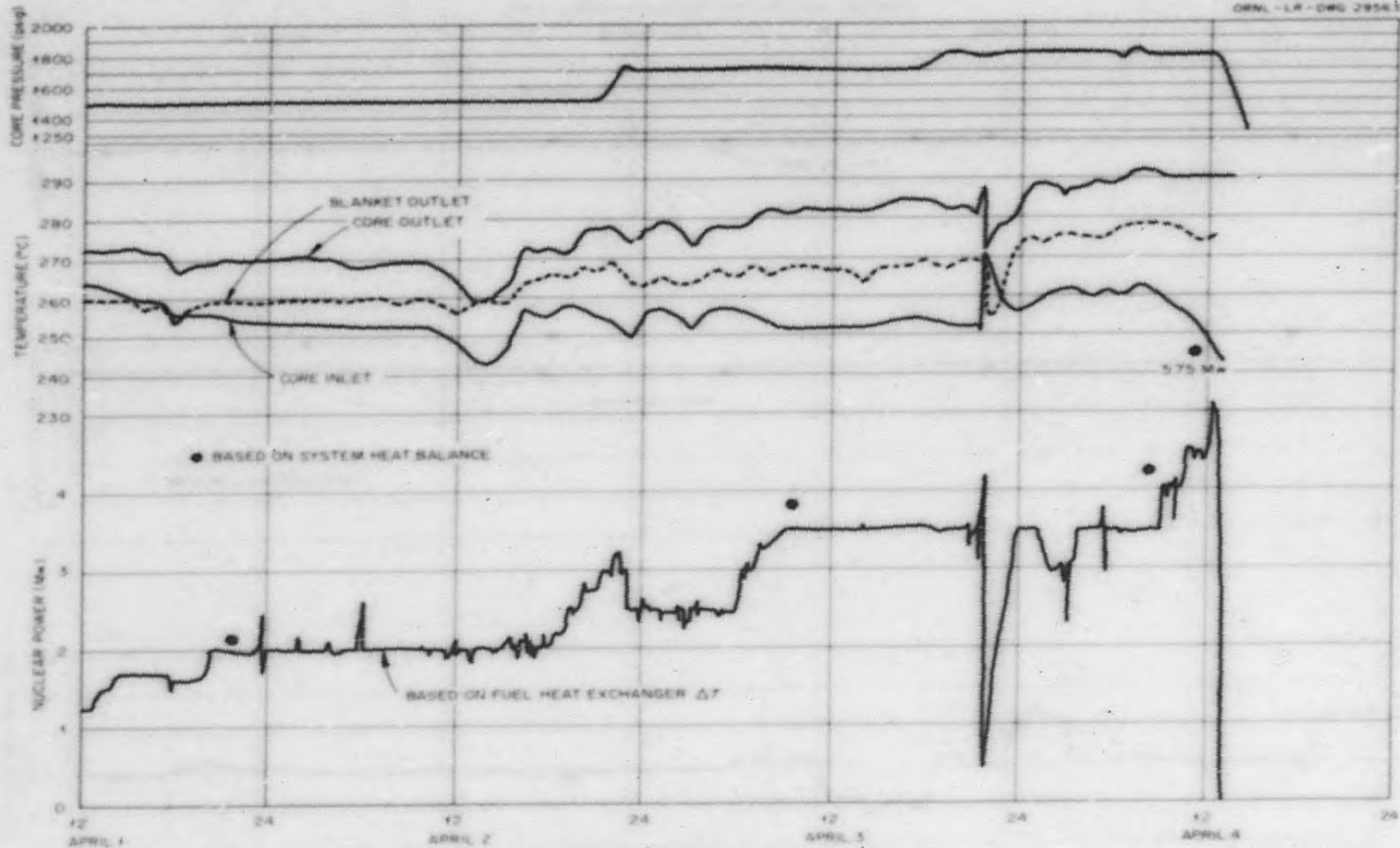


Fig. 4. Operating Parameters for HRT - Run 14

-02-

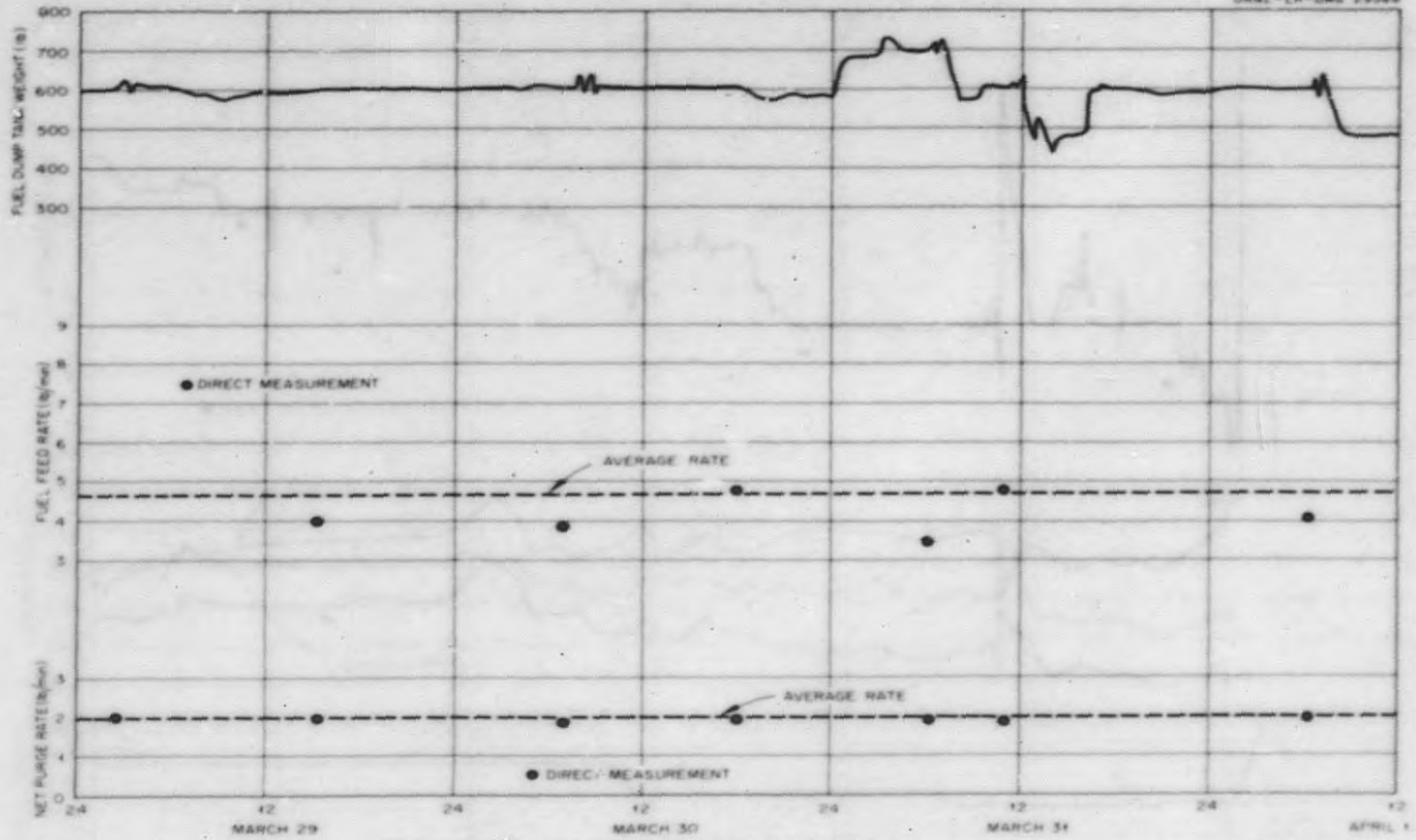


Fig 5. Operating Parameters for HRT - Run 14

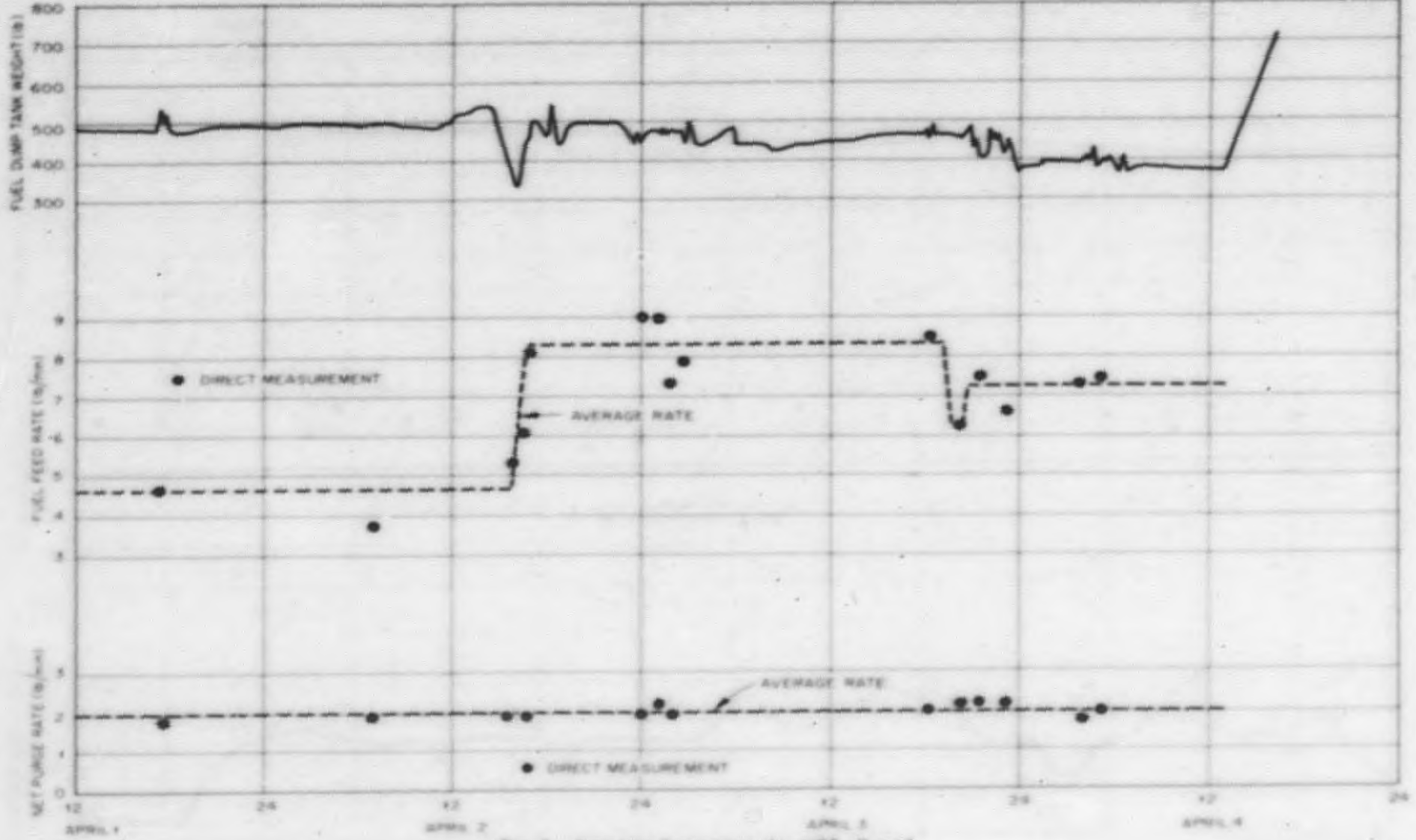
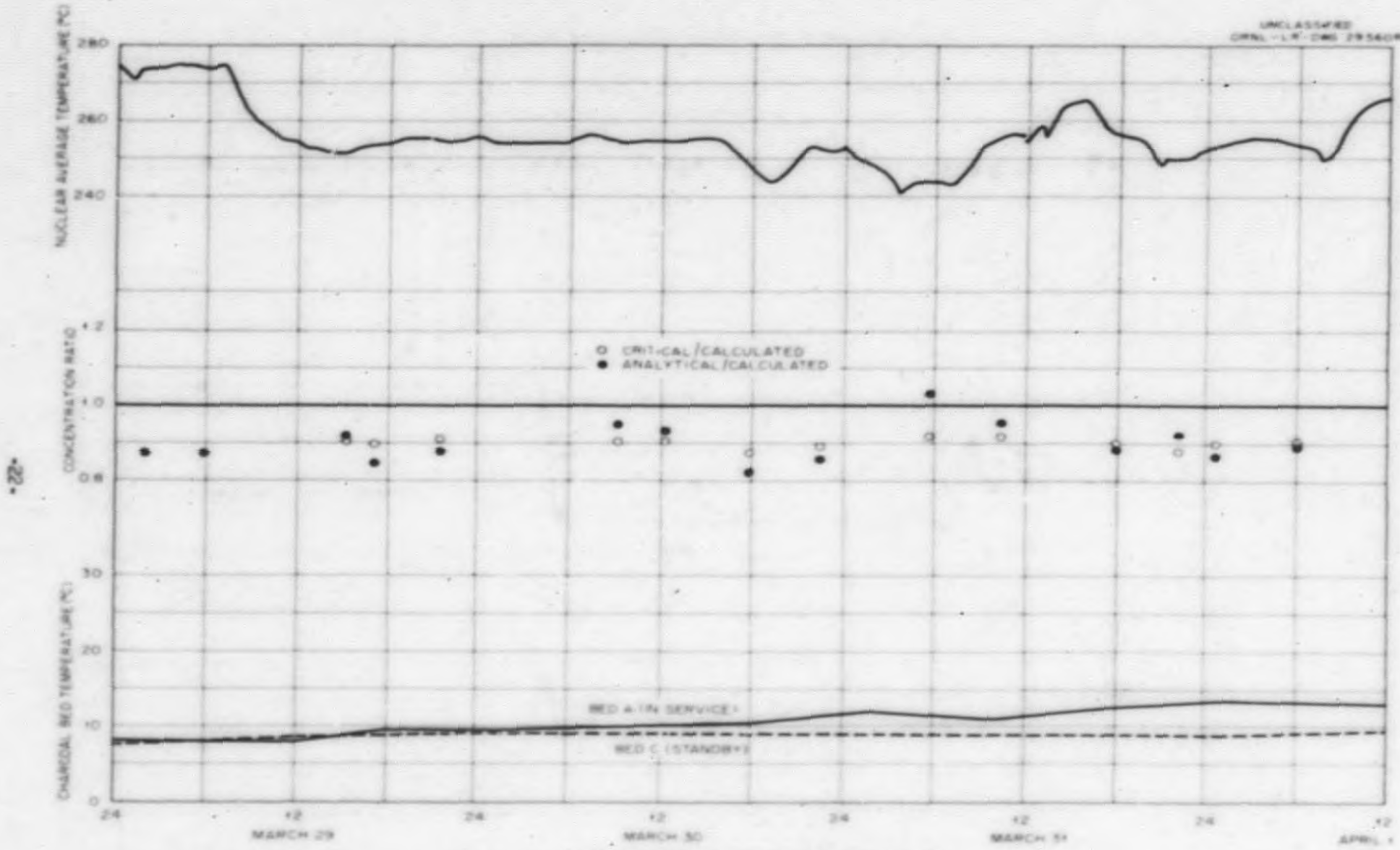


Fig 6. Operating Parameters for HRT - Run 14

-12-



•22•

Fig 7 Operating Parameters for HRT - Run 14

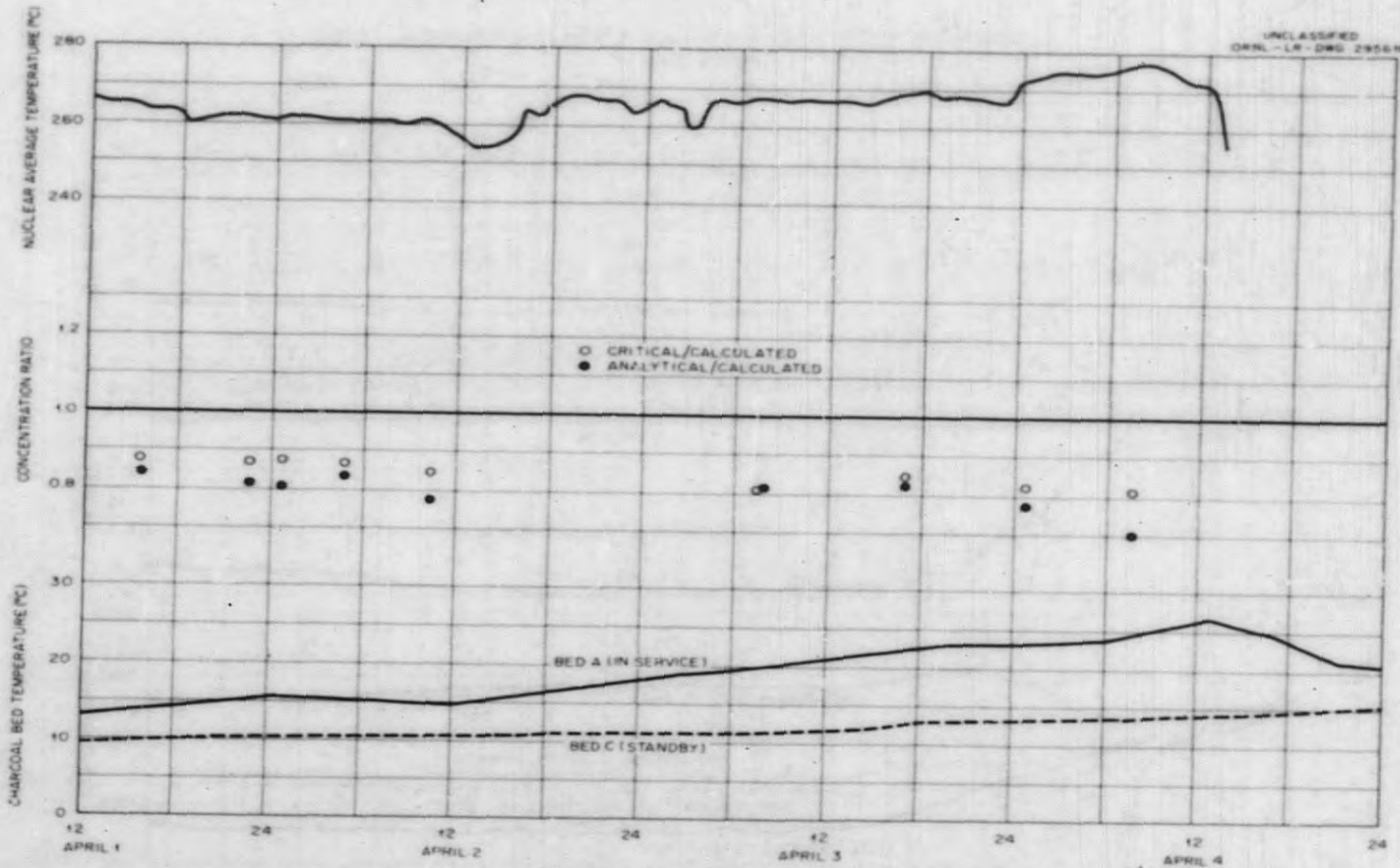


Fig. 8. Operating Parameters for HRT - Run 14

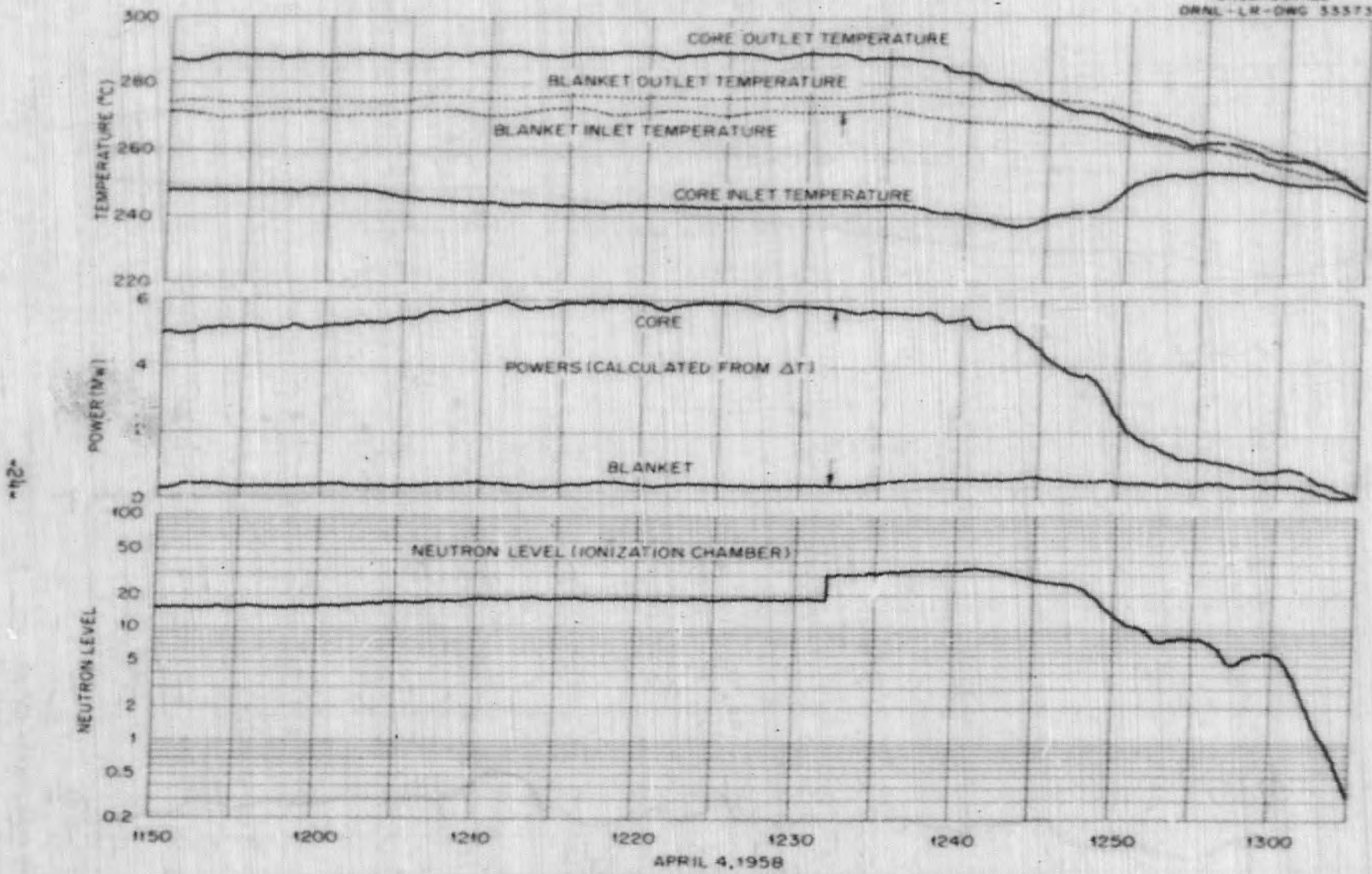


Fig. 9. Temperatures and Powers Around the Time of the Core Leak, April 4, 1958.

AUXILIARY SYSTEMS OPERATIONS

While the foregoing description is confined, for the most part, to the operation of the primary part of the reactor complex, i.e., the fuel and blanket systems, operation of various auxiliary systems is an essential part of the overall operation. The operation and performance of the oxygen and off-gas systems, the steam system and the flange leak-detector system are discussed in the following paragraphs. Other systems which operated as designed, automatically with only routine periodical checks, include:

- demineralized cooling water system
- tower cooling water system
- auxiliary steam system
- refrigeration system
- instrument air system
- DC control power

Oxygen Addition and Off-gas Systems

Oxygen was continuously injected into the fuel high-pressure system at a rate of 2 liters/min (STP), beginning with the first circulation of D_2O on March 24, and continuing throughout the remainder of the run. No oxygen was injected into the D_2O blanket. The oxygen was supplied from special high-pressure cylinders (2400 psig) and was metered into the discharge of the fuel feed pump through a calibrated capillary flowmeter. The consumption of oxygen at the supply cylinder station was followed as a check on the integrated flow measured by the flowmeter. Oxygen flow was controlled by a differential-expansion type valve, consisting of a tantalum plug inside a stainless-steel sleeve. By raising the temperature with electric heaters, the clearance between these two parts was increased to permit the desired gas flow. (1)

After the beginning of nuclear operation in Run 12, there was no longer any means of measuring the concentration of dissolved oxygen in the fuel solution, which is the real measure of the effectiveness of the oxygen injection system. Experiments during preliminary testing of the reactor in 1957, showed that concentrations in solution averaged 70% of the concentration which should have existed if all the injected oxygen had gone into solution. (2) Subsequent to these experiments, the oxygen injection point was modified to give better dispersion of the oxygen in the feed stream. Thus 70% of theoretical is probably a conservatively low figure for the actual dissolved oxygen concentration. On the basis of 70% dissolution efficiency and a 2 liters/min injection rate, the oxygen concentration in the high-

(1) D. S. Toomb et al., "HRP Quar. Prog. Rep. Jan. 31, 1958," ORNL 2493, p 20.

(2) R. Van Winkle and R. R. Wiethaup, "Oxygen Concentration in Fuel and Blanket High-Pressure Systems, HRT Run 7," ORNL CF-57-6-56 (June 12, 1957).

pressure fuel solution should have been 670 ppm at the start of the run and 475 ppm at the conclusion of the run when the liquid letdown rate was higher.

Outwardly, the injection system operated satisfactorily.

The oxygen and fission-product gases from the high-pressure system, along with possible traces of D_2 and O_2 not recombined in the low-pressure system recombiners and any D_2O vapor that may have escaped from the off-gas cold traps, was passed to the reactor charcoal beds. The charcoal bed discharge was passed through a dry gas meter and a wet test meter before being released to the stack. Essentially all of the reactor off-gas followed this path. However, very small amounts were released directly to the stack via the storage cold trap by venting the transfer tanks during sampling operations.

Throughout the period of operation a running balance was maintained between the oxygen additions and the off-gas flow to give some indication of oxygen consumption in the reactor system by corrosion and/or other mechanisms. In spite of efforts to obtain an accurate balance, the calculated net oxygen consumption for the entire run was slightly negative (106 liters STP in a total of 30,000 liters throughput). At the nominal throughput rate of 2 liters/min, a corrosion rate of 8 mpy in the reactor could be completely masked by the estimated 2% error in present oxygen balance technique.

Shortly after the start of power operation on March 29, the temperature in the two in-service charcoal beds were observed to start rising, indicating the presence of adsorbed fission products. These temperatures started at 8°C and rose steadily to a maximum of 26°C at the time of shutdown (see Figs. 7 and 8). On April 1, a rotameter was installed in the line which supplies cooling and flooding water to the charcoal beds, so that heat removal from the beds could be calculated. Later, on April 3, a temperature controller was installed to provide constant temperature water for this purpose. However, the forced shutdown on April 4 prevented the accumulation of such data.

Steam System

During the subcritical operation at the beginning of Run 14, while the heat exchangers were condensing steam from outside sources, they were blown down manually at short intervals to maintain a level of 55 to 60% in the fuel heat exchanger and 45 to 50% in the blanket unit. These levels were set by the amount of condensing surface required to maintain the system temperatures. The blowdown was sent directly to the 7500 area waste pond. During power operation, the blowdown schedule was set to control the accumulation of solids in the heat exchangers. Although the blowdown rate varied with power level and chemical addition rates, a typical schedule was 1-1/2% of heat exchanger level every 2 hours.

During the early part of the power run, each heat exchanger blowdown was followed by a surge in reactor power level as cool feedwater was admitted to the heat exchanger to make up for the blowdown loss. Later these power surges

were reduced by first raising the heat exchanger level slowly and then lowering the level set point as the blowdown was made. However, power surges still occurred on occasions when the pneumatically-operated blowdown valves failed to close rapidly enough to hold the heat exchanger level at the desired point. Leakage through the blowdown valves also resulted in blowdowns that exceeded the desired amount. At 0956 on April 4, the fuel heat exchanger level dropped 6% below the normal operating point before the blowdown could be stopped. On this occasion, the level controller set point was lowered and then brought up slowly, partly to avoid thermally shocking the heat exchanger with a large addition of cool feedwater and partly to prevent an excessive surge in reactor power level.

All of the heat exchanger samples were monitored for activity at the time they were taken. At no time was any activity detected in the steam samples. The shell samples, particularly from the fuel heat exchanger, began to show small amounts of activity in the latter part of the run. The activity levels were so low that identification of the radioactive species would have been highly questionable and no such analyses were attempted.

From the beginning of the power run until 0915 on March 31, the fuel heat exchanger feedwater level was held at 79% on the level recorder. This corresponded to a water level several inches higher than that at which the heat exchanger was designed to operate. Therefore, the level was lowered to 70% to reduce the possibility of feedwater entrainment in the steam. The activity in the feedwater was too low to permit evaluation of the entrainment from steam activity and solids analyses of steam samples were very erratic although some did indicate the presence of solids.

The air-cooled condenser was in service throughout the entire power run to condense any steam that was not sent to the turbine. With the fan off, the condenser could remove less than 1 Mw of heat. Therefore it was necessary to operate the fan. However, the heat removal capacity with the fan in service was so great that the condensate at the discharge of the condenser was subcooled nearly to ambient temperature. The result of this was that the thermostatic traps on the condenser gas vents remained open, permitting streams of cold condensate to pour out continuously. Therefore, it was necessary to keep the hand valves in the trap discharges closed, opening them briefly on a two-hour schedule to release any accumulation of gas. Significant quantities of gas did collect at the high steam rates associated with the higher power levels where the condensate discharge was warmer. The gas was rich in ammonia from decomposition of the hydrazine in the feedwater.

Flange Leak-Detector System

The flange leak-detector system performed very satisfactorily during Run 14. Under normal operation, all of the high-pressure headers were interconnected through the common fill and vent lines. They were, however, isolated from the pressurizing header which contains the interface between the

D₂O and the pressurizing gas. The low-pressure headers were similarly connected. The individual header pressures were read every 2 hours and, once each shift, the headers were opened to the pressurizing header and the liquid level was recorded. Under stable operating conditions the loss of D₂O was extremely low, about 25 cc in a 24-hour period.⁽¹⁾

(1) J. A. Watts, "Operating Experience with the HPT Flange Leak-Detector System, 12-6-57 through 2-11-59," ORNL CF-59-5-116 (May 22, 1959).

COMPONENT PERFORMANCE

Samplers

During the period of operation from March 25, 1958, to April 4, 1958, 64 samples were isolated from the reactor systems, 41 of which were taken from the fuel high-pressure system. Table 1 shows the results of these operations in terms of the volume of sample obtained.

Three fuel high-pressure sampling operations initially yielded less than 2 ml volumes. In each case, a second sample flask was evacuated and placed under the isolation chamber to withdraw the sample. In two of the three cases a successful sample was obtained by this procedure. Two blanket high-pressure samples yielded less than 1 ml. In each of these cases fluid was recirculated through the sampler to isolate a fresh sample. In the remaining cases where a small sample was obtained, as many analyses were made as the volume permitted.

The major operational difficulties were encountered with the fuel high-pressure sampler. The high-pressure sample stream flow rate is normally controlled by a hand-throttling valve at the sample station. An interlock is provided to close the inlet block valve, HCV-136, if the flow rate becomes high enough to raise the pressure in the isolation chamber above 9 psig. In the case of the fuel sampler, the inlet throttling valve, HV-135, leaked so that the isolation chamber pressure occasionally rose to the trip point, causing a temporary interruption in the sample flow. In addition, the leakage past the seat of the fuel letdown valve increased steadily during the run. This factor, combined with the high sampler flows, caused occasional losses of fuel pressurizer level during sampling.

Besides the difficulty in obtaining full-sized samples, other troubles were encountered in the blanket low-pressure sampler. Mechanical alignment problems prevented positioning the sample flask holder under the isolation chamber with the result that no blanket low-pressure samples were attempted after March 31.

In general, the operation of sampling the fuel dump tanks had a pronounced effect on the operation of the entire system. In order to get flow through the sampler, the dump tanks are pressurized by turning off the cooling water to the recombiner condenser and closing the condenser vent valve, HCV-344. The condensate makeup valve, HCV-336, is also closed to prevent gas-binding the fuel purge pumps. This results in a decrease in condensate tank weight during sampling and a corresponding increase in dump tank weight. The dilution of the dump tank solution is ultimately reflected as a dilution of the fuel high-pressure system and a decrease in the core temperature. The drop in core temperature causes a drop in reactor steam drum pressure and makes it necessary to adjust the reactor steam throttling valves if a constant power level is to be maintained. The D₂O boilup rate is greatly reduced when the recombiner condenser cooling water is turned off. This allows the temperatures of the iodine bed and catalytic

recombiner to rise toward that of the steam in the heating coils. Although the low-pressure system conditions return to normal rather rapidly, two hours or more are required to re-establish equilibrium in the high-pressure system.

TABLE 1
 SUMMARY OF SAMPLING OPERATIONS IN RUN 14
 (March 25 - April 4, 1958)

Sampler	Fuel High-P	Fuel Low-P	Blanket High-P	Blanket Low-P	Total	
Samples Isolated	41	10	9	4	64	
Sample Obtained in Range	0-1 ml	2 ⁽¹⁾	0	2 ⁽²⁾	0	4 ⁽¹⁾
	1-2 ml	1 ⁽¹⁾	0	0	3	4 ⁽¹⁾
	2-3 ml	2	1	1	0	4
	3-4 ml	1	0	0	0	1
	4-10 ml	23	9	5	1	38
	> 10 ml	14	0	1	0	15

(1) After withdrawal of one sample in this range, redrain of same isolation gave a larger sample.

(2) Operation immediately repeated to get > 4 ml.

MISCELLANEOUS

Radiation Measurements

A significant amount of effort in Run 14 was devoted to radiation measurements. These measurements were made to provide information about the reactor operation as well as for personnel protection.

Process Monitors.--Seven strategic points in the reactor system are continuously monitored by Victoreen radiation elements. Two elements are used at each location and the readings are recorded on a 16-point recorder in the control room. Only one pair of instruments indicated any radiation at all during the power run. The two elements of the steam and cooling water monitor in the steam valve pit appear to have been located so that they were exposed to some direct radiation from the reactor cell. This may have been scattered radiation coming into the pit through the lower portion of the fuel sampler cavity, or it may have been radiation streaming along the main steam lines. In either case, the readings of these elements were proportional to the reactor power level, following any changes immediately. Aside from the unexpectedly high readings, these monitors displayed some abnormalities during the power run. At 0400 on April 2, the readings of the two elements began to diverge, with one drifting slowly downscale. After this time, the readings were somewhat erratic and, at 2145 on April 3, the readings from both elements started downscale and finally stopped at the low end of the recorder chart. A few brief periods followed where upscale readings were observed but they were not comparable to earlier readings. Since this type of instrument is extremely temperature sensitive, and breaks down completely above 125°F, the anomalous behavior has been attributed to this effect. These elements were subsequently relocated in the steam valve pit to provide a lower ambient temperature.

Outside Radiation.--Periodic radiation surveys were made of the area outside the main reactor cell. In places where radiation was detected, surveys were made at various reactor power levels to establish a basis for restricting access to these areas. Except for the specific locations discussed below, the radiation level generally was not significantly above background.

Table 2 lists the maximum radiation levels at various conditions for the four areas around the reactor cell where activity was detectable. The listed dose rates include fast and thermal neutrons as well as gamma radiation, with the appropriate factors for relative biological effectiveness.

As may be seen from Table 2, the highest radiation levels existed at the fuel sample station, where readings up to 40 times tolerance were observed under some conditions. As much as 85 to 90% of the total dose at this point was due to fast neutrons. Accordingly, a portable block of borated paraffin was placed over the sampler and left there except during sampling operations. With this block in place, the neutron dose rates

were negligible, although there was some gamma radiation. The shielding section which is an integral part of the sampler contained at this time only lead shot, which provided insufficient neutron shielding. Afterwards, the interstices were filled with borated water, as originally intended, to provide the desired neutron attenuation.

At the other locations listed in Table 2, gamma radiation was the major factor in the total dose. An area of radiation existed at the top of the instrument thimble just outside the main control room. Other small spots existed at one of the reactor cell roof plugs (B-C, which is just above the core), and outside the north shield wall on the third level of the control area.

In addition to the above items, a short burst of activity appeared at the stack filter each time a fuel sample was taken. The sampling procedure required that the fuel transfer tank be vented to atmospheric pressure through the holdup tank and storage cold trap to the inlet side of the stack filter. Small amounts of fission-product gases accumulated in the transfer tank and escaped in the venting operation. Although the filter activity levels were quite high at these times, the periods were very short and no activity was detected by the stack activity monitor. The procedure was later revised to allow several hours' decay in the hold-up tank.

Reactor Cell Radiation.—A number of high-level gamma ionization chambers are located at various positions within the reactor cell to permit measurement of the radiation level during operation. Some of the readings obtained during Run 14 are listed in Table 3 along with the corresponding reactor power level. These readings represent gamma activity only, since the chambers do not respond to neutrons. It may be noted that earlier calculations of the total background radiation level in the vicinity of the fuel circulating pump predicted about 10^5 r/hr at the 5-Mw reactor power level. This compares favorably with the observed value of 5.5×10^4 r/hr at 3.5 Mw.

Reactor Cell Leakage

The main reactor cell was operated at a pressure of about one-half atmosphere absolute, throughout the run. The cell was evacuated on March 26, and the end of the run, 10 days later, came before it was necessary to operate the vacuum pump again. A continuous plot of cell pressure and air temperatures was used to calculate the overall rate of air leakage into the cell. Over most of the run, a rate of about 1 liter/min was observed, considerably below any previous determinations. Toward the end of the run, the apparent rate was even lower.

Diaphragm-Pump Flow Measurements

Provisions are made in the reactor system to measure the pumping rates of all of the feed and purge pumps by a heat balance technique. In the

TABLE 2

MAXIMUM RADIATION LEVELS OUTSIDE THE SHIELD DURING RUN 14

Date	Time	Nominal Reactor Power Level	Fuel Sample Station	Roof Shield Block 8-C	Inst. Thimble	North Wall 3rd Level ⁽⁴⁾
(1958)		(Mw)	(mrem/hr)	(mrem/hr)	(mrem/hr)	(mrem/hr)
Mar. 31	1520	2	168 ⁽¹⁾	--	--	--
Mar. 31	1935	2	97 ⁽³⁾	22	0.7	5
Apr. 1	1430	1.7	2.5 ⁽²⁾	--	--	--
Apr. 2	1935	2.6	6 ⁽²⁾	21	1.2	6
Apr. 2	2045	2.6	286 ⁽¹⁾	--	--	--
Apr. 3	1015	3.5	6 ⁽²⁾	30	3.5	9
Apr. 3	1645	3.5	248 ⁽³⁾	--	--	--

(1) During sample circulation, portable shield removed.

(2) Portable borated paraffin shield over sampler.

(3) No sample circulation, portable shield removed.

(4) One small spot, general radiation was about 0.5 mrem/hr at 3.5 Mw.

TABLE 3
 REACTOR CELL RADIATION DURING RUN 14

Date (1958)	Nominal Reactor Power Level (Mw)	At Fuel Circ. Pump (r/hr)	At Bl. Circ. Pump (r/hr)	Above Reactor Thermal Shield (r/hr)	Above Fuel Dump Tanks (r/hr)	Above Fuel Feed Pump ⁽¹⁾ (r/hr)
Mar. 31	1.2	1.7×10^4	6.2×10^3	2.8×10^4	1.55×10^4	2.0×10^4
Apr. 1	1.5	2.4×10^4	9.4×10^3	4.1×10^4	1.85×10^4	2.35×10^4
Apr. 2	2.0	4.0×10^4	1.26×10^4	7.0×10^4	2.15×10^4	4×10^4
Apr. 3	3.5	5.5×10^4	2.4×10^4	1.0×10^5	3.5×10^4	7×10^4

(1) This element is exposed to direct radiation from the core through openings in the thermal shield.

case of the feed and total purge measurements, a heat balance is made around a cooler, while for the circulating pump purge, the purge flow heater is used. These measurements are limited by the accuracy with which the cooling or heating water flow rates and temperatures can be controlled. In addition, the thermocouples on the fuel feed cooler apparently got wet in the cell flooding operation which preceded Run 14, so that heat balances on this particular cooler were of no value for the entire run.

A more direct measurement of pumping rate may be made by pumping from the condensate tank and observing the rate of decrease in the condensate weight. However, when this method is used for the fuel feed pump, a dilution of the high-pressure system occurs which results in a drop in core temperature and power level. All of the fuel pump flow rates were measured in this way about once per shift with more frequent measurements when the flow rates were being changed. Whenever possible, these measurements were made immediately after the isolation of a high-pressure sample, to allow more time for the disturbance to die out before the next sample.

Physical Inventories of D₂O

Measurements defining the physical inventory of D₂O in the reactor system were taken at least once per shift, usually in connection with a fuel high-pressure sample. This information combined with the sample results and the pump flow data, provided a basis for calculating solute inventories as well as D₂O inventories.

The calculated D₂O inventories were used to determine the net D₂O vapor transfer rates through the hole in the rupture disc between the fuel and blanket pressurizers. The operating cycle of the blanket pressurizer heaters was then controlled manually to minimize the transfer. This mode of operation was quite successful, since very low transfer rates were observed and these were usually reversed within a single shift.

Some changes in the separate fuel and blanket D₂O inventories were caused by transfer from the blanket to the fuel system in connection with the experiments on solution stability at various reactor temperatures. Other changes in the fuel inventory resulted from special additions to the fuel dump tanks. In general, the physical inventories of D₂O followed the changes quite well, although the total was consistently about 220 lb below the book value of 6570 lb.

Measurements of Reactor Power

The total power level of the reactor can best be determined by considering the entire reactor cell as a unit and measuring the energy added to or removed from the cell by all possible routes. The total nuclear power, i.e., the total rate of heat generation by nuclear processes anywhere in the cell, is the difference between measured removal and additions. The major additions of energy to the cell are through the dump tank

evaporators, pressurizer heaters and pump motors. Heat removal is principally through the reactor steam system, and the cooling water from the space coolers, recombiner condensers and letdown heat exchangers. Removals also include heat losses from the cell, estimated from zero-power heat balances. The drawback to the overall heat balance is that it requires a great number of observations, is time-consuming and, for accurate results, all flows and temperatures must be steady while the data are being collected.

A convenient, instantaneous indication of nuclear power is furnished by the nuclear instrument channels which contain compensated ion chambers connected to Log N recorders. The principal advantages of this instrumentation for following reactor power are the very wide useful range and the speed with which the instruments can follow transients in power. The indications from the neutron instruments are only relative, however, and the actual power can be determined only if the instrument calibrations have been determined by heat balances. Unfortunately, the factor relating the power to the Log N indication is subject to change for a number of reasons. First of all, the instrument itself is particularly subject to drift because the amplification of the ion chamber output depends on the characteristics of a diode, which change with time. More important, at least in the HRT, are variations in neutron leakage probability, which determines the relation between neutron flux at the ion chamber and the fission rate in the reactor. The leakage probability is strongly influenced by reactor temperature and by the distribution of uranium in the reactor. The temperature effect makes the leakage probability at 250°C only 0.84 of that at 280°C. Any shift in the distribution of uranium toward the outer part of the reactor greatly increases the leakage probability. A very small amount of uranium fissioning in the blanket can easily double the leakage, and if the blanket concentration is 0.1 of that in the core, the leakage probability for neutrons above 1000 ev is five times the probability with no uranium in the blanket. The effect of uranium in the blanket on neutron level indication is strikingly illustrated in Fig. 9. At the same time that the pressurizer level and dump tank weights indicated that transfer started from the fuel to the blanket systems, the Log N indication rose very abruptly, even though the nuclear heat generation actually began to decrease.

The justification for the statement that nuclear power actually began to decrease when the hole appeared in the core comes from the observed temperature rise of fluid passing through the core and blanket. This is another convenient way of arriving at the power level, or at least the portion generated in the fluid while it is in the reactor vessel. A recorder, TDR-6530, was set up to indicate core power directly in Mw and was used during Run 14 as the guide in maintaining power levels.

In Table 4 are tabulated power levels determined in various ways at the times when complete system heat balances were made. The core ΔT and the blanket ΔT were obtained from inlet and outlet temperatures recorded on a Brown 12-point recorder having a 0-350°C scale. (The precision in reading the temperature differences from the chart is only about 0.5°C.)

Table 4

RUN 14 HEAT BALANCE RESULTS

Balance Number	2	3	4	5	6	7	8	9	10
Time	1250	0413	1025	1717	0615	2157	0959	0825	1120
Date (1958)	Mar 29	Mar 30	Mar 30	Mar 30	Apr 1	Apr 1	Apr 3	Apr 4	Apr 4
Neutron Level (Log N)	2.7	2.3	2.4	4.8	2.6	5.0	13.0	12.0	15.3
Core ΔT ($^{\circ}C$)	8.5	8.0	8.5	15.0	10.0	18.0	30.0	30.0	41.0
Blanket ΔT ($^{\circ}C$)	1.0	1.5	1.5	3.5	1.0	1.5	4.0	5.0	5.5
Nominal Core Power (TDR-6530) (Mw)	0.95	0.98	1.02	1.80	1.20	2.00	3.51	3.47	4.50
Nuclear Power (Heat Balance) (Mw)	0.97	0.98	0.96	2.03	1.03	2.12	3.82	4.31	5.75
Core Power (From ΔT) (Mw)	1.05	1.00	1.06	1.86	1.24	2.26	3.79	3.81	5.19
Blanket Power (From ΔT) (Mw)	0.07	0.10	0.10	0.25	0.07	0.10	0.30	0.37	0.41
Total Power (From ΔT) (Mw)	1.12	1.10	1.16	2.11	1.31	2.36	4.09	4.18	5.60
Blanket Power/Core Power	0.07	0.10	0.09	0.13	0.06	0.04	0.08	0.10	0.08
Log N Factor (Heat Balance) (Mw)	0.36	0.42	0.40	0.42	0.40	0.42	0.29	0.36	0.38

The ΔT listed is actually the difference between the indicated ΔT and the ΔT indicated when the reactor was at zero power (subcritical). The nuclear power is therefore presumably proportional to the listed ΔT . The powers indicated as being calculated from the ΔT 's are based on flow rates actually measured in the HRT. These rates are around 460 gpm through the core and 270 gpm through the blanket.⁽²⁾ The instrument, TdR-6530, from which the nominal core power is read in megawatts, is a pen-type Brown strip-chart recorder, whose input is the E.M.F. difference between thermocouples on the inlet and exit of the fuel heat exchanger.

The agreement between the powers indicated by ΔT and the heat balance results is generally good, particularly at the higher power levels where both should be more accurate. The TdR-6530 indication is low because it was set up on the basis of a flow rate which was too low. The Log N factor, which is the ratio of the power given by the heat balance to the neutron level indication, was fairly steady throughout the run. The very low value for balance No. 8 is probably due to the heat balance power being low. Although the reason for the different heat balance results for balances 8 and 9 cannot be definitely established, between these two balances most of the steam flow was switched to the turbine. The discrepancy may be related to the fact that different steam orifices were used to measure steam flow, the largest item in the heat balance.

The ratio of nuclear heat generation in the blanket to that in the core appeared to be higher than the 0.044 which was predicted.⁽³⁾ Blanket powers were so low, however, that error in temperature measurement introduced considerable uncertainty in the calculated ratios.

(2) R. Van Winkle and D. Bruley, "Estimated Flow Rates in HRT Fuel and Blanket High-Pressure Systems," ORNL CF-58-7-36 (July 3, 1958).

(3) M. C. Edlund and P. M. Wood, "Physics of the Homogeneous Reactor Test Statics," ORNL-1780 (Aug. 27, 1954).

INDICATIONS OF SOLUTION INSTABILITY

During Run 14 there were many indications that the amounts of uranium, sulfate and copper in circulation changed from time to time. The chemistry of the fuel solution is discussed on pages 49 through 65. The paragraphs which follow present some observations, not all of which depend on chemical analyses, relative to the behavior of the solution.

Comparison of Concentrations

There are three ways in which the concentration of uranium in the core can be estimated.⁽¹⁾ First, it can be calculated from a material balance on the core and the fuel dump tanks, using the "book inventory," i.e., the "known" amount of uranium in the reactor. The uranium concentration can also be estimated from the critical temperature and a calibration of critical temperature vs concentration. Finally, the concentration can be determined by taking a solution sample and having it analyzed. Each of these methods of estimating the core concentration, with its peculiar advantages and disadvantages, is discussed in the Appendix. By comparison of the concentrations obtained by the three methods, certain conclusions can be reached regarding the behavior of the fuel.

Figures 7 and 8 show ratios of critical and analytical concentrations to the concentrations calculated from a material balance based on the book inventory. The purpose in presenting the data in this form is to normalize out the large variations in concentration produced intentionally by changes in dump tank weight or feed rate. The book inventory used in these calculations includes some material known to be out of solution at the beginning of the run. Therefore, it is not surprising that the ratios are, with one exception, all less than unity. Although the points are low from the start, there appears to be a downward trend during the run. Any decrease in analytical concentration relative to the calculated concentration indicates a decrease in the amount of uranium in circulation. A decrease in the ratio of the effective critical concentration to the calculated concentration indicates a decrease in the average nuclear effectiveness of the uranium in the reactor. This would result from deposition of some of the uranium in a region where the nuclear importance is less than the average importance of the uranium in solution.

As may be seen from Figs. 7 and 8, there are sometimes considerable differences between the analytical concentrations and the concentrations based on the critical temperature. These two concentrations are compared directly in Fig. 10, in which each shaded point is the ratio of the effective critical concentration at the time of the sample to the concentration observed in the sample. Differences between analytical and

⁽¹⁾J. R. Engel et al., "Summary of HRE-2 Run 13 (Initial Power Operation)," ORNL CF-58-10-115, p 29, 30 (Oct. 29, 1958).

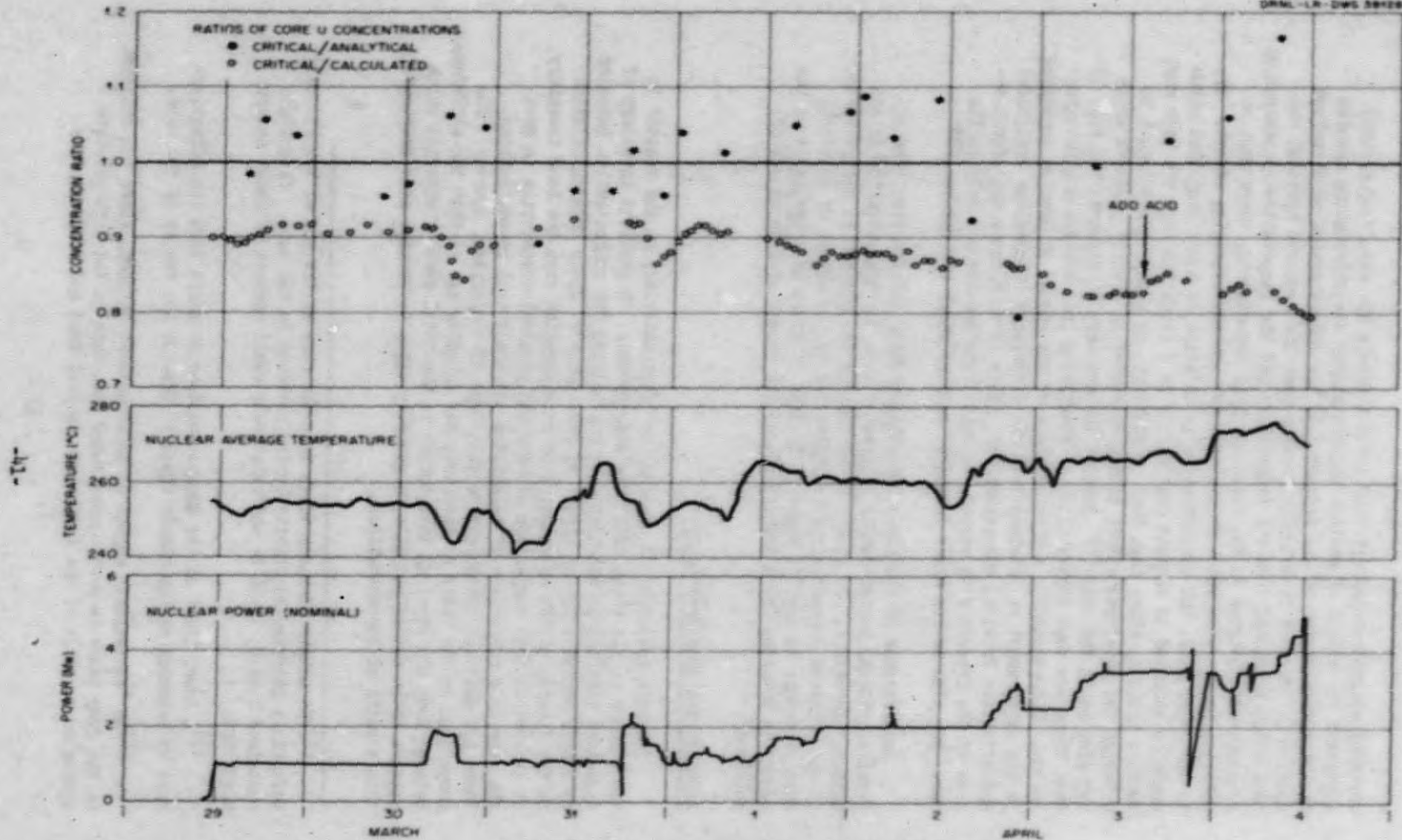


Fig. 10 . Concentrations , Temperature and Power During Run 14.

critical concentrations (if they are not merely the result of errors) represent the effect of neutron poisons and/or non-circulating uranium in the active region of the reactor. If the concentration of dissolved uranium in the core is actually greater than the required critical concentration predicted from the temperature and the temperature-concentration relationship observed in the clean critical experiments, there must be material in the core which is having a net neutron poisoning effect. If, on the other hand, the concentration in solution is lower than that which would be required to sustain criticality in a clean reactor, the core must contain uranium, other than that in solution, in quantity sufficient to outweigh the poisoning effect of any corrosion or fission products which may also be in the core.⁽²⁾ Despite the scatter, it appears from Fig. 10 that there was some contribution to reactivity from uranium not in circulation. Unfortunately, a firm estimate of the amount of uranium deposited in the core cannot be obtained from the difference between the analytical and effective critical concentrations. This is so because the effectiveness of the deposit with respect to reactivity depends not only on its mass, but also on its exact location, both of which are unknown.⁽³⁾

Because some of the non-circulating uranium is effective, the effective critical concentration is greater than the concentration in solution. Therefore, the inventory of uranium in solution is overestimated if the critical concentration is multiplied by the weight of solution. This procedure is useful, however, in that it gives an upper limit on the amount in solution (or a lower limit on the amount of uranium out of solution).

Physical and Book Inventories

The book inventory of uranium (or other material) in the reactor is the balance of all known additions and removals. A physical inventory of dissolved material is calculated by multiplying the concentration revealed by sample analysis by the amount of solution.⁽⁴⁾ Since this calculation is the inverse of the calculation of concentration from the book inventory, it is obvious that comparison of book and physical inventories is much the same as a comparison of calculated and analytical concentrations. There are distinct advantages to each type of comparison, however. The comparison on the basis of inventory has an advantage in that the difference between book and physical inventories is the total mass of material which appears to be out of solution, a quantity which is perhaps more meaningful than a ratio of concentrations.

⁽²⁾The sample of scale taken from the core at the end of Run 13 would definitely increase reactivity by its presence in the core. This scale contained 2.0% U, 10.4% Fe and relatively small amounts of other neutron absorbers.

⁽³⁾A lower limit on the amount could be obtained from its effectiveness by assuming an importance equal to that at the center of the core.

⁽⁴⁾An "equivalent" amount of solution must be used, since the solution in the dump tanks is more concentrated than that in the high-pressure system as a result of the difference in fuel feed and letdown rates.

Figure 11 compares book and physical inventories of uranium, copper and total sulfate. Other components, such as nickel and free acid, are not presented because book inventories cannot be maintained for them, at least not in the same way as for uranium, copper and sulfate. On the plot of uranium inventories, the dashed line represents the upper limit on the amount of uranium in solution, estimated from the effective critical concentration.⁽⁵⁾

Power Disturbances

During the part of the run when the power was above 2 Mw, there were numerous occurrences in which the nuclear power, as indicated on the Log N instrument, abruptly surged up and quickly fell back. (The largest excursion was about 20% of initial power, and the duration of the surges was generally less than half a minute.) These nuclear power excursions produced no great disturbances in the system operation, but they were accompanied by surges in fuel pressurizer level and minor temperature changes. Because of the compressed power scale on the Log N instrument and because the automatic instrument standardization produced a very similar swinging of the pen, the existence of the power excursions was not recognized during the course of the run. These excursions or "pips" became the subject of study when, in later runs, they became larger and their effects more noticeable.⁽⁶⁾ Reexamination of Run 14 chart records disclosed the existence of numerous pips at powers above 2 Mw. At 3.5 Mw the average frequency of pips larger than 7% (the lower limit of sure detection) was about 0.75 per hour. Although the details are not known, the mechanism causing the pips is most likely the random shifting of uranium within the active region of the reactor, as by uranium breaking away from a deposit and being swept up through the core.

Discussion

The solution behavior, at least in its gross aspects, is probably revealed most clearly by the comparisons of criticality information with book values of concentration or inventory. Such comparisons avoid the errors of chemical analysis and give a more nearly continuous picture of the situation. The same picture is shown by either the plot of critical/calculated concentrations in Fig. 10 or the inventory plot in Fig. 11, which shows the upper limit of circulating inventory based on the effective critical temperature. This comparison does involve calculating

⁽⁵⁾As remarked before, the presence in the core of deposited uranium makes the effective critical concentration greater than the actual concentration of dissolved uranium.

⁽⁶⁾P. R. Kasten et al., "HRP Quar. Prog. Rep. Oct. 31, 1958," ORNL-2654, p 19.

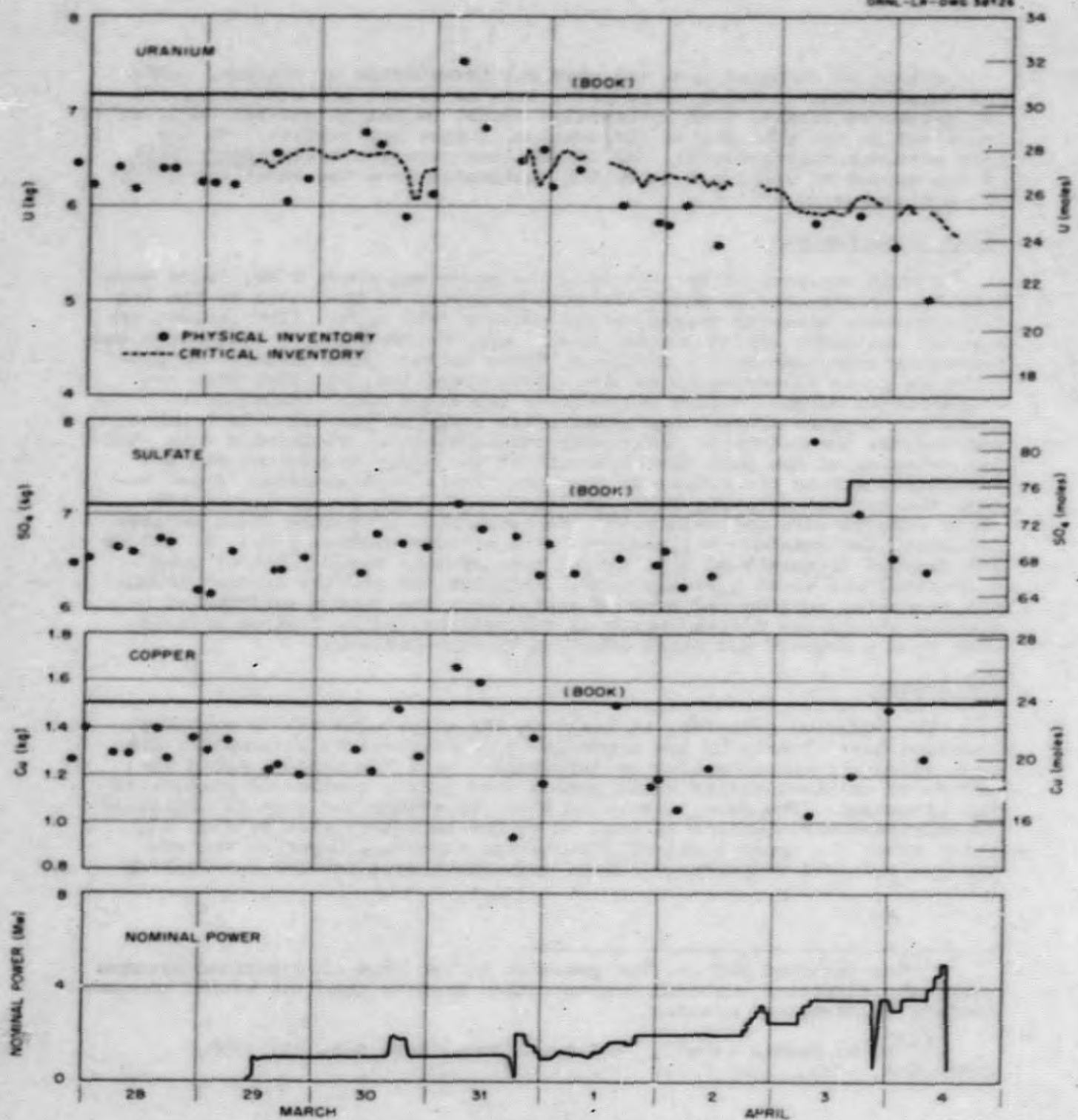


Fig. 11. Book and Physical Inventories of U, SO₄ and Cu During Run 14.

an equivalent amount of solution, but no calculations were made for times when the system conditions were changing very rapidly, so that errors are minimized. The possible error in critical concentration due to uncertainty in arriving at the nuclear average temperature from the inlet and outlet temperatures is appreciable at the higher powers near the end of the run, however. (It appears from other considerations that the approximation used is not seriously in error.) Some of the noteworthy features of the comparisons of criticality and book values for concentration and inventory are as follows:

1. The points at the 1-Mw power level suggest that between March 29, and the power increase on March 31, there was actually an increase of perhaps 2% (about 150 g) in the amount of effective uranium.
2. The first power increase to 2 Mw was quickly followed by the loss of at least 450 g of uranium. When the power was dropped back to 1 Mw, there was a rapid recovery of most of the uranium, followed by a gradual recovery of the remainder.
3. Variation of core temperatures from 245 to 270°C at a power of 1 Mw had no detectable effect on the amount of effective uranium.
4. After the fast rise to 2 Mw on March 31, there was no significant loss for about one hour, then in the next ninety minutes there was a loss equivalent to about 350 g of uranium. Recovery was gradual as the power was lowered in stages to 1 Mw.
5. Some uranium was lost during subsequent operation at 2 Mw, although not as much as during the first rise to that power.
6. The loss increased when the power was raised to 3.5 Mw on April 3.
7. The addition of the acid was followed by an increase in the effectiveness of the uranium, equivalent to the recovery of about 130 g. The effect may have been only temporary, however.
8. There was a continuous downward trend in the amount of effective uranium during the final four hours of power operation. (Some of the apparent decrease is probably due to increasing error in evaluation of the nuclear average temperature.)

The comparison of effective critical concentration with the analytical results (Fig. 10), which should reveal the effect of uranium deposited in the core is somewhat clouded by scatter which is probably due to analytical errors. A few of the extreme points can definitely be attributed to errors. The point at 2114 on April 2, is suspect because of abnormally high ratio of uranium to other ions and a poor balance of cations and anions in the sample (see Fig. 13). Several other samples show the same evidence of error, but to a lesser degree. The concentrations reported for the

sample at 0548 on March 31, appear to be too high for all components, particularly uranium and copper. Use of the analytical concentrations gives physical inventories even higher than the book inventories. This cannot be attributed to error in the equivalent amount of solution used in calculating physical inventory, because the criticality indication and the book inventory were compared using the same value, and results were not abnormal. The very high point at 0750 on April 4, appears to be valid. Although the scatter in the data makes the conclusion somewhat questionable, there appears to be an upward trend during the run. Thus it appears that there was an amount of deposited uranium in the core which increased as time went on, presumably due to the increasing power. (7)

An observation regarding the approximation of nuclear average temperature is in order at this point. The approximation used in the calculations, namely, that the core average is the mean of the inlet and outlet, probably underestimates the nuclear average temperature and effective critical concentration. Thus a portion of the apparent drop in critical concentration during the power increases on April 3 and 4, is due to an increasing error at the higher powers and temperature spreads. Use of a core average temperature nearer the outlet would have made the apparent decrease in critical concentration less; it would have greatly enlarged the difference between effective critical concentration and the analytical concentration, however.

It appears that the best approximation for the core average temperature is to use a value between 0.65 to 0.80 for θ in the equation

$$T_{av} = T_{in} + \theta (T_{out} - T_{in})$$

(see page 67). To illustrate the effect of a more realistic approximation, using $\theta = 0.7$ instead of 0.5 would raise the critical/analytical ratio at 0750 on April 4, from 1.165, as shown on Fig. 10, to 1.206 and the critical/calculated from 0.828 to 0.858. The critical/calculated ratio at 1215 would be raised to 0.832 by the changed temperature approximation.

Comparison of book and physical inventories of uranium shows that at the beginning of the run some 800 to 900 g of uranium was out of solution. With the beginning of power operation, scatter increased but the average physical inventory at 1 Mw was 220 g higher than the average while the reactor was subcritical. The physical inventories decreased later as the power was raised. Figure 12 shows a plot of differences between book and physical inventories as a function of power level. The line is a least-squares fit of all the points (including the negative point). If

(7) In later runs, particularly Run 17, the effective critical concentration was found to be greater than the analytical concentration by an amount which very definitely increased with power level.

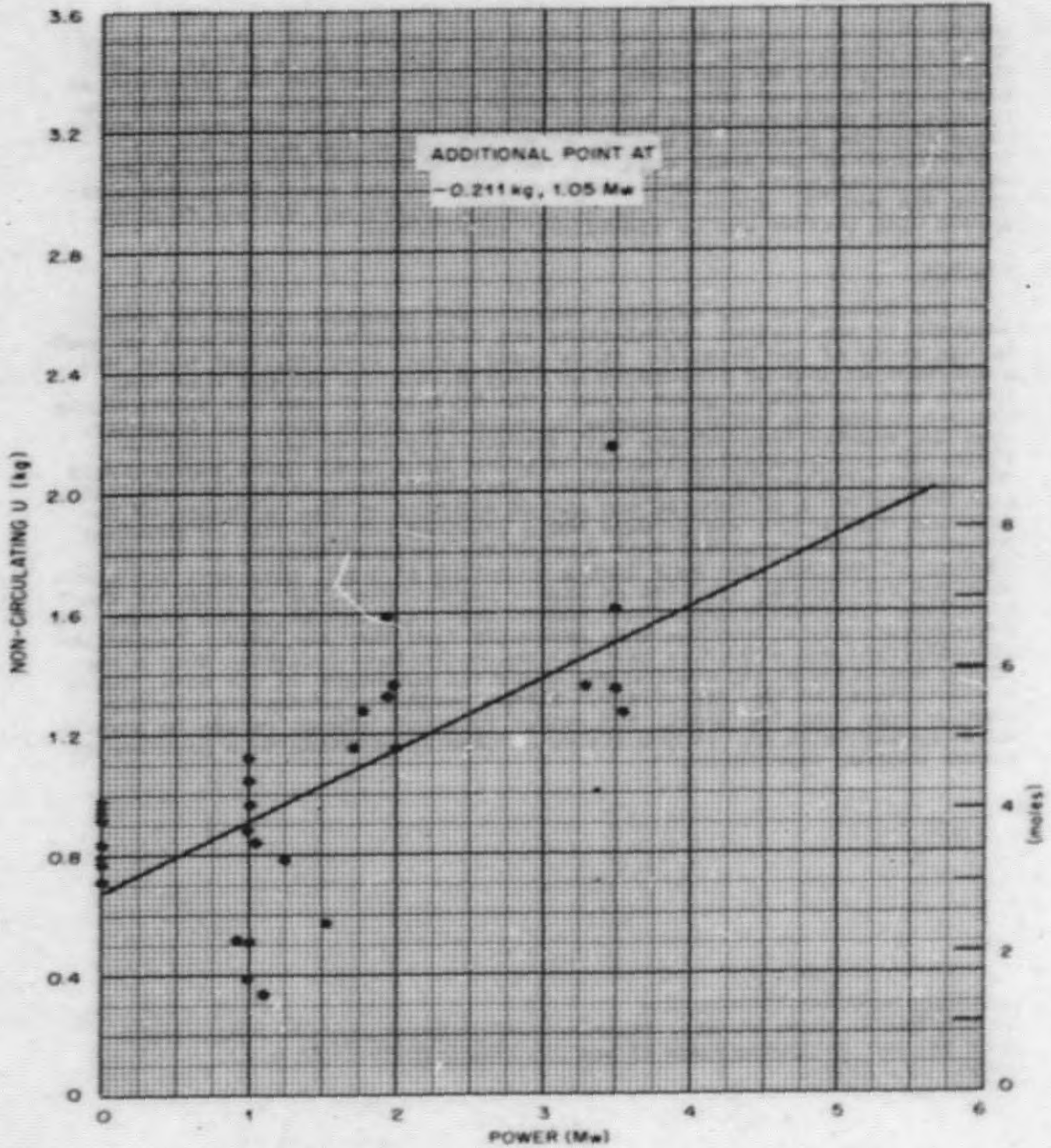


Fig. 12. Non-Circulating U vs. Reactor Power During HRT Run 14.

the decrease in non-circulating uranium between zero and 1 Mw is assumed to be an effect of time, rather than power, and a line is fitted to only those points at 1 Mw and above, the slope is steeper and the intercept at zero power is around 400 g. It is enlightening to compare the foregoing figures for non-circulating uranium with the results of analyses of scale taken from the reactor at the end of Run 13 and at the end of Run 14. The evidence of the scale analyses was that about 700 g of uranium was held up on the walls at the beginning of Run 14 and about 300 g remained after the run.⁽⁸⁾ Plots of non-circulating copper and sulfate vs power showed wide scatter and no significant correlation.

Summary

The details of the solution behavior are somewhat clouded by errors inherent in the various calculations and determinations which must be used in the study of the details. It is quite clear, however, that there was a substantial loss of reactivity at times during the run and that the losses were related to power level. The evidence of loss was particularly obvious during the early attempts to raise the power above one megawatt, when the reactor temperatures fell dramatically after power increases. A plot of non-circulating uranium vs power level shows quite convincingly the effect of power on the solution. From this plot, it may be inferred that 800 to 900 g of uranium was out of solution at the beginning of the run, and that at the end perhaps 400 g remained out. A plot of physical inventory vs time suggests that some of the material missing at the beginning of the run came back during operation at 1 Mw, then went out again when the reactor was operated at higher power. The time behavior of the effective critical concentration showed that time constants for either loss or recovery of uranium were generally less than one hour. Comparison of analytical and effective critical concentrations indicates that a part of the non-circulating uranium was deposited somewhere in the core, where it contributed to the reactivity. Further evidence of non-circulating uranium came from the small, but rather frequent, abrupt surges in power, which appear to be due to quantities of uranium shifting to a position of higher nuclear importance.

⁽⁸⁾G. H. Jenks et al., "Examination of Specimens and Scales Taken From the HRT Following Runs 13 and 14," ORNL CP-58-9-37, p 11 (Sept. 11, 1958).

FUEL SOLUTION CHEMISTRY AND CORROSION

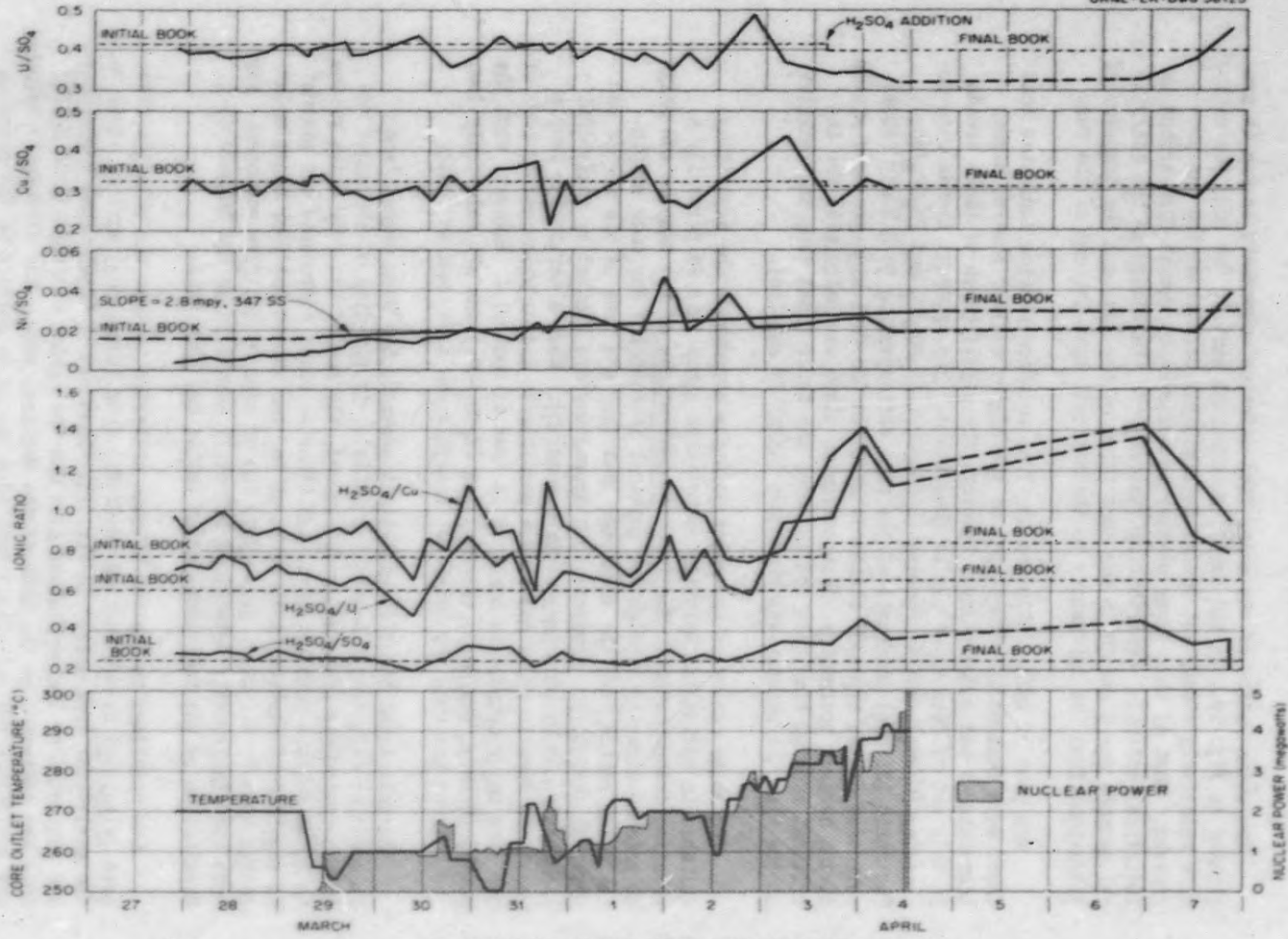
The molar ratios of the major fuel solution components in Run 14 are plotted in Fig. 13. In the early, subcritical part of the run, the ratios were relatively steady but became increasingly erratic as the run progressed. Most of this large scatter is believed to result from errors in analytical control. Contributing factors were the change in the analytical methods for U and Cu in the last third of the run, and the transfer of titrations from the laboratory to the hot cell near the end of the run.

The plot of molar ratios of the fuel solution samples indicates that there were changes in composition at the higher nuclear powers, higher temperatures and longer times; however, the significance of these trends in this set of data alone may be questioned. The apparent trends were decreasing (U/SO_4) , and increasing (H_2SO_4/Cu) , (H_2SO_4/U) and (H_2SO_4/SO_4) . The ratio (Ni/SO_4) increased at the rate corresponding to 2.8 mpy (180%) corrosion of the type-347 stainless steel in the fuel high-pressure system during nuclear operation. This corrosion slope was calculated for the critical running time only and includes the three final dump tank samples at the time the reactor was shut down (total 29 samples).

The molar ratios of the major solution components are especially valuable in estimating solution composition since this is primarily a relative determination. The ionic concentrations of two samples can only be related directly through the physical inventories for each sample. The molar ratios have the advantage that they are free of the errors of physical inventory. However, both approaches are limited by analytical inaccuracy and by nonrepresentative sampling. When a particular sample result is erratic it is not always possible to deduce which one or more of the five major components analyzed is 'out of control.' However, when the ratios (U/SO_4) , (Cu/SO_4) , (Ni/SO_4) and (H_2SO_4/SO_4) are all relatively low, the chances are that the total SO_4 analysis was high, and so forth.

An additional index of analytical accuracy has been termed 'ion balance' or 'electrical neutrality ratio': if all five of the major solution components are exactly determined, the sum of the four molar ratios to total sulfate must be 1.00. (If other solution components are present in significant amounts, such as Mn or Fe, the sum of the four major molar ratios must be < 1.00 .) Here again two or more of the five components may have been incorrectly determined but still result in an ion balance of 1. Examination of the individual molar ratios may show this.

The corollary to the ion balance criterion is that the solution free acid calculated by subtracting $(U+Cu+Ni)$ from (total SO_4) must be equal to the analytical free acid. In Fig. 13, the ratios of (analytical H_2SO_4/SO_4) are connected by the continuous line and the ratios of (calculated H_2SO_4/SO_4) are indicated at the ends of the vertical bars. The difference between the analytical and calculated free acid ratios to sulfate is the error in 'ion balance' and is shown graphically by the length of the vertical bars. The usefulness of the ion balance criterion has been limited by the poor



-05-

Fig.13 HRT Run 14 Molar Ratios of Chemical Data.

dependability of the analysis for free acid. However, since pH measurement in reactor solutions has not proved useful, the free acid analysis is essential as a check on the analytical control of the other major solution components.

The molar ratios calculated from "Book Inventory" are indicated by dashed lines in Fig. 13. This book inventory is based on the additions made to the system at the start of Run 14, and upon an estimate of the fraction of Run 13 inventory which remained in the reactor after transfer of this inventory for processing. The estimated fraction of U remaining was 0.1443 and was based on the recovery of Run 13 uranium after processing. The estimated fractions of Cu and SO₄ remaining were 0.1312 and 0.0876, respectively, and were based on the average (U/SO₄) and (Cu/SO₄) ratios of the last eleven Run 13 dump tank samples and the estimated fraction of U remaining. The estimated fraction of Ni remaining was assumed to be identical to the fraction of U remaining; the Ni inventory estimated at the end of Run 13 was the average of

$$\left[\left(\frac{\text{Ni}}{\text{SO}_4} \right) \times \text{Book SO}_4 \right]$$

for 23 samples after nuclear operation. The average of the first eleven, (subcritical) samples is below the book ratio of Fig. 13 for (U/SO₄), (Cu/SO₄) and (Ni/SO₄); it is above the book ratio for (H₂SO₄/Cu), (H₂SO₄/U) and (H₂SO₄/SO₄).

In order to examine the Run 13 to 14 inventory transition more closely, the molar ratios at the end of Run 13 and at the beginning of Run 14 were plotted in Fig. 14. The book ratios for Run 14, shown as dashed lines in this plot, assume that the same fraction of Run 13 inventory for all components, 0.1443, was transferred to Run 14. There is better agreement between the Run 14 subcritical samples and 'book ratios' in this instance. For comparison, other book ratios are indicated at the right margin of Fig. 14. Book ratios labeled A were calculated from the fresh Run 14 fuel charge, book ratios labeled B are those drawn in Fig. 13 and were defined previously.

Although conclusive proof is not possible, the evidence from the molar ratios of the Run 14 subcritical solution samples is that there was proportional transfer (0.1443 fraction) of Run 13 inventory to Run 14 and that this inventory was well mixed with the new fuel charge, although not necessarily completely in solution or accounted for by physical inventory in Run 14. If this was the case, it is possible that the failure to transfer the last 14% of Run 13 inventory from the reactor system had the same cause as the physical inventory discrepancy during previous runs. In an early U-238 run the physical inventory discrepancy was -15% while the solution was being circulated, but about 98% was recovered when the reactor was drained.

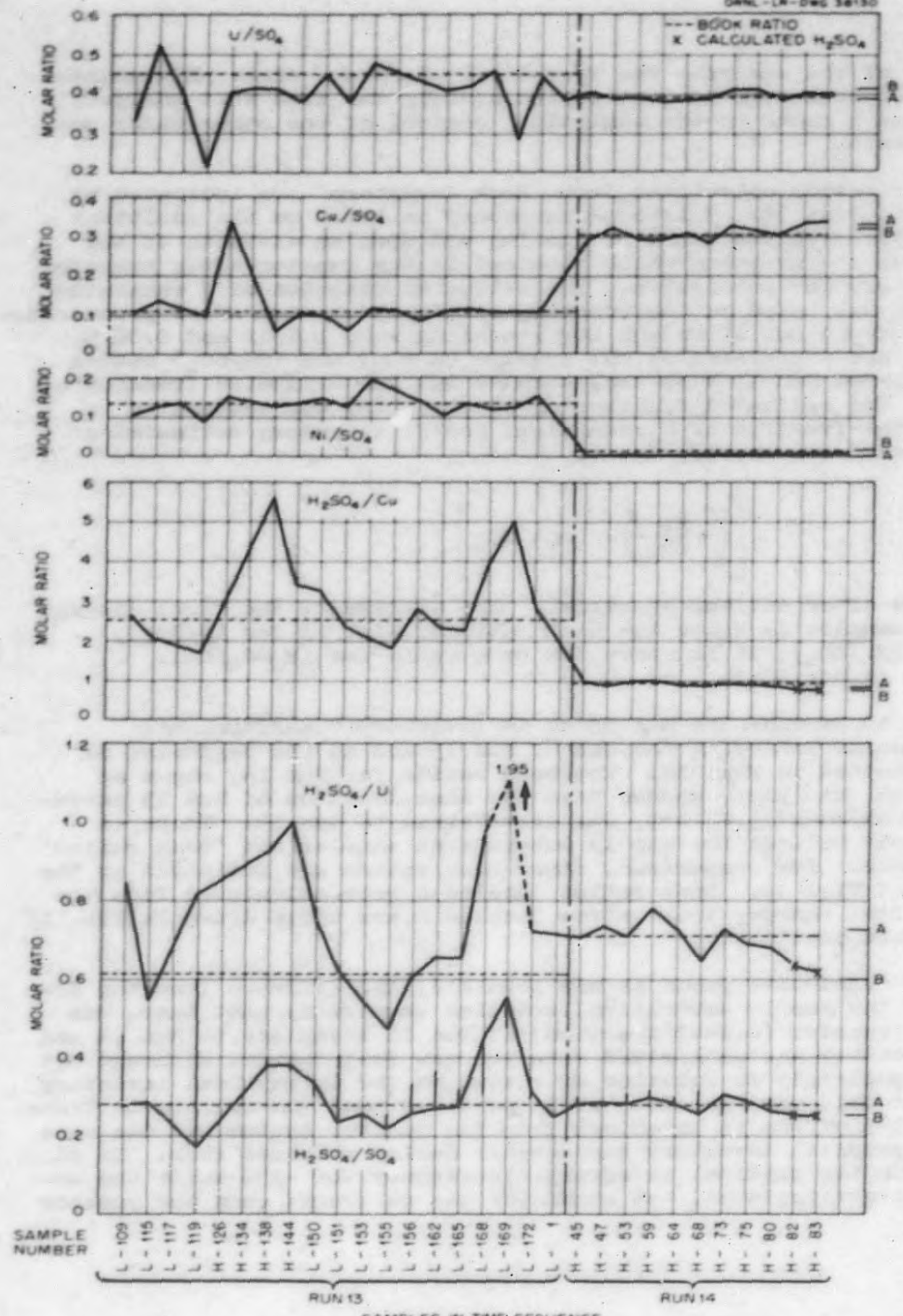


Fig. 14. Run 13 to 14 Inventory Transition.

The alternative explanation for the transfer of part of the Run 13 inventory to Run 14 is that the greater part of the transferred inventory remained in the reactor as partly, or temporarily, insoluble materials. This explanation is supported by data on the composition of reactor scales.⁽¹⁾ In either case, or with a combination of both explanations for the inventory transfer, the book inventory for Run 14 cannot be exactly determined.

The ratio (Ni/SO_4) is considerably below the estimated initial book level in Fig. 13 during subcritical operation and, as plotted, gives the impression of rising sharply after the start of nuclear operation. (The nickel level of the subcritical samples was confirmed by the last two samples before nuclear operation when these samples were analyzed one year later and found in close agreement with the earlier analyses.) However, the (Ni/SO_4) data can also be described by a smooth line which rises with time and then may decrease toward the end of the run when (U/SO_4) apparently decreases. This description is consistent with an interpretation of the composition of reactor scales which has been proposed.⁽¹⁾

The corrosion attack rate on a type-347 stainless-steel specimen at the core access flange in Run 14 was reported to be essentially zero mpy, and the decrease in the Ni content of corrosion scales between the end of Run 13 and the end of Run 14 was reported to account for most of the Ni which appeared in solution during Run 14.⁽¹⁾ Therefore, the actual corrosion rate of type-347 stainless steel in Run 14 was postulated to be much less than the apparent 2.8 mpy which was calculated from the run data: the estimated decrease in the Ni content of scales was 91 g, the average solution inventory of Ni during Run 14 power operation was 100 g, the difference is equivalent to an average corrosion rate of 0.5 mpy during nuclear operation.

The complete chemical and spectrographic fuel solution data for Run 14 are recorded in Table 5. It may be noted that some data are presented here which are not plotted on Fig. 13. The reasons for these omissions follow. If a particular sample was obviously too small for wet chemical analysis, it was submitted for spectrographic analysis only (samples 97 and 131). One small sample, 181, was submitted for both wet chemical and spectrographic analysis. However, since only partial chemical results were obtained and no duplicate analyses were run, the information for this sample was not plotted on Fig. 13. Another sample, 102, was spilled before the chemical analyses were completed, so it was not shown on Fig. 13 even though the partial results are given in Table 5.

⁽¹⁾G. H. Jenks, A. R. Olsen, and W. C. Yee, "Examinations of Specimens and Scales Taken from the HRT Following Runs 13 and 14," ORNL CF-58-9-37 (Sept. 11, 1958).

Many radiochemical analyses were also performed on these reactor samples, but these data are not recorded here because of the volume and the complexity involved in interpretation; the conclusions reached on the behavior of radionuclides in the HRT have been reported elsewhere. (2), (3).

The frequent occurrence of Al in the spectrographic analyses has been attributed to contamination during analysis, because the Al concentration was very erratic by comparison with other ions. However, it also may have come from the alumina pellets in the catalytic recombiner in the low-pressure system. Later control studies of the spectrographic Cu analysis showed that there was variable contamination from brass equipment in the spectrographic hot cell. Since this contamination might occur either in the reactor sample or in the standard solution run in series with every reactor sample, the wide range in the spectrographic Cu data resulted.

The spectrographic Cr analyses on the subcritical samples consistently showed a solution inventory of 1 mole, but Cr dropped below detection shortly after the beginning of nuclear operation. Near the end of the run Cr again became detectable and was first observed at 2.5 Mw. In the last two samples taken during nuclear operation, at 3.5 Mw, 1.2 and 1.4 ppm Zr were also reported. (These results were believed to be low due to poor recovery of Zr from the ion exchange columns used to separate uranium from spectrographic samples.)

A similar increase in Cr concentrations in in-pile studies has been reported to occur at high temperatures, power densities and sulfuric acid concentrations. (4) The appearance of Zr in solution is evidence of accelerated attack of the Zr-2 core vessel at the higher power levels. This effect was also originally observed in in-pile studies. (5)

Although the steady-state concentrations of Cr and Zr in the fuel solution at higher powers cannot at this time be quantitatively related to corrosion rate, their appearance in solution seemed to be a good indicator of trouble since the core vessel failure occurred shortly thereafter at the nuclear power of 5 Mw (nominal).

The data obtained in Run 14 blanket samples are recorded in Table 6. There is no evidence in these results that the leakage of uranium occurred prior to 1232 on April 4. The blanket pH apparently did drop to 3 several

(2) R. A. McNees et al., "HRP Quar. Prog. Rep. April 31, 1958 and July 31, 1958," ORNL-2561, p 267-68.

(3) R. W. Stoughton et al., "HRP Quart. Prog. Rep. April 31, 1958 and July 31, 1958," ORNL-2561, p 321-24.

(4) J. C. Banter, J. E. Baker, and R. J. Davis, "Analysis of the Status of Chromium in Solution Under In-Pile Conditions," ORNL CF-58-7-63 (July 15, 1958).

(5) G. H. Jenks, "Effect of Radiation on the Corrosion of Zircaloy-2," ORNL CF-57-9-11 (Sept. 30, 1957).

hours prior to the occurrence of the hole, but it is unlikely that the leakage of uranium solution was responsible for the low pH; no abnormal fission-product activity was detected in this sample.

With minor exceptions, all of the radionuclides identifiable in blanket samples by gamma spectrometry were either gaseous nuclides or the longer-lived descendants of gaseous nuclides and could be accounted for by the transfer of fission gases from the core pressurizer to the blanket pressurizer. These gases were probably dissolved in the condensate pumped from the low-pressure system to the core pressurizer as purge, and transferred to the blanket via the hole in the rupture disc between the two pressurizers.

TABLE 5
SUMMARY OF HWY RUS 14 FUEL SOLUTION DATA

Sample No., Date ('58), Time	Volume (ml) ⁽¹⁾	Analytical Data: Chemical (Spectrographic) ⁽²⁾									
		pH	Acid (mg SO_4^{2-} /ml)	U^{235-2} (mg/ml)	Total (mg SO_4^{2-} /ml)	Cu (mg/ml)	Ni (μg /ml)	Mn (μg /ml)	Cr (μg /ml)	Fe (μg /ml)	Other (μg /ml)
Initial Stock Inventory: A. 6259 g U^{235-2} , 6985 g SO_4 , 1445 g Cu, 0 g Ni; fresh fuel charge. 2-27, 1700											
B. 7175 g U^{235-2} , 7096 g SO_4 , 1501 g Cu, 71 g Ni; see footnote (3).											
C. 7175 g U^{235-2} , 7408 g SO_4 , 1507 g Cu, 71 g Ni; see footnote (4).											
B-45 3-27, 2148	~12	1.35	2.40	8.33	8.37	1.635 (1.16)	20 (30)	(2.6)	(51)		
B-47 3-28, 0048	14.1	1.34	2.39	8.67	9.09	1.949 (0.8)	28 (20)		(400)		
B-53 3-28, 0624	15	1.05	2.55	8.80	9.12	1.771 (2.12)	35 (44)	(4.2)	(60)		
B-59 3-28, 0946	9.7	1.00	2.74	8.61	9.17	1.790 (1.46)	27 (37)	(3.7)	(60)		
B-6A 3-28, 1525	3.0	1.10	2.64	8.90	9.40	1.941 (0.38)	36 (15.6)		(35)		Cl 0 F 0

TABLE 5 (Continued)

Sample No., Date ('58), Time	Volume (ml) (1)	Analytical Data: Chemical (Spectrographic) (2)									
		pH	Acid (mg SO ₄ ²⁻ /ml)	U ²³⁵⁻² (mg/ml)	Total (mg SO ₄ ²⁻ /ml)	Cu (mg/ml)	Bi (μg/ml)	Mn (μg/ml)	Cr (μg/ml)	Fe (μg/ml)	Other (μg/ml)
B-68 3-28, 1745	4.6	1.0	2.35	8.90	9.34	1.770 (1.24)	43 (54)	(40)	(86)	(68)	(Al 8) (Zn 200)
B-73 3-28, 2310	4.6	1.2	2.59	8.71	8.62	1.885 (1.4)	34 (52)	(4.6)	(98)		
B-75 3-29, 0800	4.3	1.1	2.45	8.70	8.60	1.816 (1.6)	43 (58)	(4.6)	(100)		(Al 4.1)
B-80 3-29, 0557	6.9	1.1	2.40	8.61	9.12	1.865 (1.0)	44 (52)	(3.9)	(60)		
B-82 3-29, 0645 (Analyzed 3-29-59)	4.6			9.03	9.18	2.05	53 (43)	(2.5)	(116)	(24)	(Al 3.7)
B-83 3-29, 0945 (Analyzed 3-29-59)	4.3			9.00	9.11	2.03	51 (43)	(2.3)	(116)	(20)	(Al 4.8)
B-86* 3-29, 1508	15	0.90	2.31	8.95	8.75	1.668 (0.96)	62 (60)	(12.8)	(7.4)		(Al 12.6)

TABLE 5 (Continued)

Sample No., Date ('58), Time	Volume (ml) ⁽¹⁾	Analytical Data: Chemical (Spectrographic) ⁽²⁾									
		pH	Acid (mg SO ₄ ²⁻ /ml)	U ^{235.2} (ng/ml)	Total (mg SO ₄ ²⁻ /ml)	Cu (mg/ml)	Ni (μg/ml)	Mn (μg/ml)	Cr (μg/ml)	Pb (μg/ml)	Other (μg/ml)
H-88 ^a 3-29, 1705	~15	1.10	2.26	8.24	8.74	1.691 (1.12)	77 (104)	(15)			(A1 14.8)
H-95 ^a 3-29, 2131	~15	1.20	2.35	8.62	8.96	1.644 (1.4)	86 (128)	(15.4)			
H-97 ^a 3-30, 0110	1	1.29				(0.84)	(106)	(9.6)			(A1 4.2)
H-102 ^a 3-30, 0622	5.0	1.25		9.29		1.884	103				
H-103 ^a 3-30, 0912	4.2	1.20	1.78	9.30	8.75	1.795 (1.24)	74 (90)	(14.6)			(A1 5.6)
H-106 ^a 3-30, 1225	5.0	1.4	2.16	9.11	9.28	1.667 (0.94)	95 (54)	(17.6)			
H-108 ^a 3-30, 1755	6.5	1.40	2.45	8.04	9.14	2.021 (1.02)	95 (84)	(8.8)			(6) (A1 VFT)

TABLE 5 (Continued)

Sample No., Date ('58), Time	Volume (ml) (1)	Analytical Data: Chemical (Spectrographic) (2)									
		pH	Acid (mg SO ₄ ²⁻ /ml)	^{235.2} (mg/ml)	Total (mg SO ₄ ²⁻ /ml)	Cu (mg/ml)	Ni (μg/ml)	Mn (μg/ml)	Cr (μg/ml)	Fe (μg/ml)	Other (μg/ml)
H-112 ^a 3-30, 2226	9.5	1.40	5.05	8.50	9.21	1.778	139				
H-118 ^a 3-31, 0548	9.7	1.90	2.74	7.29	8.77	2.039	94				
H-123 ^a 3-31, 1030	4.6	1.70	2.98	9.29	9.30	2.170 (2.56)	84 (176)	(16)			(Al 7.2)
H-126 ^a 3-31, 1530	15	2.00	2.21	10.09	9.92	2.435 (2.0)	144 (106)	(7.4)			
H-131 ^a 3-31, 1804	1.6	1.70				(0.86)	(108)	(12)			
H-132 ^a 3-31, 1804	10.5	1.70	2.21	8.84	9.25	1.282 (2.52)	105 (150)	(10)			(Al 13.4)
H-133 ^a 3-31, 2223	9.40	1.85	2.99	9.06	8.73	1.858 (2.12)	154 (140)	(15)			

TABLE 5 (Continued)

Sample No., Date ('58), Time	Volume (ml) (1)	Analytical Data: Chemical (Spectrographic) (2)									
		pH	Acid (mg CO_2 /ml)	U^{235-2} (mg/ml)	Total (mg SO_4 /ml)	Cu (mg/ml)	Mn ($\mu\text{g}/\text{ml}$)	Nb ($\mu\text{g}/\text{ml}$)	Cr ($\mu\text{g}/\text{ml}$)	Fe ($\mu\text{g}/\text{ml}$)	Other ($\mu\text{g}/\text{ml}$)
H-154* 4-1, 0045	7.30	1.86	2.40	8.50	9.16	1.508 (8pt11)	158 (150)	(17)			(Al 4.2)
H-159* 4-1, 0630	6.0			8.74	8.71	8.000(1 (4.6)	127 (136)	(11.8)			
H-147* 4-1, 1535	15	1.50	2.51	9.12	9.90	2.262 (0.78)	112 (158)	(15.2)			
H-146* 4-1, 1736	4.9	1.20	5.51	15.11	15.66	5.262 (8.5)	145 (250)	(16.4)	(K2, PT)		(Al 3.1)
H-150* 4-1, 2245	15	1.40	2.74	8.79	9.71	1.750 (1.6)	175 (162)	(18.4)			
H-153* 4-2, 0049	4.5	1.30	6.24(7)	8.67	10.02	1.790 (2.72)	248 (140)	(16)			(Al 11.6)
H-158* 4-2, 0442	10.2	1.50	2.40	8.99	9.32	1.570 (1.2)	112 (162)	(12)			

TABLE 5 (Continued)

Sample No., Date ('58), Time	Volume (ml) ⁽¹⁾	Analytical Data: Chemical (Spectrographic) ⁽²⁾									
		pH	Acid (mg SO ₄ ²⁻ /ml)	^{235.2} (mg/ml)	Total (mg SO ₄ ²⁻ /ml)	Cu (mg/ml)	Ni (μg/ml)	Mn (μg/ml)	Cr (μg/ml)	Fe (μg/ml)	Other (μg/ml)
H-166* 4-2, 1029	11.0	1.40	2.79	8.40	9.79	1.900 (1.4)	154 (112)	(9.2)			
H-169* 4-2, 1900	3.8	1.40	2.45	9.73	9.64	2.137	227				
H-174* 4-2, 2114	5.0	1.5	2.93	12.39	10.30	2.589 (17.2)	137 (200)	(13.6)			
L-180* 4-3, 0514	5.0	1.1	4.08	10.58	11.71	3.354 (8.2)	154 (260)	(20)	(PT)		
H-181* 4-3, 0750	2.80	1.20		9.90	13.17	1.750 (1.7)	(129)	(10.4)	(<2 , PT)		
Rock Inventory Change (By Addition): + 252 g SO ₄ ²⁻											
4-3, 1515											
H-190* 4-3, 1709	9.05	1.53	3.79	9.68	11.44	1.959 (1.45)	177 (130)	(15.4)	(<2 , PT)		(Al 4.8)

TABLE 5 (Continued)

Sample No., Date ('58), Time	Volume (ml) ⁽¹⁾	Analytical Data: Chemical (Spectrographic) ⁽²⁾									
		pH	Acid (mg SO ₄ ²⁻ /ml)	U ²³⁵⁻² (mg/ml)	Total (mg SO ₄ ²⁻ /ml)	Cu (mg/ml)	Ni (μg/ml)	Mn (μg/ml)	Cr (μg/ml)	Fe (μg/ml)	Other (μg/ml)
H-197 ^a 4-4, 0100	10.4	2.50	5.28	9.76	11.46	2.47 (2.0)	185 (135)	(15)	(c2, FT)		(Er 1.2)
H-200 ^a 4-4, 0750	8.1	2.20	4.18	9.05	11.52	2.302 (0.575)	138 (150)	(9.6)	(2.7)	(10.8)	(Al 4.3) (Er 1.4)
L-206 4-6, 2230	4.80	0.95	8.55	15.51	19.26	4.012 (3.15)	247 (250)	(17.5)	(18.6)	(FT)	F O
L-208 4-7, 1130	5.30	1.00	6.24	17.45	18.92	3.490 (2.3)	215 (150)	(14.8)	(17.2)	(60)	Er 1 (r) F O
L-209 4-7, 2055	2.10	0.90	7.20	22.48	20.10	5.027	471				

Final Book Inventory (Corrected for samples and nuclear burnup):

- B. 7156 g U²³⁵⁻², 7344 g SO₄, 1501 g Cu, 131 g Ni; see footnote (5).
 C. 7156 g U²³⁵⁻², 7676 g SO₄, 1506 g Cu, 137 g Ni; see footnote (5).

REFERENCES FOR TABLE 5

1. No suspended solids were observed in Run 14 fuel solution samples.
2. In most samples analyzed spectrographically, the following elements were sought: Ag, Al, Cr, Cu, Fe, Mn, Ni, Si, Sn, Ti, Zn, Zr; and if not otherwise indicated in the table were sought for but not found. Faint Trace, or Very Faint Trace, indicates that some of the lines of the element were visible, but that the line calibrated for quantitative analysis was not. See text for comment on Al and Cu analyses.
3. Book inventory ratios plotted in Fig. 13. See text for basis.
4. Book inventory ratios plotted in Fig. 14, assumes 0.1443 proportional transfer of Run 13 inventory.
5. Assumes 0.1443 proportional transfer of Run 13 inventory.
6. VFT indicates "very faint trace."
7. FT indicates "faint trace."
- * Identifies samples during nuclear power operation.

TABLE 6

SUMMARY OF HBT RUN 14 BEARER SOLUTION DATA

Sample No. Date ('58), Time	Volume (ml)	Analytical Data: Chemical (Spectrographic)							Radiochemical Data	
		pH	% D ₂ O	Cl (µg/ml)	F (µg/ml)	U ²³⁵⁺² (µg/ml)	SO ₄ (µg/ml)	Cu (µg/ml)	Date Examined	Observations
L-28 3-26, 2304	1.50			0.5	0					
L-31 3-28, 0440	1.70	5.2		1.2	0					
L-72 3-28, 2147	9.5	4.0	92.18	0	1					
E-81 3-29, 0450	6.3	6.85	99.49	3	0					
E-99 3-30, 0355	14	5.00	97.88	0	0				3/30/58	gross \bar{V} 5.91×10^5 c/n/ml.
L-114 3-31, 0145	2.0	7.0							3/31/58	Gamma spectrometer: 0.08 Mev, Strong, probably Xe-135 0.14 Mev, Moderate, probably Xe-135 0.15 Mev, Moderate, probably Kr-85m 0.25 Mev, Strong, probably Xe-135 0.36 Mev, Weak, probably I-131 0.54 Mev, Moderate, probably I-131 0.76 Mev, Weak, probably Mo-99, Sr-91(?) 1.0 Mev, Weak, probably Sr-91 (?) 1.36 Mev, Weak, probably Sr-91 1.6 Mev, Weak, probably La-140 (Ba-140 also present)

TABLE 6 (Continued)

Sample No. Date (**58), Time	Volume (ml)	Analytical Data: Chemical (Spectrographic)						Radiochemical Data		
		PH	% H ₂ O	Cl (ug/ml)	F (ug/ml)	U ²³⁵ -2 (ug/ml)	SO ₄ (ug/ml)	Ca (ug/ml)	Date Examined	Observations
E-158 4-1, 0210	2.3	4.99							4/1/58	Gamma Spectrometer: 0.08 Mv, Strong, probably Ra-153 0.15 Mv, Strong, probably Sr-90m 0.25 Mv, Strong, probably Ra-153 0.54 Mv, Moderate, probably I-133 0.76 Mv, Weak, probably Sr-91 1.0 Mv, Weak, probably Sr-91 ~1.4 Mv, Weak, probably Ra-24 ~1.6 Mv, Very Weak, probably La-140 (Gamma activity approx. 100 times E-154.)
E-154 4-2, 0235	3.00	3.80	94.85	0	0					
E-179 4-3, 0040	5.3	4.30				<0.1			4/9/58 4/10/58	Gamma Spectrometer: Most of the activity was Ra-La-140; small concentrations of Co-143, Pu-239, and Sr-90-95 may have been present.
E-158 4-4, 0430	0.7	7.45				<0.1				
E-199 4-4, 0615	5.4	3.0				<0.1		(0.74)	4/9/58 4/10/58	Gamma Spectrometer: Most of the activity was Ra-La-140; small concentrations of Co-143, Pu-239, and Sr-90-95 may have been present.
E-205 4-4, 1405	5.0	2.1				600.	666.		4/4/58	Sample surveyed 10 r/hr at 2-1/2 feet with Cattle Pie.

STEAM AND WATER SYSTEM CHEMISTRY

The fuel and blanket steam generators were operated with continuous addition of buffered phosphate solution and hydrazine solution, and with intermittent blowdown of the shell water. The pH-phosphate concentration relationship in the shell water remained below the curve for K_2PO_4 throughout this run. The pH of the reactor steam, however, ranged from 9 to 10 due to the presence of 1 to 10 ppm NH_3 . Phosphate was detected in 2 of 15 steam condensate samples but these results were questionable because of internal inconsistencies. Although hydrazine was added at increasing rates with increasing nuclear power, to compensate for the radiolytic decomposition of this material, the concentration remained below 1 ppm in the shell samples. A dissolved oxygen determination of 0.1 ppm in the fuel steam was obtained at 3.5 megawatts.

Chloride analyses ranged from 0.5 to 1.5 ppm on the fuel shell samples during this run. This high level resulted from insufficient rate of blowdown and from the relatively high chloride content of commercial hydrazine solutions.

The determination of the efficiency of hydrazine in scavenging oxygen from the reactor steam generators will require, in the future, careful control of all flow rates and analytical data in the parts per billion range so that a material balance can be made.

No radioactivity was detected in the reactor steam. Activation of the heat exchanger blowdown and of the demineralized cooling water solution ($0.1\% K_2CrO_4$) did not reach significant levels.

BEHAVIOR OF CORE TEMPERATURES

The original experiments on flow patterns in a mockup of the HRT core indicated very little mixing in the core. It was predicted on the basis of these experiments that when the HRT was operated at power, the effective average temperature in the core should be the mean of the inlet and outlet temperatures. During Run 13, temperature behavior was not inconsistent with "straight-through" flow. The power during this run was always low, however, and temperature spreads were very small so that the data were inconclusive. In the first part of Run 14, as the power was changed, the core temperatures appeared to behave as expected, i.e., the average was near the mean of the inlet and outlet. Later, during the last 4 hours of power operation, the situation in the core appeared to be different. Then the outlet temperature remained almost unchanged while the inlet temperature dropped each time the power was raised. This behavior, which is similar to that observed in HRE-1, would be expected if the core were perfectly mixed.

In considering the significance of changes in core temperature accompanying changes in power, it is necessary to take into account any concurrent changes in other system variables. Blanket temperature in particular is significant because of its strong effect on reactivity. Dump tank weight is also significant, and there is normally an interaction between core temperature and dump tank weight. If the condensate weight is held constant, as it usually was in Run 14, when the core temperature decreases because of loss of uranium, the dump tank weight decreases due to the increased density of solution in the core. This tends to minimize the temperature decrease. If, on the other hand, the dump tank weight were held constant by decreasing the amount of stored condensate, the effect would be to dilute the solution and to exaggerate the temperature decrease. Aside from the effects of changes in high-pressure system density, the inventory of solution or condensate in the low-pressure system is affected by the transfer of vapor between the pressurizers, which further complicates the analysis of the reactor behavior. This transfer also affects the effective core purge rate which influences the relation between the dump tank and core concentration.

Figure 15 depicts temperature behavior over a 15-hr period, during which the reactor operated at various powers up to 2 Mw. The flow in the core at this time appeared to be nearly "straight-through." The strongest support for this assertion is the behavior during the rapid reduction and increase in power around 1820 on March 31. During the five minutes or so in which most of the power change occurred, the blanket temperature changed only about 0.5°C. Therefore, assuming the reactivity remained constant over this time, the change in core average temperature should have been insignificant. The core average temperature may be represented by

$$T_{av} = T_{in} + \epsilon (T_{out} - T_{in}) .$$

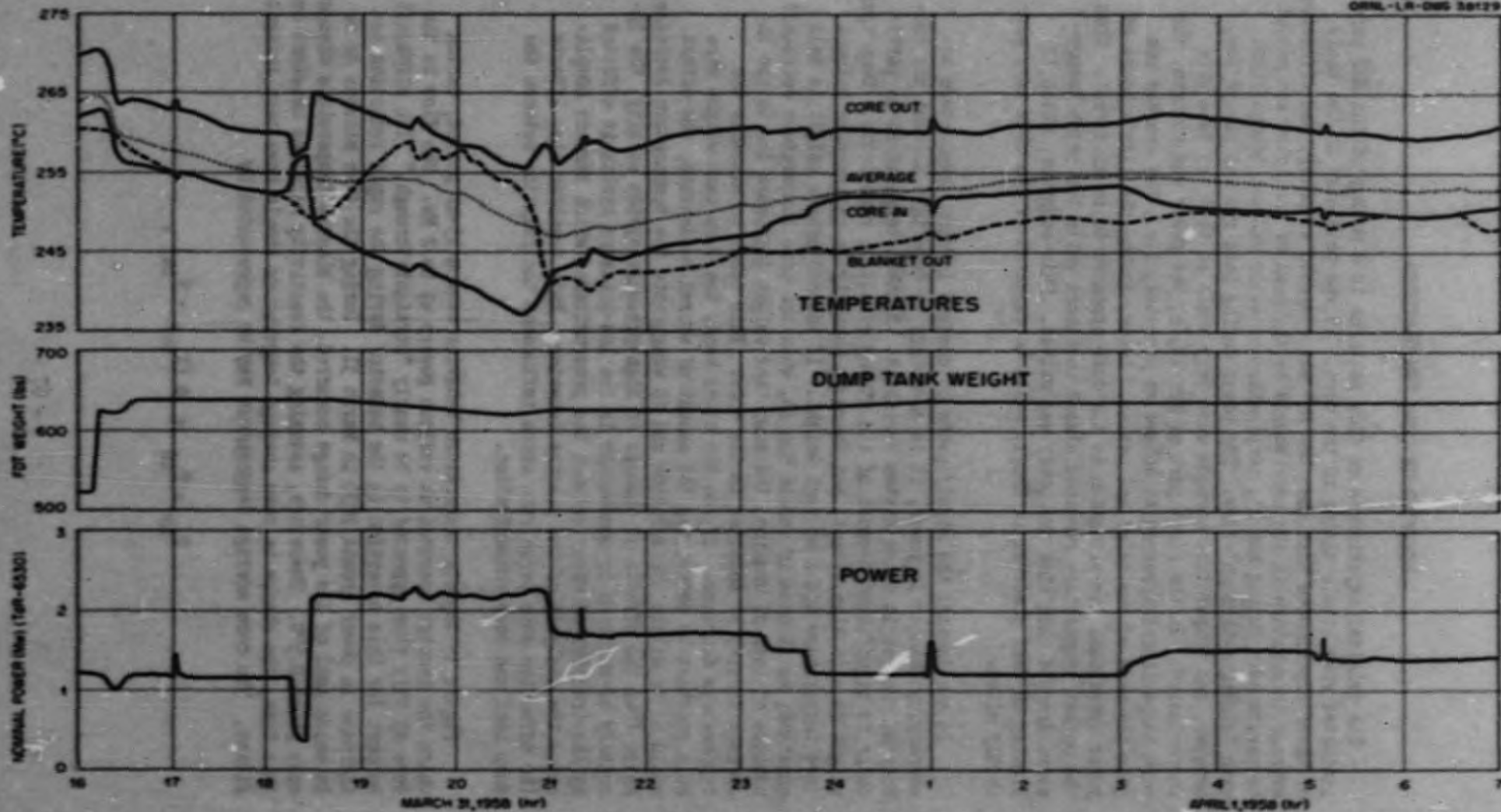


Fig. 15. System Temperatures During Power Changes on March 31 and April 1, 1958.

The value of θ depends upon the degree of mixing in the core. For "straight-through" flow, $\theta = 0.5$ and approaches 1.0 as mixing increases. The temperature behavior during the large changes in power gave a value of 0.55 for θ .

At the times of heat exchanger blowdown there were brief surges in power. Such occurrences appear on the plot at 1702 and 2120 on March 31, and at 0100 and 0509 on April 1. The temperature changes on these occasions were too small for calculation of reliable values of θ , but the appearance suggests "straight-through" flow.

Other prominent features of Fig. 15, not directly related to the question of flow patterns, are the temperature behavior following the dilution at 1610 and the seemingly aimless wandering of the blanket temperature. The abrupt drop in temperature at about 1615 resulted from the fuel feed pump pumping pure condensate for a period of over two minutes while the condensate tank was draining through the feed line back to the dump tanks. The more gentle decline during the next ninety minutes came as dilute solution from the dump tanks mixed into the core. Until 1930 steam withdrawal from the blanket heat exchanger was blocked off, so that blanket temperature fell or rose according to whether the heat production in that region was smaller or greater than heat losses. At 1930 the blanket steam withdrawal was put on automatic control to hold the blanket temperature steady relative to the core. At 2045 the blanket steam withdrawal was manually increased to bring the blanket temperature down and the core temperature up.

System temperatures and nominal power on the last day of power operation are shown in Fig. 16. The period after about 0700 is of particular interest in that responses to increase in power superficially indicated a perfectly mixed core in which the outlet and average temperatures are equal. However, as has been pointed out, a number of factors influence the core nuclear average temperature and any change in this quantity during a power change would affect the behavior of the core inlet and outlet temperatures. So far as could be determined, no variations occurred in the feed and purge pump rates or in the vapor transfer rate between the pressurizers during the time when the power was being raised. Small decreases, of the order of 5 to 10 lb, did occur in the fuel dump tank weight, but the effects of these changes on nuclear average temperature would be too slow to be detectable during the power changes. It was assumed that any reactivity changes due to power-dependent uranium deposition would be negligible over the short periods when the reactor power was actually changing. Thus, in evaluating the mixing behavior, it was assumed that the reactor nuclear average temperature was constant during the power changes. When corrections were applied to the core nuclear average temperature for the changes in blanket temperature, values of θ between 0.65 and 0.8 were obtained.

These values are significantly greater than the 0.55 observed during the rapid changes in power on March 31. The reason for the difference



Fig. 16. System Temperatures and Power on April 4, 1958

cannot be definitely established. It seems likely, however, that the thermal convection currents induced by the large temperature gradients at high power would cause more mixing, i.e., an increase in θ . Part of the explanation might be loss of reactivity while the power was being raised. On April 4, the power increases took place over much longer periods of time than on March 31. Depending on the mechanism, significant reactivity could conceivably have been lost during the power rises, and since this was not taken into account the calculated value of θ would be high.

The degree of mixing in the core also determines how the bulk mean temperature in the core vessel is related to the inlet and outlet temperatures. The small decreases in dump tank weight attending power increases were caused, at least in part, by the decreases in the mean temperature in the fuel heat exchanger and high-pressure piping. Any decrease in the bulk mean temperature in the core would cause a further decrease in dump tank weight. Thus, it is theoretically possible to evaluate the change in core bulk mean temperature from the dump tank weight change and the calculable changes in solution holding in the external parts of the core circulating loop. However, the dump tank weight record is not sufficiently precise to permit such an evaluation.

Many of the perturbations in temperature shown on Fig. 16 between midnight and 0700 are due to feed pump flow measurements in which the reactor was briefly diluted. Such measurements were made at 0417, 0458, 0547 and 0604. The feed pump stroke controller was changed at 0439, 0545 and 0604, all of this in efforts to restore the flow rate which had existed before the system pressure was raised the evening before. The change in system pressure was also indirectly responsible for other disturbances. The blanket pressurizer heaters were being controlled manually to minimize transfer of vapor between the two pressurizers, and because of the change in system pressure it was necessary to change the control settings. In seeking the proper settings, the blanket pressure was allowed to fall below the core and at 0643 there was an automatic pressurizer vent initiated by differential pressure. First the fuel pressurizer, then the blanket vented until the system pressure finally leveled off at 1675 psig, 125 psi lower than the starting point. Vapor transfer between the pressurizers, which had been from the core to the blanket, was reversed. The increase in the effective purge into the core is believed to be responsible for the gradual decline in temperature which began at about 0700. (The increase between 0600 and 0700 was caused by the increase in feed rate at 0604.)

APPENDIX

Determination of Core Uranium Concentration

There are three ways that the concentration of uranium in the core may be estimated: by a material balance using the book inventory, from the critical temperature and by sample analysis. Each of these methods has certain advantages and disadvantages as discussed below.

Material Balance.--The material balance method does not give the actual concentration because it assumes that all of the uranium in the reactor is in circulation, a situation that apparently has never existed in the HMF. The method requires a knowledge of fuel feed and letdown rates, which cannot be measured directly without a considerable upset of the system. The material balance calculation must either assume a steady-state ratio of core and dump tank concentrations, or the actual changes in dump tank weight, etc., must be approximated by finite step changes so that unsteady state equations can be applied. Despite these disadvantages, the concentration calculated from the material balance is useful in providing a basis of comparison for the concentrations estimated from the critical temperature or obtained from samples.

Critical Temperature.--Use of the critical temperature to estimate the concentration is convenient in that the only observations required are a few thermocouple readings. Because all the essential temperatures are recorded on strip charts, this method provides a way of estimating the concentration at any time past. The continuity of this indication is particularly useful in following transients. One error in this method is that use of the temperature-concentration calibration curve implies that the core is clean, as it was when the calibration was obtained. Any deposits of corrosion products in the core would introduce errors because of the increased critical concentrations required. Any non-circulating uranium in the active region of the reactor would reduce the concentration required in solution to keep the reactor critical. There is additional complication because the core is non-isothermal when the reactor is operating at power. However, the main uncertainty of this method is in the use of observed inlet and outlet temperatures to obtain the "nuclear average" temperature (see page 75).

Sample Analysis.--The third means of determining concentrations, by sample analysis, is the only way of getting at the actual concentration of dissolved uranium in the core. It suffers, unfortunately, from several difficulties. The sampling and analytical operations are cumbersome and time-consuming, so that relatively few samples can be taken during a run. Furthermore, the possibility of inaccuracies or error reduce confidence in the results of any single sample. Thus if the results are available from only a few samples, one erroneous result can drastically alter the apparent behavior of the solution and the reactor.

"Nuclear Average" Temperature

Definition.--By far the largest part of the effect of temperature on reactivity in the HRT is the effect, through changes in solution density, on neutron leakage probabilities. The material multiplication, k_{∞} , changes very little with temperature, being essentially a function only of the fuel concentration. Thus if the concentration (g U-235/kg D₂O) remains constant while the temperature patterns in the reactor are changed (by changing power, or flow patterns or the relation of core and blanket temperatures), the temperatures will shift so as to keep the overall neutron leakage probability constant and keep the reactor critical. The "nuclear average" temperature is a convenient concept. It may be defined for the non-isothermal reactor as the temperature of the reactor, if it were isothermal, at which the neutron leakage probability would be the same. In other words, if the non-isothermal reactor were made isothermal (by reducing the power and equalizing core and blanket temperatures) while keeping the concentration constant, it would come to the "nuclear average" temperature. Thus it is that the critical concentration is determined by the "nuclear average" temperature.

Relative Effect of Core and Blanket Temperatures.--The effect of the blanket temperature on reactivity is measured by $\partial k/\partial T_B$, the reactivity coefficient for uniform changes in blanket temperature only. A similar coefficient, $\partial k/\partial T_C$, is defined for the core. The ratio $\partial k/\partial T_B$ to $\partial k/\partial T_C$ determines the amount by which the core temperature must shift to compensate for changes in blanket temperature. The predicted ratio for the HRT with D₂O reflector in the vicinity of 280°C was $(-0.0011)/(-0.0018) = 0.61$. During Run 13 and the early part of Run 14, while the reactor was operating at a constant low power, it was observed that changes in blanket temperature were accompanied by a change in core temperature about 0.5 as great. Thus, on the basis of these results, the "nuclear average" temperature is a weighted average of core and blanket temperatures in which the core is weighted about twice as heavily as the blanket.

Averaging Blanket Temperatures.--The assignment of an average temperature to the blanket presents no serious problem, at least not in Runs 13 and 14, because the blanket itself was nearly isothermal. The outlet temperature was assumed to be adequately representative of the temperature throughout the region.

Averaging Core Temperatures.--There is considerable room for error in the definition of the core average temperature, however, since toward the end of Run 14 the inlet and outlet temperatures differed by as much as 44°C. Should the core be perfectly mixed, the entire body of solution would be at the outlet temperature and the average temperature would of necessity equal the outlet. If, on the other hand, the flow is "straight-through," when the reactor is at power the lower part of the core will be cooler than the upper. In this case, for the total leakage from the core to remain constant, the outlet temperature must rise so that increased leakage from the upper part of the core will balance the decreased

leakage from the lower parts. The effective average temperature in the core would then be near the mean of the inlet and outlet temperatures. (This crude qualitative analysis was confirmed by detailed calculation. (1)) Actual flow patterns lie somewhere between "straight-through" and perfect mixing, so the effective average core temperature should be between the mean and the outlet temperatures.

Approximation of "Nuclear Average" Temperature.--At the start of Run 14, because there were no data to the contrary, the convention was adopted of representing the core average temperature by the mean of the inlet and outlet temperatures. The blanket average temperature was assumed to equal the outlet temperature. With the core average temperature weighted twice as heavily as the blanket, the calculated "nuclear average" temperature became, with the above assumptions, equal to one-third the sum of the core inlet, core outlet and blanket outlet temperatures. This is the quantity that is plotted in Figs. 7 and 8. This approximation to the actual "nuclear average" temperature was used throughout the run to estimate critical concentrations.

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(1) P. R. Easton et al., "ERF Quar. Prog. Rep. April 30 and July 31, 1958," ORNL-2561, p 25.