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SUBJECT: SUMMARY OF HRE-2 RUN 13
(INITIAL POWER OPERATION)

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MASTER**ABSTRACT**

The first power operation of the HRE-2, referred to as Run 13, occurred in February, 1958. In five days of power operation, the maximum sustained power was 1.5 Mw and the total power generated was 48 Mw-hr. During the first part of the power operation, the fuel solution contained very little internal recombination catalyst. This part of the run was characterized by high concentrations of radiolytic gas and significant loss of reactivity at very low powers. After an addition of copper and acid, reactivity losses were observed only at higher powers. Power operation was terminated after samples showed high nickel concentrations in the fuel solution, indicating a very high stainless steel corrosion rate. Subsequent subcritical operation ended because of power wiring insulation failures.

This report describes operation of the reactor system, its auxiliary systems and major components. It also includes analyses of the data on reactivity loss, chemical behavior of the fuel, and production and recombination of radiolytic gas.

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INTRODUCTION

The first power operation of the HRE-2, called Run 13, began early in February, 1958. The broad objective of the run was to bring the reactor to full power and demonstrate operability at design conditions.

The initial approach to power was to be in a series of steps. Information to be obtained during the rise to power included:

- (1) Calibration, by heat balances, of reactor power instrumentation;
- (2) Data on radiolytic decomposition and catalytic recombination of the fuel solution;
- (3) Samples for radiochemical studies of the buildup of fission products and heavy nuclides in the fuel;
- (4) Surveys to establish the adequacy of the reactor biological shielding.

After full power operation had been attained, certain experiments were planned to observe the nuclear behavior of the reactor. These included:

- (1) Response to rapid changes in power demand;
- (2) Effects of afterheat and thermal circulation with the fuel circulating pump off;
- (3) Response to rapid addition of reactivity by means of the negative temperature coefficient.

PRELIMINARY OPERATIONS

Upon completion of the high-temperature critical experiments (Run 12), a final leak check was performed on the reactor container, and the north shield wall was filled with barytes sand and water. Container leak rates were established at 2.5 liters/min and 4 to 4.3 liters/min at pressures of 7.5 psia and 30 psia, respectively.

The high-pressure systems were then filled with D₂O condensate, pressurized to 1750 psig, and heated to 280°C. This condition was attained about 0300 on February 7. During this period, a final check of all equipment and instruments was made. Some difficulty was encountered in the removal of samples from the sample stations, and circulation of D₂O at the above conditions was continued until it was demonstrated that samples could be successfully removed at the various samplers.

REACTOR OPERATION

Approach to Criticality

At 1207 on February 8, the pumping of concentrated fuel solution from the dump tanks to the high-pressure system was started. When fission chamber count rates indicated that the reactor was near critical, the concentration was stabilized by reducing the feed rate. Criticality was attained by reducing the temperature, which was being maintained by the package boiler. The boiler was cut back very little, so that nuclear power would rise only to a few kw. Criticality was attained at 1300 at 280°C reactor average temperature. As the power rose, the fission chambers were retracted into the boral sleeves on the instrument tubes, but the shielding provided by the sleeves proved inadequate and the count rate went offscale.* The log N recorder failed to function, so the current from the compensated ion chamber was switched to the micromicroammeter. Meanwhile, the fission chambers were retracted by hand to positions near the north wall to bring the count rates on scale. The nuclear power was held low by continuing to supply most of the heat losses with package boiler steam. By 1457, the trouble in the log N recorder had been corrected, and at this time the recorder was put into service.

Operation at Heat-Loss Power

With all nuclear instruments in service, the steam being supplied by the package boiler to the fuel heat exchanger was gradually shut off to allow the core power level to rise to balance the heat losses from the fuel high-pressure system. This power level was attained at 1506 and maintained until 1927.

During this period the feedwater level in the fuel heat exchanger, which was being maintained by the No. 1 feedwater pump, oscillated about 1% to either side of its mean value. This induced similar oscillations in the core power as the power rose each time feedwater was added. The average power level at this time was about 250 kw with the maximum peaks at about 450 kw.

After less than 2 hours of operation at 250 kw, the reactor temperature began to decrease as the first indication of a loss of reactivity. The temperature dropped about 5°C and then appeared to level off. Between 1850 and 1856 the fuel feed pump stroke control was increased from 50 to 55% to increase the concentrations in the high-pressure system and regain the lost temperature. This corresponded to an increase in feed rate from 4.0 to 5.25 lb/min. The core temperature began to rise almost immediately.

*The 12-in. long boral sleeves were so short that streaming in the open ends gave an excessively high flux at the chambers.

At 1927, it was decided to heat the blanket heat exchanger with steam from the fuel heat exchanger, and the package boiler was shut off. At the time this change was made, the blanket steam pressure was somewhat lower than that in the fuel system and a brief power surge to about 800 kw resulted. The power dropped rapidly from this peak and leveled out at about 330 kw. This power level was maintained until 2215, at which time the power was raised to approximately 500 kw by starting some reactor steam flowing to the turbine. During this operation, two brief power surges to nearly 1 Mw were caused by the operation of a steam relief valve. The power then held nearly constant until 0345 on February 10.

Operation at 500 kw

After the initial recovery in reactor temperature which occurred when the fuel feed rate was first increased, this temperature again began to decrease and the decline continued for several hours. A second attempt was made to recover some of the lost reactivity at 0518 on February 9, when the fuel feed rate was increased from 5.25 to 8.4 lb/min. However, no significant recovery was observed, and the temperature continued to fall.

Off-Gas Flow Failure.--During this period, an abnormal condition existed, the significance of which was not immediately recognized. Readings of a wet test meter on the off-gas stream from the charcoal beds are plotted in Fig. 1. These readings, which were recorded routinely on the building log, indicated there was no flow of off-gas between 2100 on February 8, and 0900 on February 9. Throughout this time, there was a continuous injection of about 920 cc/min of oxygen into the fuel high-pressure system. Dump tank pressure chart records show a decline from 15.8 to 15.2 psia between 1710 and 1735 on February 8. This probably signals the cessation of flow through the seal pots at the discharge of the charcoal beds.

Fuel Dump Tank Manipulations.--During the first hours of power operation, there was a steady transfer of about 0.07 lb/min of D_2O vapor from the fuel pressurizer to the blanket through the leaking rupture disc assembly between the two pressurizers. As the fuel D_2O decreased, the fuel dump tank weight was allowed to creep down until at 1230 on February 9, the indicated weight was only 160 lb.* About this time action was taken to adjust the rates of heat input to the two pressurizers to reverse the direction of transfer, and by 1400 a transfer of 0.17 lb/min of vapor from blanket to fuel was established.

One proposed explanation for the observed reactivity loss was that some uranium might be held up in an unavailable condition in the dump tanks as a result of poor mixing or caking out of a part of the solute.

*The volume of the evaporator legs on the dump tanks is 100 liters. Because of voids, the predicted holdup of solution in the legs under boiling conditions is only around 17 liters or 41 lb.

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Fig. 1. Off-Gas Flow During Early Part of Initial Power Run.

At 1230, although the reactor temperature was already lower than desired, the dump tanks were diluted by dropping some condensate from the condensate tank. (After this the dump tank weight was never allowed to go below 250 lb.) The addition of condensate to the dump tanks produced a further drop in reactor temperature consistent with the amount of dilution, indicating no recovery of reactivity by improvement of dump tank mixing.

In the period from 1855 to 2100 on February 9, several changes were made in the steam supply to the fuel dump tank evaporators in an effort to recover any uranium that might have caked out in the evaporator legs. The evaporator steam pressure was first reduced to 10 psig on both legs for about 30 minutes. It was then increased to 50 psig on one leg and cut off completely on the other. After 20 minutes this condition was reversed. The cooling water flow to the entrainment separator was off during this period. At 2059, the dump tank steam pressure and entrainment separator cooling water flow were restored to normal. There was no apparent gain in reactivity as a result of these actions, and it was concluded that no uranium had been caked out in the evaporators.

Fuel Pressurizer Manipulations.--There had been some previous experience with loss of uranium in the HRT Mockup which was attributed to hydrolytic precipitation in the connecting leg between the pressurizer and the circulating loop. It was found that this uranium could apparently be returned to the circulating stream by reducing the flow of hot condensate from the pressurizer to the main loop. On this basis, the HRT fuel pressurizer purge pump was shut off at 2148 on February 9, in order to minimize the flow of condensate from the pressurizer. It was intended that this pump be operated intermittently as required to maintain an operable pressurizer condensate level. However, from the time it was shut off until 0803 on February 10 (a period of 10 hours), the fuel pressurizer purge pump was not operated at all, while the condensate requirements of the pressurizer were supplied by the transfer of D_2O steam from the blanket pressurizer. Then for another 19 hours, until the pump was returned to continuous operation, the fuel pressurizer purge pump was operated an average of only 14 minutes each hour.

In addition to the reduced purge flow, there was an 8-hour period, from 0600 to 1404 on February 10, in which the liquid level in the pressurizer connecting leg was allowed to oscillate rapidly over about 50% of the range of the level element. It was felt that this rapid oscillation would help to restore to the circulating stream any uranium that might have been precipitated or otherwise deposited in the pressurizer connecting leg. There was, however, no apparent increase in reactivity or recovery of uranium as a result of the modified pressurizer operation.

Fuel Circulating Pump Purge Pump Failure.--At 0300 on February 10, a routine check disclosed that the fuel circulating pump purge pump was not pumping. A chart record of temperature ratios indicated the pump flow had stopped about 0130. The evidence appeared to indicate that the diaphragm was ruptured, so at 0348 the power was reduced to about 300 kw as the first step in shutting down. Subsequently, it was discovered that the trouble was a rupture in the pulsator in the drive unit. A replacement pulsator was installed, and the pump was put back in service at 0602. At about 0700, the reactor power was restored to 500 kw. Aside from the deliberate power reduction, there was no apparent effect of the pump stoppage on the reactor behavior.

Bubble Experiments.--Another theory which was advanced to explain the loss of reactivity was that a significant fraction of the core volume was occupied by voids. Up to 1225 on February 10, the core pressure had been held at 1700 psig, and from the catalytic recombiner temperatures it appeared that no radiolytic gas bubbles were being let down. It was felt, however, that tiny gas bubbles might be circulating around the high-pressure loop without being taken out by the gas separator. To test this theory, it was decided to reduce the reactor pressure, a procedure which would cause the hypothetical bubbles to grow, resulting in more loss of reactivity.

At 1225 on February 10, the pressure reduction was started. The reactor temperature was about 250°C and the reactor power level was held nearly constant at 500 kw. The D₂O steam flow rate through the fuel low-pressure recombiner catalyst bed was also held constant. The system pressure was first decreased to 1600 psig and held for about one-half hour before additional reductions were made. This pressure decrease was accompanied by an increase of about 30°C in the temperature of the low-pressure recombiner catalyst bed and a noticeable increase in dump tank pressure.

From 1339 to 1520, the pressure was reduced from 1600 psig to 1100 psig. When the system pressure reached 1500 psig, there was a sharp rise on the recombiner temperature which ultimately reached 315°C. (The recombiner temperature for 1700 psig operation had been about 140°C.) After this, until the pressure was raised, the recombiner temperature closely followed any variations in reactor power level.

The reduction in core pressure, with the attendant indications of a commencement of bubble letdown, caused no rapid decrease in reactivity. There was, however, a gradual decline in reactor temperature of about 7°C during the 6-hour period of reduced-pressure operation.

Starting at 1851 on February 10, the high-pressure system pressure was raised from 1100 psig toward 1500 psig with the latter figure being reached at 1925. The pressure increase was accompanied by a drop in the temperature of the low-pressure recombiner catalyst bed and a gradual increase in reactor temperature from 244 to 251°C. On the basis of this test, it was concluded that there was no significant reactivity effect due to the volume occupied by radiolytic gas bubbles.

Operation at Less Than Heat-Loss Power

Beginning about 2330 on February 10, the reactor power level was reduced from 500 kw to about 50 kw, the new power level being reached shortly after midnight. For this operation, the package boiler was used to make up the difference between system heat losses and nuclear power level. This reduction in power level was followed by a steady increase in reactor temperature from 249 to 270°C, indicating the recovery of a significant portion of the lost reactivity. After 4 hours following the reduction in power, there was no more significant gain in reactor temperature, although the reduced power level was maintained until 1420 on February 11.

It was decided about 0200 on February 11, that an excessive amount of condensate had been allowed to transfer from the blanket system to the fuel system through the connection between the pressurizers. Therefore, action was taken to reverse the direction of transfer to reduce the inventory of D₂O in the fuel condensate tank. D₂O steam was transferred to the blanket at a rate of 0.13 lb/min until midnight of February 11. At that time the pressurizers were balanced to stop the transfer.

Internal Recombination Experiment

On the afternoon of February 11, the power was raised to observe the effect on recombiner temperature, with the object the determination of the threshold power for bubble evolution. The power started up at 1420 and almost immediately the reactor temperature began to decline from an initial 272°C. The recombiner temperature at first rose slowly with increasing power level. By 1545 the power was 440 kw and the recombiner, which had started at 133°C, was up to 147°C. The reactor temperature in the meantime had reached 266°C. Then, as the power was raised further, there was a sharp upward break in recombiner temperature. At the maximum power of 670 kw, the recombiner temperature was 243°C, and was following closely the minor variations in power level. The break in recombiner temperature, which was taken to signal the advent of bubble formation, was not accompanied by any sudden drop in reactor temperature, although the general decline continued. There was an increase in dump tank pressure as the recombination superheat threw a heavier load on the condenser, forcing out part of the non-condensibles. When the power was reduced to around 300 kw at 1625, the recombiner quickly cooled to below 150°C. The reactor temperature continued down until about 1700, when it finally leveled off at 261°C. The lower final temperature was consistent with the increase in power level from 50 kw at the start of the experiment to 300 kw at the end.

At the conclusion of the above experiment, 63.8 grams of copper as CuSO₄ and a total of 423.4 grams of sulfate, the rest as H₂SO₄, was added to the fuel dump tanks, the addition being completed at 1810 on February 11. The effect on reactivity was immediate as the reactor average temperature rose from 261 to about 270°C in the first 60 minutes. The power level was raised, starting at 2020, toward a peak value of about 1500 kw which was reached at 2130. For about four hours thereafter, the reactor temperature continued to rise slowly. At 0130 on February 12, the reactor temperature reached a maximum of 277°C and then began to drop very slowly in spite of the fact that the reactor power level had by that time drifted down to about 1200 kw.

The 1500 kw power level was obtained initially by sending reactor steam to the air-cooled condenser and operating the condenser fan. However, the low outside ambient temperature and water leaks from the air-cooled condenser caused ice to form on the fan blades. This resulted in an unbalance in the fan, making it temporarily inoperative. Consequently, the steam flow to the air-cooled condenser was stopped at 0353 and was sent instead to the turbine along with service steam. During this change the power level was reduced to less than 500 kw, and the reactor temperature rose about 3°C to 277°C. The reactor power was raised to 1200 kw at 0420, and the temperature again began a steady decline.

Starting at 0552, the core pressure was raised from 1500 psig in steps until it reached 1700 psig at 0650. These pressure increases seemed to check the temperature decline temporarily but had no lasting effect. Finally, at 0741, the power level was reduced to about 850 kw and the temperature decrease seemed to stop at 265°C. However, there was no recovery and the temperature later began to decrease again at this power level. At 1452, the reactor power level was again reduced to 500 kw, and a gradual increase in reactor temperature followed.

Fuel Circulating Pump Experiments

On the evening of February 12, some experiments were performed to observe the effects of stopping the fuel circulating pump. At 1737, all deliberate steam withdrawal from the reactor was stopped, reducing the power level to just balance system heat losses, which were about 350 kw. Then, at 1740, the fuel circulating pump was stopped for about one-half minute. As expected, the nuclear power level dropped sharply when the pump was stopped and rose rapidly when the pump was restarted. The surges in power level and pressurizer level made it necessary to put the pump in service by jogging it for a few seconds at a time until circulation was established. The maximum power peaks during this test were about 500 kw.

The reactor power was again allowed to stabilize at heat-loss power level, about 350 kw. At 1916, another experiment in stopping the fuel circulating pump was performed. This time, the pump was left off for nine minutes and the power surges on restarting reached about 2500 kw. When the circulating pump was first restarted, a momentary upward surge in the fuel pressurizer level occurred which actuated a safety interlock to stop the fuel feed pump. This pump was off for about 5 minutes before it was noted, resulting in a loss of fuel level due to leakage through the fuel letdown valve. When the feed pump was restarted, the rapid recovery of level created a pressure unbalance between the fuel and blanket systems. This initiated a series of several automatic venting operations from both the fuel and blanket pressurizers in turn before the system stabilized. A loss of 500 psig in core pressure accompanied these venting operations.

At 2324, a third experiment was performed, this time at about 500 kw nuclear power, with essentially the same results as in the other two tests. However, the pressure behavior was much less violent as the feed pump operation was more carefully controlled.

The power part of Run 13 was terminated at 2343 on February 12, when the reactor was switched to "shutdown" to dilute the core. The decision to terminate power operation was made after confirmation of sample results which showed a very high nickel concentration (over 1000 ppm) in the fuel solution. This high nickel indicated an alarming amount of corrosion and raised questions as to the solubility of nickel and the solution stability.

Operation With Condensate

During the initial part of the high-pressure system dilution, the decline in reactor temperature was retarded by introducing package boiler steam to the

main heat exchangers to prevent accidental criticality by cooling the core solution more rapidly than the rate at which the critical temperature was being reduced by dilution. After 5 hours of dilution, the system temperature was allowed to drop to the highest value that could be maintained with building steam (170°C), and the pressure was reduced to 1000 psig, while dilution continued.

The above conditions were maintained until 1600 on February 16. During this period, a number of samples were withdrawn from the fuel dump tanks to check the inventory of U²³⁵. The samples indicated that essentially all of the uranium that had been present at the start of the run was in solution in the fuel dump tanks. These samples also verified the high nickel content that was observed in the fuel solution shortly after the start of power operation.

At 1830 on February 16, 5.65 moles of copper as CuSO₄ and 10.15 moles of excess H₂SO₄ were added to the fuel dump tanks. It had been calculated that this additional copper would be sufficient for complete internal recombination of radiolytic D₂ and O₂ at a core power level of 5 Mw. Shortly after this addition, the reactor operating conditions were changed to 1200 psig and 260°C, still with only condensate in the fuel high-pressure system.

Operation With Subcritical Concentration

Following the addition of copper and acid, a period of operation was planned with fuel in the high-pressure system, but in a sub-critical condition to observe the stability of this solution at temperature and pressure. Accordingly, the injection of fuel into the high-pressure system was started at 2137 on February 17. Conditions were then adjusted so that the reactor temperature was about 5°C above the critical temperature. This condition was maintained until 0136 on February 19. Several high-pressure fuel samples were taken during this period to observe the uranium behavior.

Difficulty was encountered throughout Run 13 with the operation of power equipment within the reactor cell. This was caused by repeated short-circuiting of electric wiring, presumably due to moisture in the conduit. During the sub-critical run, two of the four cell air coolers and the blanket circulating pump stopped operating and could not be restarted. Therefore, the high-pressure system was again diluted on February 19, and finally shut down completely on February 21 for replacement of the electric wiring in the reactor cell.

GRAPHS OF OPERATING CONDITIONS

To facilitate the analysis of Run 13, many of the operating conditions were plotted, side by side, for ready inspection. Figures 2 and 3 show the variables which were plotted. (The original graphs, used in the study, were on a scale of one inch per hour.) The significance of each variable is explained below.

Reactor and Core Temperatures

The core temperature shown in Fig. 3 is simply the average of the inlet and outlet temperatures. Assuming a "straight-through" type of flow in the core, this is a good average for use in computing weight of solution in the core. The core "nuclear average temperature" should also be near the average of the inlet and outlet temperatures when the flow is "straight-through" without much eddying and mixing. In establishing critical concentrations, it is necessary to take into account the effect of the blanket temperature as well as the core temperature. The coefficient of reactivity for changes in blanket temperature alone is close to one-half that for the core temperature alone; that is, if the blanket temperature is changed with the reactor remaining critical, the core average temperature will change in the opposite direction about half as much. The blanket temperature should be fairly uniform at any time, so the outlet is near the average. From the foregoing, it follows that an arithmetic average of core inlet, core outlet, and blanket outlet temperatures should give a reasonable "nuclear average temperature" for the reactor. This is the function plotted in Fig. 2.

Core Pressure

The core pressure as presented in Fig. 2 is transcribed directly from a continuous chart record. This parameter can be varied at will and is normally controlled within very narrow limits by adjustment of the pressurizer heater power. Perturbations in core pressure will result only in the case of major system disturbances. Only two such perturbations occurred during the power operation: one at 0200 on February 9, when the reactor was in shutdown for an extended period for measurement of feed rate, and one at 1900 on February 12, when the fuel circulating pump was restarted after a deliberate stoppage.

Fuel Dump Tank Weight

Fuel dump tank weight is significant because of its relation to fuel solution concentration. The dump tank weight is fixed by these factors:

1. Inventory of D_2O in fuel system.
2. Amount of D_2O in high-pressure system.
3. Amount of D_2O in condensate tank.

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Fig. 2. Basic Operating Conditions During Initial Power Run.

Fig. 3. Basic Operating Conditions During Initial Power Run.

Fuel system D_2O inventory varies because of transfer between pressurizers. The amount of D_2O in the high-pressure system is determined by the temperatures there. The other factor, the condensate weight, is controlled automatically at any desired setpoint up to 450 lb. Fuel concentration changes normally are made by adjusting the condensate weight setpoint.

If the condensate weight remains steady, any change in fuel-side inventory is reflected as a change in dump tank weight. This, along with decrease in core temperature, was responsible for the loss of dump tank weight during the first 20 hours of power operation. The recovery was accomplished by decreasing the amount of stored condensate and by varying the fuel system inventory through transfers from the blanket system.

Since feed and purge flow measurements from the condensate tank result in temporary decreases in condensate tank weight, these operations are reflected as temporary increases in dump tank weight. The peaks in the curve at 0200 and 1800 on February 9, 0100 on February 11, and 1200 and 1500 on February 12, illustrate this effect.

Feed and Purge Rates

The pumping rate of the fuel feed pump depends upon the high-pressure system pressure and the pump stroke setting on the pressure controller HIC-115B. However, other factors such as gas in the pump, under- or over-phasing, and check valve leakage affect the feed rates so that it must be measured periodically. The fuel feed rate may be calculated from heat balances around the fuel feed cooler, or it may be measured directly by switching to "shutdown" and observing the rate of decrease of the fuel condensate weight. Although the latter technique is more accurate, it produces perturbations in the power level, core and reactor temperatures, and dump tank weight. The disturbances in these curves at 0200 and 1800 on February 9, and 0100 on February 11, were caused by feed pump flow measurements. As may be seen from the curve, the feed rate remained fairly constant except for changes in pressure and pump stroke.

The fuel purge rate, as plotted in Fig. 2, is the net purge into the circulating stream, corrected for the D_2O transfer rate between the pressurizers. As shown by the curve, the purge flow was subject to much wider variations than the feed flow. This is partly because the effect of pressure on purge flow is much greater. It is also influenced by the fact that the pressurizer purge was not operated continuously and the circulating pump purge was out of service for some time. The method for measuring purge flows is the same as for feed flow. However, purge flow measurements from the condensate tank have a much smaller effect on the system operation because the only primary effect is a change in dump tank weight. Purge flow measurements were responsible for the dump tank weight disturbances at 1200 and 1500 on February 12.

Recombiner Temperature

The plot of recombiner temperature is based upon hourly readings of a thermocouple in a well imbedded in the platinized alumina catalyst bed.

Additional data were taken during periods of particular interest, as when bubbles of radiolytic gas were being deliberately let down from the high-pressure system.

The recombiner temperature gives a reasonably accurate indication of the concentration of deuterium in the vapor being boiled up from the dump tanks. Thus, when the boilup rate is constant, this temperature is an indication of the amount of gas coming from the high-pressure system. However, any reduction in boilup rate with a constant flow of deuterium, is also reflected as an increase in recombiner temperature. Such boilup rate reductions occur whenever the low-pressure system is sampled, and dump tank samples were responsible for all except three of the peaks in recombiner temperature. The peak at 1900 on February 9, occurred when boilup rate was deliberately reduced in an attempt to recover some of the lost reactivity. The peaks from 1400 to 1900 on February 10, and at 1600 on February 11, were associated with bubble let-down experiments.

Reactor Power

The plot of reactor power shown in Fig. 3 was taken from the log N neutron level recorder. Heat balance calibrations gave a conversion factor of 490 kw per log N unit. For convenience, a conversion factor of 500 kw per unit reading on the log N chart was used in preparing the plot.

Uranium and Nickel

These curves are related to the chemical behavior of the system and will be described more fully in later sections.

COMPONENT PERFORMANCE

A number of the major reactor components are developmental, and their individual performances were closely observed along with the overall reactor operation. It is expected that a comprehensive report on each of the components will be made when sufficient information is available. However, some of the preliminary observations made during Run 13 are presented below.

Valves

Except for the fuel letdown valve, none of the reactor valves indicated any leakage or other malfunction during Run 13. The fuel letdown valve that was in service for this run was installed on November 12, 1957. Leakage through this valve started shortly after the start of Run 13 and became progressively worse as the run proceeded.

Since the maximum leak rate was less than the normal letdown rate, the leakage presented no particular problem in normal operation. However, when the fuel high-pressure system is sampled, the normal letdown flow must be reduced by an amount equal to the flow rate through the sampler in order to maintain the high-pressure system liquid level. On several occasions, the combined sampler flow and letdown valve leakage exceeded the total fuel pump

rate resulting in a temporary loss of level. During the time that the reactor was operated with condensate at low temperature (February 13-17), these level losses resulted in momentary stoppages of the fuel circulating pump by a safety interlock. However, since the actual level drop was not great, there were no serious effects.

Diaphragm Pumps

Six high-pressure diaphragm pumps, with a total of eight remote pumping heads are available for use in the reactor operation. These include a feed pump, pressurizer purge pump, and circulating pump purge pump for the fuel and blanket systems. Each feed pump has two remote heads, both of which are normally used when the pump is in service.

Except for a short period when the pressurizer purge was stopped to observe its effect on the chemical behavior of the fuel solution, all of the fuel system pumps were operated continuously. In the blanket system, only the circulating pump purge pump was operated continuously, and the other two were used as required to maintain the desired liquid level in the high-pressure loop.

All of the pumps were checked for proper phasing in the intermediate drive lines and general operation before the start of the run. The pulsator failure in the drive unit of the fuel circulating pump purge pump (see page 5) was the only abnormal occurrence during Run 13. This pump was out of service for 5-1/2 hours on February 10 while a replacement pulsator was installed. The pulsator had accumulated 1100 hours of operation before the failure occurred.

Since these pumps may continue to pump fluid even if the diaphragm in the remote head has failed, a check was made of the four purge pumps on February 17 to demonstrate that the diaphragms were still intact. This was accomplished by adding excess D₂O to the intermediate drive lines. Any loss of this excess indicates a leak, either through the diaphragm or to the outside. Each of the pumps was observed for a minimum of 1 hour with no sign of leakage. Following the tests each pump was restored to normal service.

Leakage through the fuel feed pump diaphragms results in the appearance of radioactivity in the intermediate lines which are continuously monitored by radiation elements. Since there was no indication of activity, no further investigations were made on this pump. The diaphragms of the blanket feed pump heads were not checked.

Power Leads

A considerable amount of difficulty was encountered in this run with short-circuiting of the power leads to the cell air coolers and the circulating pumps.

The original wiring system consisted of stranded copper wire insulated with a double layer of Fiberglas-woven sleeving lightly impregnated with an oleo-resinous varnish. The wiring was routed to the equipment location in

sealed boxes containing glass-to-metal seals to which ceramic-beaded wires from the remote-disconnect boards were connected. On the control-room end, the conduits and wires were sealed in a box filled with 100% air-blown asphalt. However, small amounts of moisture apparently leaked into the conduit, causing the insulation to break down at the operating voltage, 460 v AC.

Prior to the start of Run 13, the conduits were purged with gas to dry them, and all of the leads were shown to be in satisfactory operating condition. However, at about 0700 on February 18, short-circuits developed in leads to two cell air coolers and the blanket circulating pump. On February 21, the system was completely shut down for replacement of all the power wiring to the pressurizer heaters, cell air coolers, and circulating pumps.

Circulating Pumps

Except for the trouble with the power leads to the blanket circulating pump, both pumps performed about as expected. During startups, when the high-pressure circulating systems were at low temperature, the motor of the fuel circulating pump operated near its maximum capability. Under these conditions, the motor drew 29 to 30 amps as opposed to a rated 24 amps at design conditions. This overload was reduced when the operating temperature was reached.

The heat removed from the motor ends of the pumps was checked periodically as an indication of excessive leakage past the thermal barrier. During Run 13 the heat removal rates for the fuel and blanket pumps, with the system at 250°C, were about 23 kw and 8 kw, respectively. These numbers were obtained with the normal purge flows of about 0.6 lb/min to each pump.

Cell Air Coolers

It is desirable to maintain the cell air temperature below 50°C to inhibit deterioration of the neoprene gaskets used in the service air line disconnects. Accordingly, the cell is equipped with four air-to-water heat exchangers through which the cell air is continuously circulated by electrical blowers. Thermocouples at the inlets of the radiator sections are used to monitor the air temperatures. With all four coolers in service, temperatures below 40°C could be easily maintained at these points. The temperatures were somewhat higher directly above the core vessel, in the vicinity of the pressurizer heaters. However, this was not objectionable since there are no organic materials in this region of the cell.

Samplers

During Run 13, there were 69 attempts to remove reactor samples at the east and west sample stations. (For a description of the sampling facilities, see ORNL CF-57-1-87.¹) These attempts included fuel and blanket high-pressure, dump tank, and fuel condensate samples. The 69 attempts resulted in 43 samples with volumes greater than 3 ml. Faulty transfer from the isolation chamber to the sample flask was responsible for most of the failures. In some cases, mostly low-pressure samples, it appeared that the isolation chamber was empty.

The sampling program for Run 13 may be roughly divided into five periods as shown by Table 1.

During the pre-critical operation with only D₂O in the high-pressure systems, three successive, successful samples were obtained as an indication of the reliability of the sampling mechanism. These samples merely confirmed previous experience with the samplers. With the start of power operation, most of the samples were withdrawn from the fuel high-pressure system to observe the solution behavior under these conditions.

In view of the severe loss, or apparent loss, of uranium from solution at power, it was necessary to check the U²³⁵ inventory after the termination of the power run. Additional information was also required on the amount of nickel in solution. During this period of operation with only D₂O in the high-pressure systems, 18 of the 28 sampling attempts involved the fuel dump tanks. The proportion of adequate samples was lowest in this period. During the operation with a subcritical concentration of uranium in the core, samples were taken from the fuel high-pressure system to determine whether the uranium remained in solution at conditions of zero power. Another large group of samples was taken from the fuel dump tanks after the reactor was shut down, following the sub-critical run.

The major difficulty encountered in sampling the reactor system was that of removing samples from the isolation chambers. A number of flaws were discovered and corrected in the course of the run which improved the efficiency of sample removal. The pneumatic piston which positions the sample flaskholder under the isolation chamber was found to have insufficient air pressure to adequately seal the flaskholder against the bottom of the isolation chamber. This permitted loss of the vacuum (which is required to pull the sample into the flask) in the sample flask before the isolation chamber valve was opened. Changing the piston loading pressure from 80 to 100 psig markedly improved this situation. It was also noted that the two parts of the sample flaskholder (which are remotely assembled in a manipulator cell at the analytical laboratory) were not being adequately tightened by the remote machine. The result of this failure was to prevent the sample needle from passing completely through the sample flask diaphragm, thereby impeding the sample withdrawal. After this, each flaskholder was carefully inspected and, in several cases, removed and hand-tightened before using. This procedure was permissible because of the relatively low level of radiation associated with a flaskholder containing a new flask. Subsequent improvements in the remote assembling machine eliminated the hand operations.

Although the above modifications substantially improved the efficiency of the sampling operation, absolute reliability was not achieved in this run.

TABLE 1

SAMPLING ATTEMPTS DURING RUN 13

Period of Operation	Nature of Reactor Operation	Total No. of Sample Attempts	No. of Samples >3 ml	Sampling Objective
Feb. 7,8	D ₂ O,* Pre-critical	3	3	Establish sampling procedure.
Feb. 8-12	Nuclear Power	19	13	Study solution behavior at nuclear power.
Feb. 13-17	D ₂ O*	28	12	Re-establish U ²³⁵ inventory.
Feb. 18,19	Sub-critical	6	4	Study solution behavior at zero power.
Feb. 19-23	Shutdown*	13	11	Re-establish U ²³⁵ inventory.

*No U²³⁵ in high-pressure system.

Iodine Bed

The performance of the iodine bed during Run 13 was inferred from iodine concentrations observed in fuel samples.² The bed apparently was effective, as samples showed from 1.0 to 8.6% of the iodine-131 which would have been in solution had there been no removal.

Pressurizers

Overpressure and surge volume for the high-pressure circulating systems are provided by two electrically heated pressurizers. The construction of the fuel system pressurizer is essentially that described by Michelson.³ However, the blanket system pressurizer has an additional pipe connection between the condensate header and the main inlet line, below the normal liquid level. As long as only D₂O is circulated in the blanket system, this connection can be used to provide a satisfactory liquid level in the boiler section without the use of the pressurizer purge pump. Provision is made for blanking off this line if other solutions are used in the blanket system.

The rupture disc which normally separates the vapor regions of the two pressurizers had failed prior to the start of Run 13. Although the disc was ruptured, the resistance to flow was high enough so that significant differential pressures sometimes developed (see page 8). Rather than replace the rupture disc assembly, it was decided to operate with the leaking rupture disc, controlling the gradual vapor transfer between the systems by adjusting the relative power inputs to the two pressurizers. This transfer was reduced essentially to zero during the latter part of the run when more effort was devoted to the problem.

Since the fuel pressurizer was considered as a possible factor in the reactivity loss during Run 13, a number of variations were employed in its operation. These variations have been described earlier. The blanket pressurizer was, for the most part, operated without any purge flow. However, when blanket high-pressure samples were taken, the blanket pressurizer purge pump was used to compensate for the liquid let down through the sampler.

There are a number of factors which affect the total power requirement of the two pressurizers. Some of these are: (1) system temperature, which determines the degree of preheat of the purge flow, (2) system pressure, (3) purge rate, and (4) cell ambient temperature. The relative power requirements of the two pressurizers are influenced by the rate of vapor transfer and the relative purge rates. The effects of these variables are shown in Table 2. The cell air temperatures listed in this table are averages for the entire cell. While the temperatures in the immediate vicinity of the pressurizers are, generally, 20 to 30°C higher than the averages, they exhibit the same kind of variation.

TABLE 2

PRESSURIZER HEATER ELECTRICAL INPUT

Time Period		High-Pressure System Conditions		Cell Air	D ₂ O Vapor Transfer		Pressurizer Purge Pump Operation		Pressurizer Heater Input	
From Date / Time	To Date / Time	Temp. (°C)	Pressure (psig)	Temp. (°C)	Direction	Rate (lb/min)	Fuel	Blanket	Fuel (kw)	Blanket (kw)
Feb. 7/0030	Feb. 9/0030	280	1700	44	F → B	0.07	Continuous	Off	12.6	12.6
Feb. 9/2305	Feb. 10/1230	260	1700	44	B → F	0.17	Part Time*	Off	7.7	17.5
Feb. 13/0100	Feb. 16/1300	170	1000	42	----	0	Continuous	Off	14.6	12.3
Feb. 17/0453	Feb. 19/0140	260	1700	49	----	0	Continuous	Off	11.6	12.4
Feb. 20/0100	Feb. 21/1700	100	1000	56 ⁺	----	0	Continuous	Off	13.0	10.3

* Fuel pressurizer purge pump operated intermittently to minimize spillover from boiler section.

⁺Two cell air coolers out of service.

AUXILIARY SYSTEMS OPERATIONS

The preceding sections deal primarily with the operation of the reactor system as such. However, the operation of the various auxiliary systems is closely related to the reactor performance. Some of these auxiliaries required significant amounts of attention to obtain the required performance while others continued to operate automatically, with only periodic checks. The operations associated with the principal auxiliary systems are described in the succeeding paragraphs.

Oxygen and Off-gas

Throughout Run 13 oxygen was continuously injected into the high-pressure fuel circulating system at a rate of about 1 liter/min (STP). No oxygen was injected into the D₂O blanket. The input rate was metered by a calibrated capillary flowmeter in the addition line. A rough check on the flowmeter was obtained from the rate of pressure drop at the cylinder stations used to supply the high-pressure oxygen.

The reactor off-gas flow from the charcoal beds was continuously monitored by two integrating, displacement type flowmeters (a dry gas meter and a wet test meter). Consumption of oxygen in the reactor was estimated by comparison of off-gas and addition rates. During the 15-hour period on February 8 and 9 when there was no off-gas flow, this comparison indicated a consumption of 880 ± 100 liters (STP) of oxygen in the reactor system. An additional 800 ± 100 liters (STP) was consumed during the remainder of the run. The total oxygen consumption is equivalent to a corrosion of 12.6 ± 1.5 lb of stainless steel.

Two of the three reactor charcoal beds (A and B) were operated in parallel during this run. No activity was observed in the off-gas, indicating that the breakthrough point for the long-lived fission product gases was not reached before the run was terminated.

Reactor Steam

Treatment of the HRT feedwater and steam systems by means of chemical additives was started with Run 13. Prior to this time, no chemical treatment had been employed. The chemicals used were hydrazine and a mixture of K₃PO₄ and K₂HPO₄. The purposes of the chemical treatment were: 1) to maintain a high pH which helps inhibit corrosion of carbon steel, 2) to prevent caustic embrittlement of the carbon steel, and 3) to scavenge dissolved oxygen for additional corrosion inhibition. The desired conditions in the feedwater were a pH of about 10.5 (a pH range from 9 to 11 was deemed acceptable), a phosphate concentration of about 60 ppm, and a hydrazine concentration of about 100 ppm.

During normal power operation, while feedwater was being continuously added to the heat exchangers, the chemical treatment was accomplished by injecting solutions of hydrazine and phosphate into the feedwater pump

suction line by means of two Milton-Roy proportioning pumps. However, during periods of sub-critical operation or operation at less than heat-loss power when heating steam instead of feedwater was being supplied, this method was not applicable. Under these conditions, periodic, batchwise additions of chemicals were made to each of the heat exchangers. During power operation, in order to prevent the buildup of excessive amounts of phosphates, a small amount of the heat exchanger water was periodically discharged to the waste pond. While the reactor was being heated externally, condensate formed in the heat exchangers was discharged regularly to maintain a low liquid level.

In addition to the chemical treatment, some gas was stripped from the heat exchanger feedwater in the deaerator. All water entering the surge tank, which supplies the feedwater pumps, is passed countercurrent to a stream of live steam to remove non-condensable, dissolved gases. The deaerator was operated with a small steam discharge to assure purging of the non-condensable gases.

Samples were taken periodically of the liquid in both heat exchanger shells, the feedwater from the deaerator surge tank, and steam from the fuel heat exchanger. These samples were variously analyzed for pH, phosphate, hydrazine, and dissolved oxygen. In addition samples of the non-condensable gas associated with the fuel heat exchanger steam were taken for oxygen and hydrogen analysis.

A relatively small total number of samples was obtained in this run and all the results, except pH of the heat exchanger shell water, scattered very widely. The pH averaged between 10 and 10.5 for the entire run. Although the hydrazine concentration in the heat exchanger shell water dropped to zero shortly after the start of power operation, the dissolved oxygen concentration remained satisfactorily low. Results of the gas analyses of steam samples were somewhat inclusive although significant quantities of both hydrogen and oxygen were found.

Much of the scatter in the analytical data may be attributed to the batchwise chemical additions that were used for much of this run, and to the batchwise blowdown scheme. However, the fact that this was the first attempt at routine feedwater treatment of the HRT system was also responsible for some of the imperfections.

Flange Leak Detector

The flange leak-detector system was operated throughout Run 13 without incident. No significant flange leaks were encountered in the course of the run. The combined leak rate from all of the high- and low-pressure flanges averaged 50 ml/day. This includes several periods of heating and cooling the high-pressure systems, during which the leak rate is usually 3 or 4 times as great as for steady operation.

Chemical Processing Plant

Since Run 13 was the first reactor power run, the chemical processing plant was left off-stream to help reduce the complexity of the operation.

Although some check-out operations were in progress in the chemical plant during the period under discussion, they were not associated with the reactor operation.

Auxiliary AC Power

The HRT turbine-generator is operated continuously, when the reactor is in operation, to serve as an emergency AC power supply in the event of a failure in the normal electrical supply. The turbine is normally operated with service steam, although reactor steam may be used for part or all of the supply.

On February 19 at 0100, an attempt was made to restart the blanket circulating pump which had stopped due to short-circuiting in the power leads (see page 9). This resulted in a current surge which tripped the building supply breaker, isolating the building from its outside power supply. The turbine picked up and carried the building electrical load for 2 hours. At that time the outside power was restored with no interruption in other operations.

Instrumentation

The entire reactor complex is very heavily instrumented, and a number of the instruments have control functions in addition to the recording of reactor data. The major instrument difficulty occurred with the nuclear instruments during the initial approach to criticality (see page 2).

Other difficulties which have not been previously mentioned are discussed below:

A steam pressure controller (PIC-552) failed to function properly when reactor steam was admitted to the turbine, and manual positioning of the control valves was substituted as a means of adjusting the reactor power level.

The control unit in the Foxboro liquid level controller for the blanket high-pressure system (LR-247) failed and was replaced by a spare unit without any interruption in the reactor operation.

Many minor adjustments were made in other instruments to improve the quality of the operation, but no other significant difficulties were encountered.

Other Auxiliary Systems

Several of the auxiliary systems that were operated throughout Run 13 received very little attention other than periodic checks to insure continued satisfactory operation. Although the operation was not completely satisfactory in all respects, the effects of the operation were of such a secondary nature that they are not described here. The items in this category are:

- Demineralized cooling water system
- Tower cooling water system
- Refrigeration system
- Instrument air system
- D-C power supply

MISCELLANEOUS

Reactor Cell Leakage

The reactor containment cell was operated at the design condition of 7.5 psia during this run. The rate of inleakage of air was sufficiently low, so that operation of the vacuum pump was not required during the run. (The vacuum pump was operated briefly on February 9 to obtain a sample of air from the reactor cell.) The apparent leak rate into the cell, based on measurements of cell temperature and pressure, was 1.8 liters/min (STP).

Two methods have been used for measuring the rate of air inleakage to the cell. The first, and most direct, is observation of the cell absolute pressure with appropriate corrections for temperature changes. An accurate determination by this method requires a period of observation of the order of several days. A more rapid means of measuring the cell leak rate is by measuring, with an inclined manometer, the difference between cell pressure and the pressure in a sealed tank within the cell. This device automatically compensates for temperature changes in the cell air. However, use of this instrument requires that it be open to the cell for periods of several hours. Since, at the time of Run 13, there were no automatic block valves to isolate the manometer from the cell in the event of an accident, the instrument was not used during this run. At the completion of the run, block valves were installed in the manometer lines so that it could be used for continuous monitoring of reactor container leakage in subsequent operations.

Liquid Leakage Into Reactor Cell

Both the fuel and blanket sumps in the reactor cell were emptied before the system was started up. However, within a very short time, enough liquid accumulated in both sumps to actuate the sump level alarms. On February 9, a misoperation while checking the water level in the reactor thermal shield permitted a quantity of water to be flushed through the thermal shield onto the cell floor. The sumps were then emptied although no measurements were made of the quantity of liquid removed. The accumulations from about 0200 on February 10 to 0600 on February 17 were 25 and 200 liters for the fuel and blanket sumps, respectively.

Chemical analyses of the liquid from the sumps indicated the presence of traces of uranium. However, since the activity of the material was very low, it was concluded that this was natural uranium remaining from earlier runs.

The origin of the water, other than that which overflowed from the thermal shield, is not known. However, the absence of chromate in the water indicates that it was not leakage from the demineralized water services in the cell. Possible sources include leakage from the cell air coolers which are supplied with potable water, and leaks of reactor or service steam or condensate. Since the rate of water accumulation was relatively low, no attempts were made to find and stop the leaks.

Reactor Cell Radiation Levels

Gamma radiation levels may be observed at seven points within the reactor cell. The remote sensing elements are, in general, located near the top of the cell in the vicinity of the equipment items listed in Table 3. (For exact locations, see ORNL Dwg. F-27939.) The only set of readings obtained during Run 13 was taken with the reactor power level at 0.58 Mw.

TABLE 3
GAMMA RADIATION LEVELS IN CELL

Monitor No.	Location	Radiation Level (r/hr)
RM-1	Fuel Circulating Pump	9100
RM-2	Blanket Circulating Pump	3600
RM-3	Top of Thermal Shield	16000
RM-4	Fuel Dump Tanks	10000
RM-5	Blanket Dump Tanks	500
RM-6	Fuel Feed Pump	17600
RM-7	Blanket Feed Pump	205

Process Radiation

Several gamma-sensitive radiation elements are located in shielded areas outside the main reactor cell to detect any leakage of radioactivity into the various service systems or into the cell air. Two monitors are placed at each location to reduce the possibility of spurious indications. The following points are monitored:

- 1) High-pressure oxygen supply line,
- 2) Fuel feed pump intermediate drive line,
- 3) Reactor cell air,
- 4) Recombiner condenser cooling water return,
- 5) Service steam condensate discharge,
- 6) Reactor steam and cooling water from cell,
- 7) Stack gas.

The minimum indication on these monitors is 10 mr/hr and, with the exception of the pair of units in the steam valve pit, which monitor the reactor steam and cooling water, no on-scale readings occurred. The indications of the steam and cooling water monitors were somewhat erratic due to the high ambient temperature in the valve pit, but they were roughly proportional to the core power level. Readings equivalent to about 100 mr/hr at 5 Mw were obtained. However, the activity in the cooling water was very low, and there was no detectable activity in the steam after it left the steam pit. It appeared, therefore, that the indications were due to direct radiation from the reactor cell.

Biological Shielding

Extensive radiation surveys were made by Health Physics personnel during power operation. The biological shielding was found to be generally quite adequate. The few spots where radiation levels were above average are described below.

The general background in the control levels outside the north wall, extrapolated to 5 Mw reactor power, was 0.5 to 1.0 mr/hr, gamma. At one point on the north wall there was a narrow beam (about 2 inches in diameter) of gamma radiation. The maximum dose rate observed at this point during Run 13 was 1.3 mr/hr of gamma.* Neutrons were not detectable.

The radiation level at the top of the water-filled tubes housing the nuclear instruments, when extrapolated to 5 Mw power, was equivalent to 5 mr/hr of gamma and no detectable neutrons.

Two narrow bands of radiation were found coming out of the top shield, directly above the reactor vessel. In the narrow zones radiation levels, extrapolated to 5 Mw reactor power, were 30 mr/hr of gamma, 0.5 mrep/hr of fast neutrons, and no detectable thermal neutrons. The locations of the radiation zones were found to correspond to joints in the lower layer of plugs. Since access to the top of the cell is not normally permitted during power operation, the radiation in these regions presented no problem.

The highest radiation levels outside the shield were encountered at the fuel sample station. The maximum readings in Run 13, observed during circulation of a fuel high-pressure sample, were 7.5 mr/hr of gamma and 2 mrep/hr of fast neutrons.**

*In later runs the level at this point reached 14 mr/hr when the reactor power was 5 Mw.

**The situation was later greatly improved by the addition of borated water to the sampler shield plug which, through oversight, had contained only lead shot during Run 13.

REACTIVITY LOSS DURING POWER OPERATION

Introduction

During the first operation of the HRT at power (February 8-13), a considerable loss of reactivity was experienced. This loss was evidenced by a reactor temperature much lower than that which would be predicted from the calculated uranium concentration in the high-pressure system and the relation between this concentration and critical temperature observed during the critical experiments.

After the termination of the run, when complete sample results were available, the loss of reactivity was analyzed more thoroughly than had been possible during the period of operation. The analysis of this aspect of the operation is described in the sections which follow.

Factors Affecting Reactivity in HRT.--The reactivity of the HRT normally is controlled by the concentration of U-235 in the fuel solution and by the reactor temperature.

Temperature affects reactivity almost entirely through the solution density, which affects neutron leakage probabilities. Voids in the reactor, by decreasing the average density, have a similar effect. Loss of reactivity may also be caused by neutron poisons, such as fission or corrosion products. These poisons may be dissolved or suspended, or they may be deposited on the surface of the core vessel or the inlet diffuser screens where they would effectively reduce reactivity. (If uranium were deposited in the same locations, reactivity would be increased.)

The reactor operator controls the fuel concentration by varying the dump tank weight and the relative purge and feed rates. The effects of these variables are illustrated by the following equation, which applies when the reactor is at steady state:

$$C_H = \frac{m}{M_H + (1 + \frac{p}{f}) M_L}$$

C_H is the uranium concentration in the high-pressure system, M_H and M_L are the weights of solution in the high- and low-pressure systems, respectively, and p and f are the purge and feed rates. The amount of uranium in circulation is represented by m .

Critical Experiments Results.--The relation between temperature and fuel concentration in the critical reactor was determined during the critical experiments. At this time the reactor was free of poisons and, at zero power, presumably free of voids. Figure 4 shows the experimental results. The circles represent results of gravimetric analyses of samples taken when the reactor was critical. The squares are the concentrations calculated from the equation in the preceding paragraph (converted to the units used by the analytical laboratory).

Fig.4. Critical Concentration of HRE-2 as a Function of Temperature.

Uranium and Sulfate Concentrations

Most of the reactivity loss encountered during the power run appears to be directly attributable to a loss of uranium from the fuel solution. Described below are the ways in which loss of uranium may be detected, along with the data which support the apparent loss.

The uranium concentration in the fuel high-pressure system can be obtained by three methods described in the paragraphs which follow.

Inventory Calculations.--If it is assumed that all of the uranium* charged into the reactor is in solution, either in the dump tanks or the high-pressure system, the high-pressure system concentration (at steady state) is given by the following relationship:

$$C_H = \frac{m}{M_H + M_L \left(1 + \frac{p}{f}\right)},$$

where:

- C_H = concentration in the high-pressure system (g/kg),
- m = weight of uranium* in the reactor system (g),
- M_H = weight of solution in the circulating part of the high-pressure system (kg),
- M_L = weight of solution in the dump tanks (kg),
- p = net fuel purge rate (kg/min), and
- f = fuel feed rate (kg/min).

This equation is strictly applicable only at equilibrium conditions where the factors involved have been constant for some time, or are varying very slowly. After a step change in any variable, the following equation applies:

$$C_H = C_0 e^{-t/\tau} + \frac{f m \tau}{M_L M_H} (1 - e^{-t/\tau}),$$

where:

$$\tau = \frac{M_L M_H}{f (M_L + M_H) + p M_L},$$

*The same procedure can be used for other materials such as SO_4 or Cu simply by using their inventory instead of the uranium inventory.

and

C_0 = high-pressure system concentration at the time a change is made (g/kg),

t = time after change (min).

Equations for the concentration during continuous changes in any of the variables are too cumbersome to work with. Therefore, the above equation was used in the analysis of transients by representing real changes in the variables by a series of small, frequent step changes.

Critical Concentration.--During the initial series of critical experiments, prior to Run 13, large samples were taken from the high-pressure circulating stream for gravimetric analysis of uranium concentration. These samples were taken with the reactor critical at very low power, with no nuclear poisons and no voids. The relationship between temperature and critical concentration resulting from these analyses is shown by the circles in Fig. 4. A curve through these points was used as a basis for determining the uranium concentration in the high-pressure system during the power run. It should be noted that, if a significant amount of reactivity is tied up either in voids or neutron poisons, the actual concentration in the critical reactor must be higher than predicted by this curve.

Figure 4 also shows the concentrations that were predicted by the inventory calculation for the conditions that existed at the time of sampling. It may be noted that the sample results were consistently about 6% below the predicted concentrations. This may be due to some unknown error in the inventory calculation, or, more likely, to a portion of the uranium being out of circulation. (Similar discrepancies between calculated and analytical concentration were seen in all of the preliminary runs of the HRT with depleted uranium. In the depleted uranium runs, the sample results consistently showed uranium concentrations five to fifteen percent lower than those calculated from inventory.)

Sample Results.--The third method of determining uranium concentration is by chemical analysis (by potentiometric titration) of samples taken during the operation. Small (approximately 5 ml) high-pressure samples were scheduled on a once-per-shift basis throughout the power run. However, not all of the sampling attempts were successful, so that the picture with respect to sample analyses is incomplete in some areas. The analytical technique used for these samples was not the same as that used during the critical experiments, so some error could exist from this source. However, as will be shown later, the two methods seem to be consistent.

Comparisons.--The ratio of the observed concentration to that predicted from the system inventory may be defined as the fraction of the uranium inventory that appears to be circulating. This ratio is presented graphically in Fig. 3. The solid curve is the ratio of the concentration from the criticality relationship (Fig. 4) to the calculated value, while the points are ratios of sample results to calculated concentrations. (As mentioned before, during the critical experiments the ratio was in every measurement about 0.94.)

This curve shows the extent of the apparent loss of uranium during the initial part of the power run. It also shows the partial recovery that seemed to result when the power level was reduced, as well as the loss that followed a subsequent increase in power. The second major recovery in the curve followed the addition of copper and excess sulfuric acid to the fuel solution. However, at no time does this curve indicate a recovery to the condition that existed at the start of the run.

As may be seen in Fig. 3, there is generally good agreement between the analytical results and the values from the criticality curve. For the first three samples, the analytical concentrations were somewhat higher. This would seem to indicate the presence of voids or poisons, but judging by later sample results, the difference is probably sampling and/or analytical error.

The good agreement between the sample results and the concentrations predicted from the criticality curve indicates either (a) there were no poisons or voids, and the analytical procedure used for these samples gives results comparable to the gravimetric analysis of the critical experiment samples, or (b) there were poisons and/or voids and the uranium analyses were giving answers that were too low. From other calculations it appears that fission product poisons were negligible, and corrosion products, if uniformly distributed throughout the fuel solution, were also negligible. Low pressure recombiner evidence indicates no bubbles being let down, so voids are unlikely. Therefore, it appears that the uranium analyses are correct, or at least directly comparable to the gravimetric results obtained during the critical experiments.

Sulfate Concentrations.--Comparison may also be made between the calculated and observed sulfate concentrations. Table 4 lists calculated and observed concentrations, along with the ratios of these quantities, for the eleven high-pressure samples that were taken while the reactor was at power. Although the first three samples say that essentially all of the sulfate was in circulation, later samples indicate a loss of sulfate. Table 4 lists the number of gram-moles of sulfate ion that were not accounted for in the samples. Shown for comparison are the numbers of gram-moles of uranium that appear to have been absent from the circulating stream. A rough equivalence may be observed in the two terms. (See page 46 for comments on significance.)

Concentrations During Subcritical Operation.--Following the period of power operation, the reactor was operated for four days with only D_2O in the high-pressure systems. Then, on February 17 and 18, the reactor was operated at $250^{\circ}C$ and 1200 psi with a uranium concentration in the core that was just subcritical. Four high-pressure samples were obtained during this period. The observed and calculated concentrations of uranium and sulfate for these samples are given in Table 5. Since the reactor was subcritical, no comparison can be made on the basis of critical concentration. However, these sample results indicate at least partial recovery of both uranium and sulfate. It should be noted that, just prior to the start of the subcritical run, about 1.5 kg of SO_4 , 541 grams as $CuSO_4$ and 978 grams as H_2SO_4 , was added to the system. This addition may have affected the apparent recovery.

TABLE 4

SULFATE CONCENTRATIONS DURING POWER OPERATION

Time	Date	SO ₄ Concentration (g/kg D ₂ O)		Ratio $\frac{\text{Obs.}}{\text{Calc.}}$	Non-Circulating Material (gm - moles)	
		Observed	Calculated		SO ₄	U
1745	Feb. 8	6.21	6.60	0.941	2.4	2.6
0010	Feb. 9	7.21	7.21	1.000	0.0	4.1
0730	Feb. 9	7.44	7.66	0.971	1.2	7.0
0400	Feb. 10	5.82	6.89	0.844	6.4	7.4
1415	Feb. 10	5.03	6.76	0.744	10.5	12.9
0500	Feb. 11	6.22	7.00	0.888	4.6	6.4
0958	Feb. 11	5.71	6.93	0.824	7.2	6.4
2312	Feb. 11	6.29	7.66	0.821	8.1	7.1
0610	Feb. 12	6.30	7.83	0.805	8.8	8.6
1300	Feb. 12	6.22	7.71	0.806	8.8	7.2
2235	Feb. 12	6.52	7.95	0.820	8.2	8.0

TABLE 5

URANIUM AND SULFATE CONCENTRATIONS DURING OPERATION
AT SUBCRITICAL CONCENTRATION

Time	Date	Uranium Concentration (g/kg D ₂ O)		U Ratio Obs./Calc.	Sulfate Concentration (g/kg D ₂ O)		SO ₄ Ratio Obs./Calc.	Non-circulating Material (gm - moles)	
		Obs.	Calc.		Obs.	Calc.		U	SO ₄
0244	Feb. 18	5.85	7.59	0.771	5.87	6.85	0.857	6.3	8.7
1500	Feb. 18	6.44	7.56	0.852	6.24	6.82	0.916	4.1	5.2
1820	Feb. 18	6.58	7.55	0.872	6.47	6.81	0.951	3.5	3.0
0130	Feb. 19	6.12	7.30	0.838	6.57	6.59	0.997	4.5	0.1

Voids

One way that power level may affect reactivity is through the formation of radiolytic gas bubbles in the core. It appears, however, that gas bubbles were not responsible for any significant part of the reactivity loss during Run 13. In the first place, the evidence is that no bubbles were let down throughout most of the run, including the period at the beginning when most of the reactivity loss was observed. In the second place, even if there had been no internal recombination and all the radiolytic gas had formed bubbles, the calculated effect would be only to reduce the reactor temperature about a degree centigrade.

Low-pressure recombiner temperatures provide a convenient (and the only) indication of whether or not bubbles are being let down. Gas solubilities being what they are, gases dissolved in the liquid letdown could produce, at most, a rise in the recombiner temperature of about 15°C. A small volume of bubbles, on the other hand, would contain sufficient gas to raise the temperature much higher. For example, with 8 lb/min boilup and the radiolytic gas equivalent to 1 Mw, the recombiner temperature would be about 350°C.

At the beginning of Run 13, the only copper in the fuel was that which had remained in the reactor from previous operations with light-water solutions. The copper concentration was less than 20 ppm, and it was expected that radiolytic gas bubbles would be formed in the core at powers above a few kw. Hourly observations of the recombiner temperatures indicated, however, that there was no bubble letdowns before 1230 on February 10, when the core pressure was reduced. According to the recombiner temperature, bubble letdown then continued until 2400 on February 10, when the power was reduced. Bubbles were let down again on the afternoon of February 11, in an experiment to determine the threshold power for bubble formation. Other than during these two periods, the recombiner temperature showed no sign of bubble letdown, only the small variation of recombiner temperature with power level which could be attributed to changes in concentration of the dissolved gases. The recombination data are interesting from the point of view of catalyst activity, as they indicated an activity higher than expected. With regard to reactivity effects, the important point is that there was no bubble letdown at all throughout most of the run, including the period in which the greatest loss of reactivity was observed.

Although it is fairly certain that no bubbles were being let down most of the time, it is informative to calculate the magnitude of the effect of gas bubbles. Using accepted values for G ,* it may be computed that above the threshold power for bubble evolution, each additional megawatt of nuclear power should increase the bubble production rate by 13 liters/min (measured at 280°C and 1700 psia). If the gas separator is effective (so there is no

* See discussion of radiolytic gas yields on page 42.

recirculation of bubbles), the volume fraction of bubbles in the core outlet stream is equal to the bubble production rate divided by the total flow rate. The flow rate of liquid out of the core is about 1600 liters/min, so an increase of 13 liters/min in bubble rate would mean an increase of about 0.0083 in the volume fraction of bubbles in the outlet stream. If the volume fraction of bubbles in the core is one-half that in the outlet stream, for each additional megawatt, the fractional decrease in average core density should be about 0.0042. For a fixed uranium concentration (g U/kg D₂O), the reactor temperature would have to decrease to compensate for the effect of the voids on neutron leakage. In the vicinity of 280°C, a decrease of only 1.2°C in the reactor average temperature would compensate for the effect of the bubbles from a power increase of one megawatt.

One other theory regarding gas bubbles and their effect on reactivity should be discussed. It has been shown above by consideration of maximum yields, that if the gas separator performs as expected, the volume of bubbles in the core during Run 13 could not have been great enough to account for more than a very small part of the reactivity loss. The only way bubbles could possibly have been present in the core to an important extent would have been if the gas separator had failed completely to separate the bubbles, so that they recirculated around the loop and back through the core. This would have resulted in a higher void fraction than could be produced in one pass. As a matter of fact, there was no sign of any bubble letdown at all during most of the run. This means that if bubbles were really recirculating, the gas separator must have actually discriminated against bubbles, taking from the bubble-laden stream a letdown of liquid containing no bubbles. This seems to be a rather unlikely performance.

In summary, study of gas evolution data makes it appear very unlikely that gas bubbles were directly responsible for the reactivity loss. The data do show, however, that during much of the run, the concentration of radiolytic gases was quite significant, often near the limits of solubility. The concentrations were power-dependent, thus admitting the possibility that they were in some way related to the reactivity loss, which was definitely dependent upon power level.

Fission and Corrosion Product Poisoning

Some calculations were made to estimate the effect of neutron poisons on the reactivity during Run 13. Results indicate that the poisoning from fission products and dissolved nickel contributed a negligible part of the observed reactivity loss. The amount of iron associated with the dissolved nickel is sufficient to account for a very considerable poisoning effect if it were located in the core. (The samples contained very little iron, however.)

Fission Products.--The concentration of fission products in the reactor depends upon many variables, several of which fluctuated widely throughout the run. In order to simplify the calculations it was decided to use average values for core temperature, nuclear power, and letdown rate. Table 6 summarizes the values used. This treatment is adequate since the purpose of the calculation was to obtain approximate rather than precise numbers. In the

TABLE 6
VALUES USED IN CALCULATION OF FISSION PRODUCT
POISONING DURING RUN 13

Reactor average temperature (°C)	265
Critical Concentration (g U/liter)	9.5
Time at Power (hr)	103
Total Power Production (Mw-hr)	47
Average Power (Mw)	0.45
Low-Pressure Solution Volume (liters)	125
High-Pressure Solution Volume (liters)	460
Fuel Feed Rate (lb/min)	8
Fuel Letdown Rate (lb/min)	10
Average Thermal Neutron Flux in High-Pressure Fuel (n/cm ² sec)	5.2×10^{12}

calculations, the fission products which contribute significantly to poisoning were divided into three groups: Xenon-135, Samarium-149, and a third group which lumps all the other poisons.

Xenon: The concentration of Xe-135 was calculated from equations describing the material balances in the reactor for I-135 and Xe-135. These equations used the simplifying assumption that no I-135 or Xe-135 returned from the low-pressure system to the core. Included in the equations were terms for production (both direct and by decay of a parent), removal in let-down, decay and, in the case of Xe-135, burnup.

Samarium: The Sm-149 concentration in the core was calculated by a method similar to that used for Xe-135. Because of the long (55 h) half life of its precursor, Promethium-149, the Sm-149 did not approach steady state during the run.

Other: The effect of the other fission products was approximated by a group with a yield of 1.44 atoms/fission and a 2200 m/s cross section of 40 b. Transformation of an atom in this group either by decay or neutron absorption was assumed to give another atom of equal cross section.

Poisoning Effect: The calculated loss of reactivity due to buildup of fission product poisons is shown in Fig. 5.

Corrosion Products.--There was a considerable amount of corrosion of stainless steel during the power run. As shown in Fig. 3, sample analyses showed that a total of about 500-600 g of nickel went into solution. This is also the amount of nickel which would be in solution if the observed difference in oxygen input and discharge were all consumed in corrosion of stainless steel. Samples showed only small amounts of the other stainless steel constituents. Presumably they were not in circulation but were deposited somewhere in the reactor.

The poisoning effect of the nickel can be established with certainty since it was in solution so that its distribution can be calculated. The location of the undissolved corrosion products cannot be established with certainty, however. Possibly a large fraction was in the sediment pot in the fuel feed line. More would be distributed throughout the reactor in scale. Some may have resided in the core in regions of lower velocity. An upper limit on the poisoning effect of the undissolved corrosion products can be had by assuming that, as they were formed, they accumulated in the core.

Figure 6 shows the calculated effect on reactivity of the corrosion products. The data in Fig. 3 were used to get the nickel inventory in solution. The other constituents were assumed to be located in the core in a region of average importance in amounts proportional to the composition of the 347 stainless steel used in the HRT.

Effect of Poisons on Critical Temperature.--The magnitude of the poisoning effect of fission and corrosion products can be conveniently seen when it is translated into the reduction in reactor temperature which would attend their buildup, the uranium concentration remaining constant. The

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Fig. 5. Effect of Fission Products on Reactivity.

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Fig.6. Effect of Corrosion Products on Reactivity.

effect on critical temperature is shown in Fig. 7. It appears that there were enough corrosion products in the reactor to produce a large temperature loss if they were collected in the core. However, for this to be the cause of the observed temperature losses, it would require that the poison be redistributed so that the poison effect increased with power.

Conclusions

As a result of their study of this aspect of Run 13, certain facts seem clear to the writers:

1. Neutron poisons and voids had little part in the loss of reactivity.
2. The loss of reactivity was a manifestation of a loss of uranium from circulation.
3. There was a definite correlation between reactor power and uranium loss.
4. The maximum fraction of uranium out of circulation was significantly greater than the fraction which had been missing during previous runs at zero power.
5. Addition of copper and excess acid materially improved the situation, permitting operation at higher power with less loss of uranium.
6. In subcritical operation after the power run, more uranium appeared to be in circulation than at power.

Although the factors in the uranium loss have not been positively identified, it seems likely that dissolved deuterium is in some way involved. The true explanation of the uranium loss, if it is discovered, will probably incorporate an explanation of the high corrosion rate and oxygen consumption at the beginning of power operation, since these events coincided with, and probably were, related to the maximum rate of uranium loss.

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Fig. 7. Effect of Poisons on Critical Temperature.

RADIOLYTIC DECOMPOSITION AND RECOMBINATION

Plans for Run 13 included measurements of the effectiveness of the copper internal recombination catalyst. As it was desired to fix a baseline in the absence of copper, no copper was charged into the reactor with the enriched fuel. Samples taken at the beginning of Run 13 showed around 10 grams of copper in solution, however, This is believed to have been residue from precritical tests with depleted uranium-copper-light water solutions.

Because the copper concentration was so low (0.0002-0.0005 m), it was expected that radiolytic gas concentrations would reach saturation and bubbles would form at low powers. When the power was raised, however, there was no sign of bubbles until February 10, when the overpressure was reduced. Bubbles were again formed on February 11, as the power was raised. After the addition of copper on the evening of February 11, no more bubbles were seen.

Information on decomposition and recombination is obtained in the HRT from the power, the liquid letdown rate, the dump tank boilup rate, and the recombiner temperature. From early tests of the recombiner, there is available a calibration of recombiner temperature vs. H_2 concentration in the vapor-gas stream to the recombiner. The boilup rate and the recombiner temperature thus give the rate of low-pressure recombination or letdown of deuterium from the core. The deuterium letdown rate divided by the liquid letdown rate gives the concentration in the high-pressure system, provided all the deuterium is in solution. The internal recombination rate is the difference between production and letdown rates. The catalytic rate constant for the solution is equal to the recombination rate divided by the concentration of dissolved deuterium. The solubility of the gases can be obtained from the concentration in the letdown at the verge of bubble evolution. At all powers above the threshold of bubble formation, the concentration of dissolved gases and the internal recombination rate are practically constant. Thus the low-pressure recombination increases at the same rate as production, and the rate of increase with power gives the decomposition yield or "G value."

The effective G value for the HRT was predicted before the beginning of Run 13 in the following manner: Estimates of total energy and energy absorbed in the high-pressure fuel solution per fission in the core are listed in Table 7. G values listed for each type of radiation are literature values for dilute solutions. The dissociation of water is the product of the energy absorption and the G value. The total effect in the fuel solution per fission is estimated to be the deposition of 182 Mev and the dissociation of 2.90×10^6 molecules of D_2O . The average G value, which must be applied to the core power is then $290/182 = 1.59$ molecules/100 ev.

In the February 11 experiment, as the power was raised the recombiner temperature increased slowly, then abruptly, indicating bubble evolution and letdown. Just at the bubble threshold the recombiner temperature and the

TABLE 7

PREDICTED ENERGY DISTRIBUTION AND GAS PRODUCTION IN HRE-2

Source of Energy	Absorbed Energy (Mev/fission)		G Value (molecules/100 ev)	Fuel Solution Dissociation (10 ⁴ molecules/fission)
	Total	In Fuel		
Fission Fragments	168	168	1.67	280
Neutrons (recoil protons)	5	3	1.5	5
Fission Gammas	5	2	0.4	1
Fission Product Decay Gammas	6	2	0.4	1
Capture Gammas	7	2	0.4	1
Capture Decay Gammas	2	0	0.4	0
Fission Product Decay Beta	7	5	0.4	2
Total	200	182		290

boilup rate indicated that 0.00586 mole D_2 /sec was being recombined. The neutron level instrument, calibrated by heat balances, indicated a total power at the bubble threshold of 0.402 Mw.* A short time later the total power was 0.598 Mw, and the recombination rate was 0.0355 mole D_2 /sec. Dividing the increase in gas letdown by the increase in total power gives a ratio of 1.45 molecules of D_2O produced in the core per 100 ev of total absorption. Not all of this energy was absorbed in the core, however. Using the estimate that 182/200 of the total energy absorption is in the fuel high-pressure system, the indicated G value is $1.45 \times 200/182 = 1.59$ molecules/100 ev in the core. The exact agreement with the predicted value may be somewhat fortuitous.

The conditions at the threshold of bubble evolution on February 11 were used to estimate gas solubilities. During the experiment the total pressure was 1500 psig and the core outlet temperature was 268°C, at which temperature the vapor pressure of D_2O is 780 psia. When bubbles were being formed the partial pressure of the gases must have amounted to the difference, $1515 - 780 = 735$ psi. At the verge of bubble evolution, the recombiner temperature indicated a concentration of D_2 in the letdown stream of 0.0768 m. Radiolytic oxygen concentration would be half the D_2 concentration. Excess O_2 , calculated from the O_2 injection rate and the liquid letdown rate, was 0.0089 m. For these gas concentrations in uranyl sulfate solutions, BMI solubility data would predict a combined partial pressure of oxygen and deuterium of 770 psi.⁴ This is only 5% greater than the 735 psi observed in the experiment. Inaccuracies in the pressure and temperature measurements could account for most of this difference. Thus there appears to be no significant difference between the solubilities observed in the experiment and those reported by BMI.

The solution rate constants for internal recombination were higher than expected during Run 13. Table 8 shows calculated and observed solution rate constants at a core temperature of 268°C. The copper concentrations are the averages of samples taken before and after the deliberate addition of copper. Concentrations in the three samples taken before the addition ranged from 0.00019 to 0.00047 molal. After the addition, samples showed from 0.0014 to 0.0020 molal copper. The calculated rate constants were computed from Battelle values of α (the solubility of deuterium) and the values for $K_s/\alpha \sqrt{[Cu]}$ observed by McDuffie⁵ in out-of-pile bomb experiments. Confidence in the results is limited because of the scatter in the copper analyses and the inaccuracies in obtaining the "observed" rate constants from boilup and letdown rates and recombiner temperatures. It does appear, however, that the contribution to the recombination from factors other than copper was of the same order of magnitude as that of the copper. The data also suggest a dependence of the rate constant on the power level.

*Total power includes heat produced in the blanket and elsewhere as well as in the core.

TABLE 8

SOLUTION RECOMBINATION RATE CONSTANTS
AT 268°C CORE TEMPERATURE

Date Observed	Feb. 11	Feb. 12	Feb. 12
Reactor Power (kw)	400	300-500	800-1300
Copper Conc. (molal)	0.0003	0.0017	0.0017
Calculated K_S (min^{-1})	0.04	0.24	0.24
Observed K_S (min^{-1})	0.11	0.24-0.38	0.37-0.54
Obs. K_S - Calc. K_S (min^{-1})	0.07	0-0.14	0.13-0.30

CHEMISTRY AND CORROSION

Reactor Chemical Data

The initial power run was also a "hot" shakedown for the complex operations involved in sampling the fuel solution and making remote quantitative micro-determinations of its composition. Unfortunately there resulted some delays and uncertainties due to breakdown of analytical equipment in the first part of the run and a complete, exact description of the chemical behavior of the reactor during power operation cannot be made. A summary of the analytical concentrations and book inventory data for the fuel solution in Run 13 is shown in Table 9. The determinations of copper, nickel, sulfate and excess acid are questionable, but the uranium analyses are thought to be reliable.

Solution Stability

As discussed on pages 30-33, the sample results indicate a definite loss of uranium in the early hours of the run, a loss which was not entirely recovered at any time during the run. The calculations also indicate loss of sulfate, but, because of uncertainties in the sulfate analyses, the loss may not have been real.

Corrosion

The average corrosion rate of type 347 stainless steel in the high-pressure fuel system during the first 18.5 hours of power operation, based on Book SO_4^{2-} and the Ni/SO_4 ratio of sample H-23, was 166 mils per year. The average corrosion rate during the remaining 88.5 hours of power operation, based on Book SO_4^{2-} and the increase in the solution inventory of Ni calculated from the average Ni/SO_4 ratio of 23 samples taken after power, was 11 mils per year. The overall average rate during the 107 hours of power operation was 38 mils per year. The corresponding total consumption of type 347 stainless steel is approximately 9 pounds.

The first period with an extreme corrosion rate is the time when there was no oxygen off-gas from the reactor, and oxygen consumption by the reactor during the entire operation is in good agreement with the corrosion estimated from Ni and SO_4 analyses.

Since the corrosion of a type 347 stainless steel coupon downstream of the fuel circulating pump was one-tenth of the average* steel corrosion in this run, it is apparent that more intensive attack in some parts of the system probably occurred.⁶

*Assuming uniform attack limited to the fuel high-pressure system.

TABLE 9
SUMMARY OF HRT RUN 13 SOLUTION DATA

Sample No., Date ('58), Time.	Volume, ml	Analytical Data: Chemical (Spectrographic) ⁽¹⁾									
		pH	Acid, mg SO ₄ ⁼ /ml	U ^{235.2} , mg/ml	Total, mg SO ₄ ⁼ /ml	Cu, μg/ml	Ni, μg/ml	Mn, μg/ml	Cr, μg/ml	Fe, μg/ml	Other, μg/ml
Initial Book Inventory: 6398 g U ^{235.2} , 3900 g SO ₄ ⁼ , no ⁽²⁾ Cu.											
L-10 2-7, 1900	5.7 ⁽³⁾	1.00	9.61	28.76	17.47	25	0				
H-16 2-8, 0920	~16 ⁽⁴⁾	1.10	5.76	10.60	6.55		0				
L-19 2-8, 1115	4.9	0.70	9.61	27.33	16.52						
H-21* 2-8, 1745	5.2	1.35	4.23	10.92	6.83		0				
H-22* 2-9, 0200	3.1	1.25		11.21	7.93	33	392			(<50)	
H-23* 2-9, 0730	3.5	1.45	3.94	10.42	8.18	17	781			(<25)	
H-33* 2-10, 0400	3.7	1.40		9.22	6.40		480				

TABLE 9 (Continued)

Sample No., Date ('58), Time	Volume, ml	Analytical Data: Chemical (Spectrographic) ⁽¹⁾									
		pH	Acid, mg SO ₄ ²⁻ /ml	U ^{235.2} , mg/ml	Total, mg SO ₄ ²⁻ /ml	Cu, μg/ml	Ni, μg/ml	Mn, μg/ml	Cr, μg/ml	Fe, μg/ml	Other, μg/ml
H-39* 2-10, 1415	2.3			6.59	5.53						
H-47* 2-10, 2242	1.1					(14.5)	(445)	(56.8)			(Al 3.3) (Ca 1.5) (Mg 0.4)
H-48* 2-11, 0500	4.9	1.40		9.82	6.84		812				
H-59* 2-11, 0958	>10	1.33	1.83	9.69	6.28	13	787				
Book Inventory Change (By Additions): Net 6398 g U ^{235.2} , 4323 g SO ₄ ²⁻ , 64 g Cu. 2-11, 1800											
H-65* 2-11, 2312	4.2	1.23	1.68	9.39	6.92		724				
H-70* 2-12, 0610	4.0	1.55	1.30	8.85	6.93		1026				
H-71* 2-12, 1300	2.7	1.33		9.39	6.84	(160)	1074 (875)	(165)		(63)	(Al 3.4) (Mg 3.8)
H-77* 2-12, 2255	>10	1.40	1.54	9.28	7.17	98	864				

TABLE 9 (Continued)

Sample No., Date ('58), Time	Volume, ml	Analytical Data: Chemical (Spectrographic) ⁽¹⁾									
		pH	Acid, mg SO ₄ ⁼ /ml	U ^{235.2} , mg/ml	Total, mg SO ₄ ⁼ /ml	Cu, μg/ml	Ni, μg/ml	Mn, μg/ml	Cr, μg/ml	Fe, μg/ml	Other, μg/ml
L-80 2-13, 0920	0.9					(840)	1586 (2600)	(520)			(Mg 12)
L-81 2-13, 1345	5.1	1.45	3.79	31.06	24.22	323	3104				
L-82 2-13, 1600	0.1						1996				
L-88 2-14, 0535	1.6			43.08	30.55		2932				
L-92 2-14, 2228	5.0	1.05	5.01		36.26		4420	0			
L-95 2-15, 0240	5.4	0.81	5.98	33.83	24.76	360	2750	470	37	20	
H-96 2-15, 0345	4.5	4.50		0	0		16.3				
L-97 2-15, 1442	0.1						3379				

TABLE 9 (Continued)

Sample No., Date ('58), Time	Volume, ml	Analytical Data: Chemical (Spectrographic) ⁽¹⁾									
		pH	Acid, mg SO ₄ ⁼ /ml	U ^{235.2} , mg/ml	Total, mg SO ₄ ⁼ /ml	Cu, μg/ml	Ni, μg/ml	Mn, μg/ml	Cr, μg/ml	Fe, μg/ml	Other, μg/ml
H-100 2-15, 2010	1.3	1.50		0	0		3				
L-101 2-15, 2248	5.0	0.80	2.83	41.14	34.57	698	3902	660	0		
L-106 2-16, 1154	5.0	1.15	3.36	25.06	15.66	198 (170)	2121 (2100)	365 (220)	(90)	(30)	(Ca < 5) (Mg 0.8)
Book Inventory Change (By Additions): Net 6398 g U ^{235.2} , 5841 g SO ₄ ⁼ , 423 g Cu. 2-16, 1800											
L-109 2-16, 2120	2.6	0.60	9.13	26.55	33.01	2262	2176	400	0		
L-115 2-17, 0310	3.8	0.91	5.28	23.89	18.62	1703 (920)	1456 (1740)	280 (160)	0 (128)		(Ca < 5) (Mg < 2)
L-117 2-17, 0525	5.3	0.80		18.82	19.43		1587	300	0		
L-119 2-17, 1141	5.0	1.2	6.24	19.01	36.88	2436 (820)	1955 (1740)	(144)	(208)		(Mg 1.8)

TABLE 9 (Continued)

Sample No., Date ('58), Time	Volume, ml	Analytical Data: Chemical (Spectrographic) ⁽¹⁾									
		pH	Acid, mg SO ₄ ⁼ /ml	U ^{235.2} , mg/ml	Total, mg SO ₄ ⁼ /ml	Cu, μg/ml	Ni, μg/ml	Mn, μg/ml	Cr, μg/ml	Fe, μg/ml	Other, μg/ml
H-126 2-18, 0244	13	1.1		6.43	6.46	1444 (190)	593 (53)	(53)	(48)		(Mg 1.2)
H-134 2-18, 1500	2.3	1.6		7.08	6.87						
H-138 2-18, 1820	9.0	1.35	2.69	7.24	7.12	309 (250)	557 (62)	(62)	(50)		(Mg 1.0)
H-144 2-19, 0130	~13	1.32	2.74	6.73	7.23	522 (280)	594 (640)	(62)	(29)		(Mg 1.2)
L-150 2-19, 2234	5.1	0.75	10.33	34.04	30.99	2072 (1700)	2851 (2600)	(194)	(146)		(Mg 2.1)
L-151 2-20, 0457	5.1	0.72	8.17	32.81	35.52	2266 (1460)	2732 (2500)	(220)	(250)		(Mg 0.98)
L-153 2-20, 1047	5.1	0.82	8.02	36.78	31.35	2551 (1900)	3800 (2700)	(260)	(150)	(34)	(Mg 0.7)
L-155 2-20, 2301	4.95	0.80	7.20	37.83	33.59	2541 (1700)	3457 (2700)	(290)	(194)	(33)	(Mg 0.9)

TABLE 9 (Continued)

Sample No., Date ('58), Time	Volume, ml	Analytical Data: Chemical (Spectrographic) ⁽¹⁾									
		pH	Acid, mg SO ₄ ⁼ /ml	U ^{235.2} , mg/ml	Total, mg SO ₄ ⁼ /ml	Cu, μg/ml	Ni, μg/ml	Mn, μg/ml	Cr, μg/ml	Fe, μg/ml	Other, μg/ml
L-156 2-21, 0506	5.1	0.72	8.65	35.60	33.70	2000 (1520)	3016 (2300)	(200)	(146)	(39)	(Mg 0.7)
L-162 2-21, 2230	5.1	0.8	7.97	30.08	29.63	2241 (1500)	1918 (2260)	(190)	(152)	(36)	(Mg 0.6)
L-165 2-22, 0942	5.0	0.85	5.81	21.96	21.12	1690 (800)	1755 (1840)	(130)	(78)		(Mg 1.0)
L-168 2-22, 1833	4.7	1.20	3.84	9.76	8.63	647 (390)	636 (860)	(54)	(60)		(Mg 0.8)
L-169 2-23, 0115	5.0	1.10	4.80	6.07	8.67	628 (296)	675 (800)	(70)	(30)		(Mg 2.1)
L-172 2-23, 0955	5.3	1.15	3.55	12.18	11.23	823 (470)	1062 (1240)	(116)	(58)		(Mg 0.9)

(1) In most samples analyzed spectrographically, the following elements were sought but not found:

Ag, B, Ba, Cd, Co, Na, Nb, Pb, Si, Sn, Ti, V, Zn, and Zr.

(2) Solution Inventory based on average Cu/SO₄ ratio for samples 10, 22, 23, 59 is 9.5 g Cu.

(3) Suspended Solids 5.2 mg total.

(4) Suspended Solids 2.82 mg total. } All others reported as zero.

* Identifies samples during nuclear power operation.

Steam and Water System Chemistry

The fuel and blanket steam generators were operated with coordinated pH-phosphate control and with intermittent blowdown during the Run 13 power run. Chemicals were added to maintain a pH of 9 to 11 (preferably around 10.50) and a phosphate level of about 60 ppm, this treatment intended to prevent caustic embrittlement of the carbon steel parts. A hydrazine concentration of about 100 ppm was maintained in the feed to scavenge dissolved oxygen and thus reduce corrosion of the carbon steel.

As soon as the reactor went to power, the hydrazine concentration of the heat exchanger water dropped to zero and remained there for the remainder of the run. Although a residual level of hydrazine in the shell-side water was not realized, there is fairly good evidence that it was doing well as an oxygen scavenger. (In addition to being consumed in scavenging oxygen, it probably was undergoing rapid decomposition due to heat and radiation.) Dissolved oxygen analyses showed 0.09 ppm on a sample of fuel heat exchanger shell water and 0.007 ppm on a sample of blanket heat exchanger shell water. Four samples of the fuel heat exchanger steam averaged 2.62 ppm dissolved oxygen. These results show that dissolved oxygen was fairly low in the shell water of the heat exchangers and about what was expected in the steam generated.

Gas bubbles from the fuel heat exchanger steam were analyzed. Results from the two samples taken gave an average of 4.62% hydrogen, 15.89% oxygen, and one sample showed 78.90% inert gas (N_2). These determinations were questionable because of the probable leakage of air into the sample bottles.

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BIBLIOGRAPHY

1. B. A. Hannaford, HRT Sampler Development, ORNL CF-57-1-87 (Jan. 22, 1957).
2. S. Peterson, Behavior of Iodine in the HRT, ORNL CF-58-3-75 (March 18, 1958).
3. C. Michelson, HRT Modified Pressurizer Design, ORNL CF-56-6-165 (May 25, 1956).
4. E. F. Stephan et al., The Solubilities of Gases in Water and in Aqueous Uranyl Salt Solutions at Elevated Temperatures and Pressures, BMI-1067 (Jan. 23, 1956).
5. H. F. McDuffie and H. H. Stone, HRP Quart. Prog. Rep. Jan. 31, 1958, ORNL-2493, p 182.
6. 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).

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