ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60440

A SUMMARY OF THE OPERATION OF THE ANL-2
PRESSURIZED WATER LOOP AT THE MTR

by

C. C. Crothers

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A SUMMARY OF THE OPERATION OF THE ANL-2 PRESSURIZED WATER LOOP AT THE MTR

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C. C. Crothers

I. ABSTRACT

This report is a summary of the operation of the ANL-2 pressurized water loop at the MTR for the period beginning January 1, 1954, and ending July 1, 1962. Special attention has been given to operational problems and to the modifications which were made to eliminate these problems. During this period of operation, twelve successful experimental tests were completed. Each test is described briefly. The operation of the loop was essentially continuous during the entire period.

On July 1, 1962, the responsibility for the operation of the ANL-2 loop by Argonne National Laboratory was terminated. Operation of the facility was assumed by the Reactor Physics and Engineering Branch of Phillips Petroleum Company Atomic Energy Division.

II. DESCRIPTION OF THE ANL-2 LOOP

The ANL-2 pressurized water loop (see Ref. 1) is located in the Materials Test Reactor (see Ref. 2) at the National Reactor Testing Station in Idaho. The function of this loop, as well as of other in-pile loops, is to contain specimens of reactor materials and components for testing the effects of neutron irradiation under controlled conditions of environment and heat transfer. The facility may be properly referred to as a pilot plant.

In its present design the loop, constructed entirely of 300 series stainless steel, operates at a maximum pressure of 105.5 kg/cm² (1500 psi) and at coolant temperatures up to 260°C. Figure 1 is a schematic diagram of the loop. A brief description of the major loop components is presented here.

In-pile Tube: The in-pile tube is mounted in horizontal beam hole HB-2 of the MTR, which extends through the graphite reflector and reactor tank to within 5 cm of the reactor core face at the horizontal centerline of the core (see Figure 2). The tube is 5.9 m in length and has an outside

106-5216

Figure 1. Schematic Diagram of ANL-2 Loop

106-7754

Figure 2. Orientation of the ANL-2 In-pile Tube with Respect to the MTR Core

diameter of 8.9 cm. With the exception of its hemispherical tip of 316 stainless steel, the in-pile tube is constructed of 347 stainless steel.

Cartridges containing test specimens to be irradiated are positioned at the reactor end of the in-pile tube. Coolant water flow is channeled through a rectangular tube, approximately 2.5 cm (1 in.) by 5 cm (2 in.) by 3.7 m (12 ft), located in the center of the in-pile tube, and out through the annular space on the outside of the rectangular tube. The maximum thermal neutron flux just inside the reactor end of the in-pile tube is 2.5×10^{14} nv. The subject of in-pile tube life is discussed in Sect. VI.

Out-of-pile Tube: The out-of-pile tube is similar to the in-pile tube but shorter in length. Control specimens mounted in this tube are subject to the same environment as those in the in-pile tube except for gamma and neutron fluxes. By comparing the control specimen to the irradiated specimen the effects due only to irradiation may be estimated more closely.

Pumps: There are three oil-cooled canned-rotor pumps in the system. The normal flow rate of 230 liters/min (60 gpm) is maintained with one pump operating and two on standby. In case of a pump failure an electronic relay system automatically places the second pump in service. If flow is not resumed after this action, the third pump is placed in service automatically. A more detailed presentation of this subject, including a discussion of expected pump life and probable causes of pump failures, is found in Sect. V.

Line Heaters: Under normal operating conditions the amount of heat dissipated through the external surfaces of the loop is greater than the heat supplied by gamma heating of the in-pile tube and by fission heat from the fueled specimen being irradiated. Gamma heating of the in-pile tube supplies approximately 20 kW heat to the system. The additional heat required to maintain the desired operating temperature is supplied by electrical heaters of the clamshell type. The heaters are clamped to an especially machined section of loop piping.

The heat input to the system is controlled by varying the amount of power supplied to the line heaters. This is accomplished with a temperature recorder-controller and a saturable reactor system. The total capacity of the line heaters is 60 kW. Figure 3 is a photograph showing the line heaters.

Pressurizer-Degasifier: System pressure is controlled by a pressurizer and degasifier unit whose total volume is 151 liters (40 gal). Heat input to the unit is provided by electrical heaters of the clamshell type mounted on circulating legs, that extend from the sides of the tank and re-enter at the base of the unit. This design, which provides for convection of water through heated portions of the legs, reduces thermal stresses in the unit.

NRTS-56 2898

Figure 3. Line Heaters Attached to Loop Piping

During periods of operation this system contains a liquid phase in equilibrium with a gaseous phase. The temperature of the liquid phase determines the pressure of the system. The desired pressure is maintained by varying the power input to the heaters. This is accomplished with a pressure recorder-controller and a saturable-reactor system. Chevrons located in the vapor phase in the upper part of the unit aid in degasifying the coolant water by distributing it over a large surface area, thus providing a large gas-to-water interface.

When necessary, a portion of the coolant water may be bypassed through the pressurizer-degasifier. The coolant water enters at the top of the unit and passes down over the chevrons. During this operation, the unit may be vented to remove noncondensable gases. The pressurizer-degasifier also functions as a surge tank, compensating for volume changes which occur during heating and cooling of the system.

Heat Exchanger: A water-to-air type of heat exchanger provides for emergency cooling. Air flow through the exchanger is controlled with an air-operated louvre system. The exchanger is either full on or off.

Purification System: The quality of coolant water is maintained with a bypass purification system, composed of a regenerative heat exchanger, three deoxygenating resin columns, three demineralizer columns, and two micrometallic filter cartridges. During periods of operation, one deoxygenating resin column and one demineralizer column are used in series.

The remaining columns are on standby. Figure 4 is a photograph showing the regenerative heat exchanger and the two micrometallic filters. Figure 5 is a photograph showing the ion exchanger columns.

NRTS-56 2890

Figure 4. Regenerative Heat Exchanger

Figure 5
Ion Exchanger Columns

NRTS-61 5459

Makeup System: A small, high-pressure, positive-displacement pump of 11-liters/hr (3-gph) capacity is used to inject makeup water into the system. The makeup system also includes a deoxygenating resin column, a demineralizer, and a degasifier. Plant demineralized water provides the influent to the degasifier. The makeup water may be bypassed through the deoxygenating column and the demineralizer before injection into the system. Figure 6 is a photograph showing the makeup tank (degasifier) and Figure 7 is a photograph of the makeup pump.

106-7757

Figure 6. Makeup Tank
(degasifier)

NRTS-63 7602

Figure 7. Positive-displacement
Makeup Pump

Process-water System: When the reactor is at power, coolant flow through the in-pile section of the loop must be continuously maintained at all times to prevent burnout of the in-pile tube or test specimen. During periods when normal coolant flow cannot be maintained, reactor process water may be circulated through the section of the loop including the in-pile and out-of-pile tubes. The pH of the process water is in the range of 5.0 and 6.0, and it is not possible to control such parameters as dissolved gas concentration and ion concentration to conform to a particular test specification. For tests requiring precise quality control of the coolant water, operation with process water is undesirable.

Instrumentation and Control: Extensive use of instrumentation has been made to monitor and control operating conditions and to provide for safe operation of the facility. A thorough analysis of all possible safety hazards is made for each test. The safety instrumentation of the loop is intimately connected to the reactor safety system, and any off-normal hazardous operating condition will automatically cause a reduction of reactor power. High temperature, high pressure, low pressure, and low flow are typical conditions which initiate reactor power reduction.

The methods used to control system temperature and pressure have already been described. Coolant-water flow is controlled with a hand-operated, 5-cm, flow-control valve. The flow is quite stable and only occasional adjustments of this valve are necessary during operating periods.

Thermocouples for monitoring system temperatures are mounted at all critical points in the loop. A six-point temperature recorder and a 24-point precision temperature indicator are used in conjunction with the thermocouples. The in-pile tube tip is instrumented with 24 thermocouples. With the aid of a 24-position selector switch the output of any one of these thermocouples may be read on a potentiometer. The thermocouple with highest reading is connected to a single-point temperature recorder and provides a continuous record of the temperature of the tube tip.

System pressure is continuously recorded with the pressure recorder-controller. In addition, system pressure and surge-tank pressure may be visually monitored at two pressure gages mounted on the main control panel.

Coolant-water flow is measured at two points in the system. At one the flow is measured with an orifice plate. The differential pressure across the orifice is converted to a low-pressure air signal with a differential pressure transmitter (dp cell). The low-pressure air signal is transmitted to an air-operated flow recorder at the main control panel. Safety considerations require that the coolant-water flow be measured at a second point in the system. The sensing device for measuring flow at this point is a modified venturi tube.

The activity level of coolant water is monitored with two small ion chambers mounted on the inlet and outlet piping to the in-pile tube. Signals from the ion chambers are amplified and continuously recorded at the control panel. The activity recorder is equipped with a high-radiation-level alarm.

III. CHRONOLOGICAL SUMMARY

The conceptual design of the ANL-2 pressurized water loop was completed by the Reactor Engineering Division of Argonne National Laboratory in 1951. The loop was prefabricated at the Laboratory during 1952.

Figure 8 shows part of the loop installation and control panel after it was temporarily assembled at the Laboratory. The component parts of the loop were shipped to the MTR and by late 1953 the installation of the loop in the MTR was completed.

105-387

Figure 8. Main Control Panel of ANL-2 Loop before Shipment to MTR

Operation of the facility was begun January 16, 1954. The first specimens to be irradiated in the loop were aluminum-cobalt-manganese foils. The foils were irradiated to determine the magnitude of the thermal neutron flux in the irradiation zone of the in-pile tube. This initial period of operation was also utilized to train operating personnel and familiarize them with the facility.

The first fueled test specimen was irradiated in the facility during the period beginning March 20, 1954 and ending June 9, 1954. During the following eight and one-half years, eleven additional experimental tests were successfully completed in the facility. Each of these tests is briefly described in Sect. VII.

Operating cycles for the MTR are three weeks in duration. This period includes a shutdown period of from two to five days for reactor

maintenance, and insertion and removal of experimental tests. For the period covered by this report, 146 MTR reactor cycles were completed for a total of 83,176 MWd.

The responsibility for the operation of the facility was given to the Metallurgy Division of Argonne National Laboratory in January 1959. The experimental fuel-testing program was continued with a special emphasis placed on nucleate boiling heat transfer. Four fueled specimens were tested. Two of these were aluminum-clad fuel plates of the ALPR type. The tests were designed to allow nucleate boiling to occur at the reactor end of the plates. Each of the following two tests consisted of an assembly of three aluminum tubes containing uranium oxide fuel pellets. These tests were also designed such that nucleate boiling occurred at the reactor end of the specimens.

A number of minor modifications have been completed on the ANL-2 loop since its installation in the MTR. These modifications are discussed in detail in Sect. IV. Early in 1962, an extensive modification of the loop Instrument Electrical system was completed with the object of reducing maintenance costs and simplifying operation.

The Reactor Physics and Engineering Branch of Phillips Petroleum Company Atomic Energy Division is currently responsible for the operation of the ANL-2 loop. Many years of economical operation may yet be expected from this facility.

IV. OPERATION OF THE LOOP

One of the prime considerations in the design of the ANL-2 loop was concerned with possible hazards resulting from radiation. Because of the possibility of the failure of a fueled test element and subsequent dispersion of fission products in the coolant water, the major loop components and associated piping are either shielded or housed in a cubicle. The cubicle walls are constructed of high-density concrete blocks and are 2 ft thick. Specifically, the cubicle houses the entire purification system, the main coolant pumps, the heat exchanger, the line heaters, the pressurizer, the filters and most of the valves. The valves are operated by means of reach rods extending through the cubicle wall to the main control panel. The piping to and from the in-pile tube is shielded. The area in front of the in-pile access flange is shielded with lead brick and paraffin blocks. Prior to the removal or insertion of a test specimen this shielding is removed. The coolant water normally contains small amounts of suspended solids present because of the corrosion of the loop surfaces. As these solids are carried through the irradiation zone of the in-pile tube they become radioactive. Over a period of time these radioactive solids build up and deposit

on the piping surfaces and low-flow areas of the system. The resultant radiation fields also are a source of radiation hazard and provide added necessity for the shielding of the loop components.

Safe, efficient operation of the facility is dependent upon maintaining the coolant water at the correct temperature, pressure, and rate of flow. Even a temporary loss of flow could lead to specimen or in-pile tube burn-out and the creation of a radiation hazard of consequence. The safety instrumentation of the loop is interconnected with the reactor safety system. If the limits assigned to coolant-water temperature, pressure, or flow are exceeded a reactor power reduction is automatically achieved. To attain efficient reactor operation the frequency of unscheduled power reductions must be minimized.

A failure-free power supply is an essential requirement for the operation of the pumps and safety instrumentation of the loop. Available commercial power cannot be considered to be failure-free because of occasional power dips and failures. To provide a source of failure-free power the original loop design included a steam-driven turbo-alternator. Although the unit remained in service during the first one and one-half years of operation it proved to be quite inefficient. Maintenance costs for the unit were abnormally high, and it required an excessive amount of personal attention. A failure-free power system for general use at the MTR, consisting of a battery-operated motor-generator set, was installed in April of 1955. With this system in service the need for the turbo-alternator was eliminated.

The main coolant heat exchanger is a necessary safety feature of the loop. If the pressure of the system cannot be maintained, coolant-water saturation may be approached, resulting in bulk water boiling. Under these conditions it is desirable to remove heat from the system as rapidly as possible. The heat exchanger provided for this purpose was a water-to-water type utilizing raw water as the cooling medium. After an operating period of one and one-half years this exchanger failed. Since the loop was at operating conditions when the failure occurred, a considerable safety hazard was presented. A later examination of the failed component revealed that the failure was due to stress corrosion resulting from a relatively high chloride-ion concentration in the raw water. Most of the cracks in the metal were transgranular and appeared to start in the weld zones. Prior to the failure the raw water side of the heat exchanger was cleaned with a solution of nitric acid. This method of cleaning may have contributed to the failure.

The entire exchanger was replaced with a water-to-air type heat-exchanger. Cooling air flow for this exchanger is provided by using the normal venting system of the MTR. The heat-removal capacity of this

heat exchanger is considerably less than the water-to-water type, but it has operated in a satisfactory trouble-free manner for the past seven and one-half years. It is still in service at the present time.

Numerous other operational difficulties occurred during the initial period of operation of the loop, such as leaks and gasket failures. To complete the spectrum of annoying problems, a paraffin fire occurred in the shielding in front of the in-pile tube. The first tests conducted in the loop were somewhat compromised by these problems, but as the facility was the first of its kind in the country, they may well have been expected.

An essential requirement for the success of an experimental test is reasonable control of the quality of the coolant water. Coolant water of relatively low purity may cause the corrosion of the test specimen to be accelerated. Any deviation from the desired test conditions may make the tests results more difficult to interpret. For maintaining high-purity coolant water a single ion-exchanger column containing mixed cation and anion resins was included in the purification system.

Operation with this single ion exchange column was not entirely satisfactory. During periods when the loop is operating, it is considered to be hazardous to enter the cubicle. During several of the tests the resins became exhausted or otherwise ceased to function properly. In general this resulted in exposing the specimen to low-purity water until such time as the column could be conveniently recharged.

Another problem associated with maintaining water purity is the control of dissolved gases in the coolant water. After the loop is filled with water most of these gases may be removed by circulating a portion of the coolant water through the pressurizer-degasifier unit. It proved to be difficult, however, to remove the last traces of oxygen from the coolant water by this procedure. At the operating temperatures required for most of the experiments, dissolved oxygen is corrosive to the stainless steel surfaces of the loop. During some of the test periods an oxygen scavenger, hydrazine, was injected into the coolant water to remove the dissolved oxygen. This method of removing dissolved oxygen from the coolant water cannot be used for tests involving aluminum alloys because hydrazine corrosively attacks aluminum.

After slightly over two years of operation, the purification system was modified. The single ion-exchanger column was replaced with a system containing six columns. Three of the columns contained mixed anion and cation resins, and three contained sulfite deoxygenating resin. For normal operation one deoxygenating column was used in series with a mixed-bed column. The remaining columns were for use as spares. The system was so designed that when one column became exhausted the spare column could be valved into the system in a few minutes.

The system has worked with reasonable success. A problem still exists with regard to maintenance of the system. The columns act as filters and accumulate radioactive particulate matter from the coolant water. The resultant radiation field complicates the procedure for removing and replacing the columns.

Several minor modifications have been made in the instrument-electrical system. The instrumentation system originally included a saturation instrument which integrated a pressure signal and a temperature signal from the system. The instrument was calibrated to actuate an alarm when coolant-water saturation was approached. Maintenance costs for the instrument were excessive, and it was replaced with separate pressure and temperature recorders. The latter monitoring system, which is in use at the present time, has functioned satisfactorily and requires less attention.

Changes in safety requirements were responsible for some modifications of the facility. The coolant flow was originally monitored with an orifice-flow-meter system. It was believed that a backup for this system was needed. A modified venturi tube and associated equipment were installed at another point in the system. Backups for monitoring temperature and pressure were also installed.

In December 1961, after nearly seven years of operation, an extensive modification of the facility was completed. The pump switching circuit was completely redesigned in order to reduce the number of electrical relays in use and simplify the system.

The safety circuits for the loop were similarly modified. The alarm panel was replaced with a "panalarm" unit. The previous systems had functioned in a satisfactory manner, but were complicated and somewhat confusing to maintenance and operations personnel.

Several differential-pressure indicators and pressure switches were mounted on the main control panel. These components were connected directly to the loop with high-pressure-tubing lines. The failure of one of these lines outside the cubicle would have presented a safety hazard. The components were replaced by a system of pressure transmitters located within the loop cubicle. Low-pressure air signals are transmitted from the pressure transmitters to the main control panel.

Valves located within the loop cubicle were adjusted from the control panel by means of reach rods extending from the valve handles at the panel through the cubicle wall. The area immediately behind the main control panel provided access for adjusting and repairing instruments. This area was congested by the transition of a large number of reach rods.

As a part of the modification, most of the valve handles were removed and mounted on the cubicle wall. Only the main flow-control valve and valves which require daily use are presently adjusted from the control panel. Figures 9 and 10 show the main control panel after the modifications were completed.

NRTS-62 3023

Figure 9. Main Control Panel after Modification in 1961

After operation for about six years, the loop was chemically decontaminated to remove radioactive corrosion products which had accumulated on the surfaces of the loop. The radiation hazard imposed by the deposition of this material had begun to limit the accessibility of various areas of the loop for necessary maintenance. Following thermal transients, quantities of this material were released and became suspended in the coolant water. Migration of the material was also observed during steady-state operation. Some of the corrosion products tended to plate out on test specimens being irradiated, with a resultant impairment of the test. The decontamination of the loop is covered in detail in Ref. 4. Table I shows radiation levels at various points in the loop before and after decontamination. From later data it has been estimated that this decontamination procedure should be repeated once every two years.

NRTS-62 3027

Figure 10. Alarm Panel after Modification in 1961

Table I

RADIATION LEVELS FOR VARIOUS LOCATIONS IN THE ANL-2 LOOP
BEFORE AND AFTER CHEMICAL DECONTAMINATION

Location	Beta-Gamma Level (mr/hr) before Decontamination	Beta-Gamma Level (mr/hr) after Decontamination	Flow
In-pile Tube Outlet	800	250	normal
In-pile Tube Inlet	700	100	normal
Auxiliary Line Heaters	280	100	normal
No. 1 Line Heaters	250	150	normal
No. 2 Line Heaters	235	130	
Pump Intake Manifold	-	200	normal
Pump Outlet Manifold	450	150	normal
Surge Tank (Bottom)	800	500	low
Pump Inlet Filter	380	380	low
Pump Outlet Manifold Filter	2700	3700	low
Particle Accumulator (before Heat Exchanger)	390	100	low
Particle Accumulator (before Line Heaters)	850	350	low

For the last six years the operation of the ANL-2 loop has been quite satisfactory. For one period of eight months, the loop operated continuously without a single incident, a record as compared with similar facilities.

V. MAIN COOLANT PUMPS

One of the major problems associated with the operation of the ANL-2 loop was the maintenance of the main coolant pumps. The Westinghouse Model 100-A canned-rotor pump that was used in the facility has been out of production for several years; as a result, most of the replacement pumps were of necessity reconditioned pumps. Under normal operating conditions a new pump may be expected to operate about 20,000 hr. If properly repaired a reconditioned pump should operate from 10,000 to 20,000 hr.

During the period covered by this report twelve pump failures occurred. Several failed through normal usage after exceeding the expected operating life.

Three failed because of operational errors. The electrical windings of the pumps are cooled with transformer oil. Figure 11 is a schematic diagram of the cooling-oil system. Each of these three pumps failed because they were operated for short periods with the isolation valves of the cooling-oil system closed. Examination revealed that the insulation on the windings had broken down due to excess heating.

106-7755

Figure 11. Schematic Diagram of Cooling-oil System for Main Coolant Pumps

Most of the pump failures that occurred short of the expected operating life of the pump are believed to have been due to improper repair. Because of normal wear, it is difficult to properly align and balance a pump. Examinations of these pumps revealed fractured or severely worn bearings.

Most of the failed pumps were repaired at Argonne National Laboratory and then shipped to the MTR for installation in the loop. On one

occasion a pump was installed in the loop and then found to be inoperable although all of the checks which could be made at the MTR indicated the pump was satisfactory. To avoid any repetition of this procedure, a small test loop was constructed at the Laboratory. Before each shipment the pumps were thoroughly tested under simulated operating conditions. After adopting this method of testing the pumps, noticeably fewer premature failures occurred.

VI. IN-PILE TUBES

The most critical component of the loop with reference to design considerations is the in-pile tube. The hemispherical tip of the in-pile tube is within 2 in. of the reactor core face and is exposed to relatively high neutron and gamma fluxes. Heat produced in the tube tip as a result of gamma radiation is equal to about 10 w/g of metal.

In designing the tube tip it was necessary to effect a compromise between a heavy wall thickness needed because of the required operating pressure and a light wall thickness needed to reduce thermal gradients through the metal resulting from gamma heating. An exhaustive stress analysis (see Ref. 3) was made of the tube-tip design to insure adequate conformance with the A.S.M.E. and A.S.A. codes. The stress analysis was confirmed by fabricating a mockup of the tube tip and subjecting it to pressures up to 352 kg/cm² (5000 psi). Stresses in the mockup were measured with strain gages.

A limiting factor in determining the maximum safe operating time for in-pile tubes is the damage to the metal caused by fast neutrons. Neutron radiation affects stainless steel by increasing the yield point and decreasing the ductility. During the early period of operation, data for determining magnitudes of these effects were lacking.

In a cooperative surveillance program, specimens of in-pile tube materials were irradiated and examined to determine the effect of radiation on the material. The maximum allowable exposure time for an in-pile tube is equivalent to the exposure time of the corresponding irradiated specimen examined even though the specimen may have retained its ductility. All specimens examined so far have retained their ductility.

Three in-pile tubes have been replaced and a fourth is still in use. The first tube was replaced after 178 full power days of operation and an accumulated fast-neutron dose of 3×10^{20} nvt. The tube tip was carefully examined. Samples of metal from the tube tip were found to be quite ductile. Brinell hardness values for these samples ranged from 107 to 112.

The second in-pile tube was replaced after 225 full power days of operation and an accumulated fast-neutron dose of 4.8×10^{20} nvt. Samples of metal from the tip of this in-pile tube were also found to be ductile. The average Brinell hardness for these samples was 216, compared with a pre-irradiation value of 170.

The operating time for the third in-pile tube was extended because of the availability of data from the surveillance program. It was removed after 750 full power days of operation and an accumulated fast-neutron dose of 2.0×10^{21} nvt. Samples taken from this tube tip were found to be quite ductile. Brinell hardness values for the samples ranged from 270 to 355, compared with pre-irradiation values of 170 to 174.

No appreciable corrosion attack was noted in any of the three tubes examined.

The availability of high-flux test positions in the Engineering Test Reactor (ETR) has made it possible to accumulate data for evaluating the safe operating life of in-pile tubes at a much faster rate. The in-pile tube presently in use in the ANL-2 facility has accumulated a fast neutron dose of 3.5×10^{21} nvt. The allowable dose of this tube will be in excess of 1×10^{22} nvt. In terms of operating time this exposure is equivalent to more than ten years.

VII. EXPERIMENTAL TESTS

Each of the experimental tests completed in the ANL-2 loop during the period covered by this report is described briefly here. Where possible, references are given which contain detailed information concerning test results and their significance. The code numbers used to identify each test are of significance only for reference purposes.

Specimens to be irradiated are mounted in a stainless steel cartridge. Because of the position of the cartridge in relation to the reactor core, a steep flux gradient exists across the length of the specimen (see Figure 2). Depending upon the length of the specimen, the difference in flux values at opposite ends of the specimen may exceed a factor of ten. Figure 12 is a plot of the unperturbed flow-neutron flux over the length of the irradiation zone.

Figure 12
Unperturbed Flux in Test
Zone of ANL-2 Loop

106-7753

MTRW-3

The first experimental test specimen to be irradiated in the ANL-2 loop was a welded prototype fuel-element assembly containing 12 w/o fully enriched uranium and 0.058 w/o boron in a Zircaloy-2 matrix, clad with Zircaloy-2. The assembly consisted of four fuel plates held together with end supports. Each plate was 18.4 cm (7.25 in.) in length by 4.763 cm (1.875 in.) in width by 0.318 cm (0.125 in.) thick.

The primary purpose of the experiment was to determine the effect of neutron irradiation on a fuel material containing a burnable poison. It expected that the burnup of the poison concurrent with the depletion of the

fuel would result in the element retaining a more nearly constant reactivity during the irradiation period.

Typical production methods were used in the manufacture of this fuel element assembly. The results obtained from the test were used to supplement the production control data used in the manufacture of this type of fuel plate assembly. No formal report covering this test was published. Experimentation similar to this test is continuing even at the present time in an effort to extend the operating life of reactor cores used in high-pressure water reactors.

Table II is an irradiation summary of the MTRW-3 test, which was carried out at a reactor power level of 30 MW. To simulate actual reactor operating conditions, the test was conducted at a pressure of 105.5 kg/cm² (1500 psi) and a bulk water temperature of 260°C (500°F). A dissolved-hydrogen concentration in the coolant water of 80 to 120 ppm was maintained by injecting hydrogen gas into the system at periodic intervals. This procedure assures a low dissolved-oxygen concentration in the coolant water and reduces the corrosion rate of loop components.

Table II

MTRW-3 IRRADIATION SUMMARY

Date	Exposure, MWd	Pressure, kg/cm ²	Temperature, °C	Thermal* Flux x 10 ⁻¹⁴
3-20-54 to 3-23-54	102	105.5	260	0.5
4-14-54 to 4-18-54	81	105.5	260	1.2
4-18-54 to 4-28-54	242	35.2	154	1.2
4-28-54 to 6-9-54	1077	105.5	260	1.2

*Unperturbed flux at the reactor end of the cartridge

The cartridge was inserted into the in-pile tube on March 20, 1954, preceded by a 15.3-cm spacer. Although the predicted heat output of the element was 40 kW, the actual heat output was 17 kW. The discrepancy was attributed to errors in the determination of neutron flux and to errors in the basic data used in the calculation of flux perturbation. After three days of operation the test was interrupted because of a weld failure in the purification system. At this time the cartridge was removed and re-inserted without the spacer. The test was removed on June 9, 1954, after 50 days of operation and a total exposure of 1502 MWd.

MTRW-2

Test MTRW-2 was conducted, at a reactor power level of 30 MW and a thermal neutron flux of 2×10^{14} nv, to determine the effect of neutron radiation on the corrosion resistance of a series of heat-treated alloys. The test cartridge contained six Zircaloy-2-clad specimens, 18 cm ($7\frac{1}{4}$ in.) in length by 4.8 cm ($1\frac{7}{8}$ in.) in width by 3.2 mm ($\frac{1}{8}$ in.) thick. Each plate

contained approximately 280 g of natural uranium. Table III is a tabulation of the composition and weight of each plate. A control cartridge was concurrently tested in the out-of-pile tube.

Table III

THE MTRW-2 CARTRIDGE

Plate No.	Alloy	Gross Weight (g)
1	U-3 w/o Nb	351
2	U-3 w/o Nb	353
3	U-6 w/o Nb	338
4	U-6 w/o Nb	341
5	U-1.5 w/o Nb-5 w/o Zr	342
6	U-1.5 w/o Nb-5 w/o Zr	357
		Total 2082

Numerous operational difficulties occurred during the test, so that its value was somewhat impaired. Table IV is an irradiation summary for the test. The cartridge received a total exposure of 1339 MWd during 44 days of equivalent full-power operation. The fuel plates were later examined, and a very little corrosion or other damage was noted. Reference 5 contains some additional comments regarding this test.

Table IV

MTRW-2 IRRADIATION SUMMARY

Date	Exposure, MWd	Pressure kg/cm ²	Temperature, °C
6-10-54 to 7-1-54	378	70	260
9-28-54 to 10-1-54*	101	70	260
4-10-55 to 5-22-55	860	7.0	38
Total 1339			

*Process water operation

MTRW-5

Test MTRW-5 was a part of an overall program to evaluate the use of aluminum alloys as construction materials for power reactors. At high temperature, pure aluminum has a high corrosion rate and a low structural strength.

The test cartridge contained six 2S aluminum specimens which were plated with nickel by the "Kanigen" dip process and by electrode deposition. Operating conditions for test were 70 kg/cm² (1000 psi), 260°C, and a coolant-water velocity of 760 cm/sec (25 ft/sec) past the specimens. Test conditions were maintained for 530 hr. The total operating time for the test was 700 hr. The total thermal-neutron exposure was 6×10^{20} nvt.

All of the samples lost their protective nickel coatings during the test. A detailed account of this test may be found in Ref. 6.

MTRW-6

Test MTRW-6 was similar to MTRW-5. The test cartridge contained six aluminum alloy plates. Plates 1 and 2 were an alloy of 2S aluminum and 0.5 w/o nickel (air cast). Plates 3 and 4 were an alloy of 2S aluminum and 1.0 w/o nickel (vacuum cast). Plates 5 and 6 were an alloy of 2S aluminum and 2.0 w/o nickel (vacuum cast).

The cartridge was irradiated intermittently between March 4, 1955 and September 3, 1955. Operating conditions for the test were 70 kg/cm² (1000 psi) and 200 to 260°C with degassed deionized water. The pH of the coolant water was maintained in the range from 6.8 to 7.2. The velocity of the coolant water past the specimens was 910 cm/sec (30 ft/sec). The results of both the MTRW-5 and the MTRW-6 tests showed that the rate of corrosion decreased as the neutron flux increased. The specimens received maximum thermal-neutron exposure of 1×10^{21} nvt.

A detailed description of the test and an evaluation of the results of the test is contained in Ref. 6.

MTRW-7

Test MTRW-7 was conducted to determine the effect of neutron irradiation on the corrosion of specimens of M-388 aluminum alloy. The M-388 alloy consists of 1.0 w/o nickel in 2S aluminum. The specimens were mounted in cartridges which were specially designed to permit the removal and replacement of individual specimens during the test, so that it was possible to compare specimens having various exposure times. An identical cartridge was exposed in the out-of-pile tube.

Because of operational difficulties, the desired operating temperature of 260°C was maintained for only 50 percent of the exposure time. Difficulties were also experienced in maintaining the correct water purity during a portion of the test. For short periods pH values as high as 8.5 were recorded. The results of the test were somewhat compromised by these difficulties. The decrease in the rate of corrosion with an increase in flux was not as pronounced as in previous tests. The duration of the test was three MTR reactor cycles. A detailed account of this test may be found in Ref. 6.

MTRW-9

Test MTRW-9 was similar to MTRW-7. The specimens were M-388 aluminum alloy. The operating period for the test included nine reactor cycles. Few operational difficulties were encountered, and reasonably good water purity and pH control were maintained. Operating conditions for the test were 105.5 kg/cm² (1500 psi) and 260°C. The coolant-water velocity past the specimens was about 730 cm/sec (24 ft/sec). Detailed discussions of the test results may be found in Ref. 6 and 7.

MTRW-8

This test was conducted to determine the dimensional stability of a prototype EBWR Core-1 fuel plate. The core material was 1.44 percent enriched U-5 w/o Zr-1.5 w/o Nb alloy roll-bonded to 0.508-mm-thick Zircaloy-2 cladding. The plate dimensions were 43 by 5 by 0.51 cm (17 by 2 by 0.20 in.).

The plate was irradiated intermittently from March 1, 1956 to March 15, 1961. The initial operating conditions for the test were 105.5 kg/cm² (1500 psi) and 242°C. After receiving a thermal neutron exposure of 5.8×10^{20} nvt maximum, the plate was removed and examined. No appreciable dimensional changes were noted. An account of this portion of the test may be found in Ref. 5.

For most of the test period, totaling 43 MTR cycles, the experiment was operated at the desired conditions with neutral high-purity water. The maximum heat flux was 16.8 W/cm² (700,000 Btu/hr-ft²). The calculated maximum burnup in the plate was 1.4 percent of the total core atoms.

The test was completed without incident or difficulty. A second examination of the plate after 27 cycles of irradiation showed that the plates decreased uniformly 7.1 μ (0.0028 in.) in thickness with no evidence of pitting or swelling. The final examination of the plate is presently being completed.

MTRW-10

Test MTRW-10 was conducted to determine the effect of neutron irradiation and high burnup on the cladding and fuel of a BORAX-IV prototype fuel element. The test specimen was fabricated by the same procedures and materials used in the manufacture of fuel plates for the BORAX-IV reactor. Figure 13 is a sketch of the specimen, which consisted of thorium-uranium pellets, bonded with lead in an extruded M-388 aluminum alloy tube plate. The pellets were 0.63 cm ($\frac{1}{4}$ in.) in diameter by 1.3 cm ($\frac{1}{2}$ in.) long. The mixture used in the preparation of the pellets consisted of 93.4 w/o ThO₂ and 6.54 w/o U₃O₈. The U₃O₈ was 93 percent enriched.

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Figure 13. Prototype BORAX-IV Fuel Plate of $\text{ThO}_2\text{-UO}_2$ pellets in Aluminum Alloy Fuel Plate Used for MTRW-10 Experiment

Few operational difficulties occurred during the test period, which totalled 7 MTR cycles. The desired operating conditions for the test were 42.3 kg/cm^2 and 241°C with neutral, high-purity water. The coolant-water velocity past the test specimen was about 4 m/sec . The experiment was designed such that nucleate boiling occurred on the forward 7 cm of the specimen. The total neutron exposure was 2365 MWd.

A detailed account of the test may be found in Ref. 8.

ANL-2-11

Test ANL-2-11 was conducted to determine the effect of high burnup on an M-388 aluminum-alloy-clad, uranium-aluminum plate of the SL-1 type. The plate was fabricated by the same procedure and materials used in the manufacture of the SL-1 core. The fuel was an alloy consisting of 17.5 w/o uranium (93% enriched), 2 w/o nickel, and 2S aluminum.

Operating conditions for the test were 26.8 kg/cm^2 and 216°C with high-purity neutral water. The experiment was designed such that nucleate boiling occurred on the forward 11 cm of the plate. The plate was irradiated for a period of eleven MTR cycles, at which time the test was terminated because of a failure in the cladding material. The total neutron exposure to the plate was 6293 MWd, resulting in a maximum burnup of slightly over 1.25% of the total core atoms.

The cladding failure occurred following a period in which difficulties in maintaining pH and water purity were experienced. During this seven-day period pH values as high as 10 were recorded, and the failure was attributed to this operation with high-pH coolant water. It is interesting to note that no operational problems resulted from the failure of this specimen.

A detailed account of the test and its results may be found in Ref. 9.

MTRW-12

Following test MTRW-11, a second SL-1 prototype plate was irradiated in the ANL-2 loop under test conditions similar to those of test MTRW-11. During previous tests, objectionable amounts of oxide scale had accumulated on the test specimens. For the most part this scale consisted of corrosion products from the loop surfaces. Just prior to this test the loop was thoroughly cleaned in the hope that the rate of deposition of this scale would be reduced.

The in-pile specimen was irradiated for 125 full-power days for a maximum total atom percent burnup of 1.0 percent. During the irradiation period the core thickness increased 16 percent. A report covering this test will be published in the near future.

ANL-2-13

The specimen for test ANL-2-13 consisted of a three-rod prototype fuel element. Each rod was 41 cm (16 in.) long and contained a column of 0.825-cm (0.321-in.)-diameter UO_2 pellets in wrought A-288 aluminum alloy tubing. One of the three rods contained square-end pellets with a $25.4\text{-}\mu$ (0.001-in.) annular gap filled with helium. The second rod contained square-end pellets with the cladding precollapsed onto the pellets. The third rod contained pellets with concave ends and a $25.4\text{-}\mu$ (0.001-in.) helium-filled annulus. After the specimen was inserted into the in-pile tube, the loop was pressure tested to 105.5 kg/cm^2 (1500 psi) in a standard procedure for leak testing. At this pressure it was expected that the cladding of all three of the rods would be collapsed onto the pellets.

The purpose of the test was to determine if ratcheting between the fuel pellets and the cladding would occur because of differences in thermal expansion. The use of corrosion-resistant alloy tubing as a cladding for oxide fuel in water-cooled reactors could lead to reduce fabrication costs.

The test conditions were 42.2 kg/cm^2 (600 psi) and 243°C with a coolant velocity of 683 cm/sec (22.4 ft/sec). For most of the irradiation period the coolant-water pH was maintained between 5.0 and 6.0. The element was irradiated for three months to a maximum burnup of 6100 MWd/MT.

An examination of the element following the test showed that the first rod had elongated 0.52 mm (0.021 in.), the second rod had elongated 0.38 mm (0.015 in.), and the third rod had elongated 0.13 mm (0.005 in.). The results of the test further showed that thermal ratcheting does not present a serious problem.

ANL-2-16

During test ANL-2-13 an identical element, ANL-2-16, was exposed to the same coolant water in the out-of-pile tube. Following test ANL-2-13, this element was irradiated in the loop under conditions similar to those of test ANL-2-13 for a period of five months to a maximum burnup of 8600 MWd/MT. During the irradiation period, low-level fission gas activity was detected from samples of coolant water removed from the loop, indicating a small defect in the element.

An examination of this element following the test showed that the first rod had shortened 25.4μ (0.001 in.), the second rod had elongated 0.508 cm (0.020 in.), and the third rod had elongated 3.05 mm (0.12 in.). Although there was no clear-cut correlation between the results of this and the ANL-2-13 tests, both indicated that no problems would be experienced as a result of thermal ratcheting.

VIII. CONCLUSIONS

At the time the ANL-2 loop was designed, the need for the facility was expected to be less than five years. If the projected life of a test loop is expected to exceed three years, some consideration should be given initially to the possible future need for decontamination and cleanup. Unnecessary traps should be avoided. Some means for the cleaning of low-flow areas in the system should be considered. The need for future repair and maintenance should also be considered initially.

Versatility of operation is a necessary feature of a test loop. The ANL-2 loop was designed to operate at 105.5 kg/cm^2 (1500 psi), but later tests required operation at 26.8 kg/cm^2 (380 psi). Certain modifications were needed before the boiling water tests could be conducted.

The ANL-2 loop has certainly withstood the test of time and very little can be added in the way of constructive criticism of the facility.

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