OAK RIDGE NATIONAL LABORATORY

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53

WATER-LOSS TEST AT THE LOW-INTENSITY TESTING REACTOR

J. A. Cox
C. C. Webster

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WATER-LOSS TEST AT THE LITR

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ABSTRACT

This is a report on the water-loss experiments performed at the LITR during 1951, 1952, and 1953 to determine the temperature rise of the fuel elements due to the decay heat of the fission products when the water was suddenly drained from the reactor. Tests were performed with the reactor operating at several power levels up to 2300 kw with and without an auxiliary cooling system.

INTRODUCTION

At the time the LITR was being built, the Reactor Safeguards Committee stipulated that a safe operating power level should be one at which the fuel plates would not be melted by fission-product heat if the cooling water, normally in contact with the fuel plates, should suddenly be lost due to rupturing of the reactor tank.

The heat-transfer routes and the thermal resistance along these routes from the fuel plates to the beryllium reflector, support castings, and other large objects in the reactor could not be determined accurately by a mathematical treatment. It was therefore necessary to measure the temperatures reached by the fuel, following the sudden loss of water during operation, under extreme conditions to assure the safety of the reactor for power operation.

This information was obtained by performing a series of tests, at successively higher power levels, at the LITR during 1951, 1952, and 1953 wherein the temperature within the fuel elements was measured when the cooling water was suddenly lost from the reactor tank during level-power operation. As a result of the tests, data tabulated in Table 1 and Table 2, it was determined that auxiliary cooling would be advisable for power operation of the LITR near 3000 kw.

Tests up to, and including, the 1250-kw power level without auxiliary cooling and from 1250 to 2300 kw with auxiliary cooling were performed. The auxiliary coolant was supplied by gravity flow from a tank of ~500-gal capacity through two spray nozzles capable of spraying 3 gal/min onto the core. The results of loss-of-water tests will be presented where the temperature was measured as a function of time when the reactor had been operating at power levels up to 2300 kw, including one test with the spray tank and nozzle system installed.

DESCRIPTION OF THE LITR

The LITR (Low-Intensity Testing Reactor) was originally built as the mock-up of the Materials Testing Reactor for hydraulic measurements and critical tests. The core contains uranium-aluminum alloy fuel elements and cadmium control elements (with a follow-section made up of uranium-aluminum fuel plates) held in place by a cast-aluminum grid work. The fuel elements are made up of curved uranium-aluminum fuel plates separated by coolant flow channels containing water to cool the fuel plate surfaces and provide the necessary neutron moderator. The core has a beryllium reflector on the sides and a water reflector on each end. The arrangement of the reactor core in the reactor tank can be seen in Figures 1 and 2.

The arrangement of the fuel, beryllium, and experiments within the lattice, on May 12, 1952, is shown in Figure 3. The shim rods had fuel-element followers which contained only 14 fuel plates and about 100 g of U^{235} as compared to 18 fuel plates and 130 to 140 g of U^{235} for the fuel elements. The total fuel content of the core was about 3100 g of U^{235}.

The reactor was normally operated with the No. 1 shim rod (lattice position C-22) and the No. 3 shim rod (C-26) completely withdrawn so that 100% of the fuel followers were in the core. The No. 2 shim rod (C-25) was withdrawn about 66%.

The coolant flow enters the reactor vessel through the side about 3 ft from the top of the tank and flows down through the lattice and fuel, from whence it is drawn up through a bundle of tubes and passes out of the reactor tank at the same elevation as the entrance. For the loss-of-water test, the bottom valve was opened and then the primary coolant pump was shut off to assure uninterrupted water flow through the fuel until the water level in the tank dropped below the core. The water was drained from the reactor through a 6-in. flanged valve attached to the bottom of the reactor tank. Two Lucite-covered manholes are located in the top cover of the

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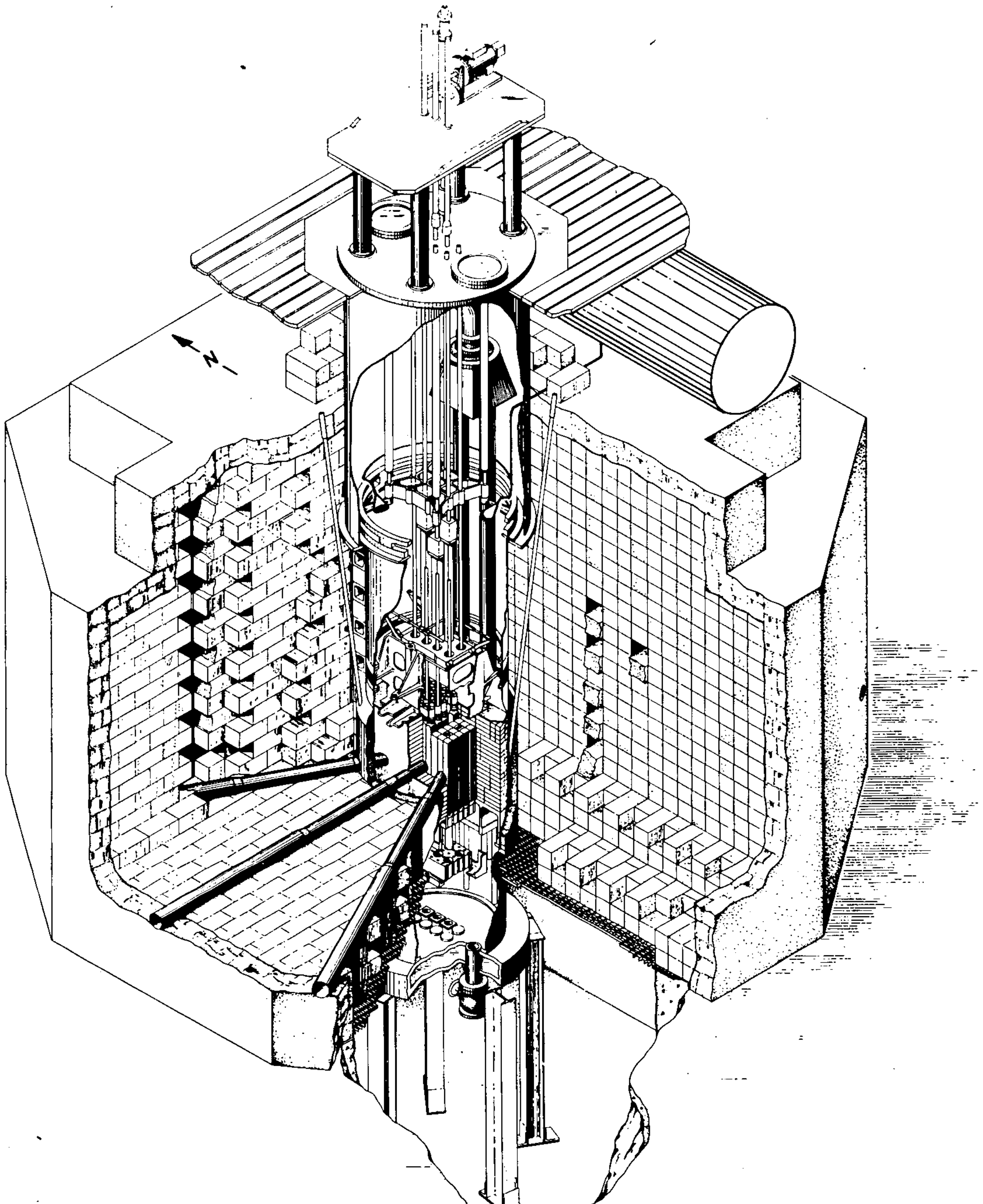

Fig. 1. Low Intensity Test Reactor

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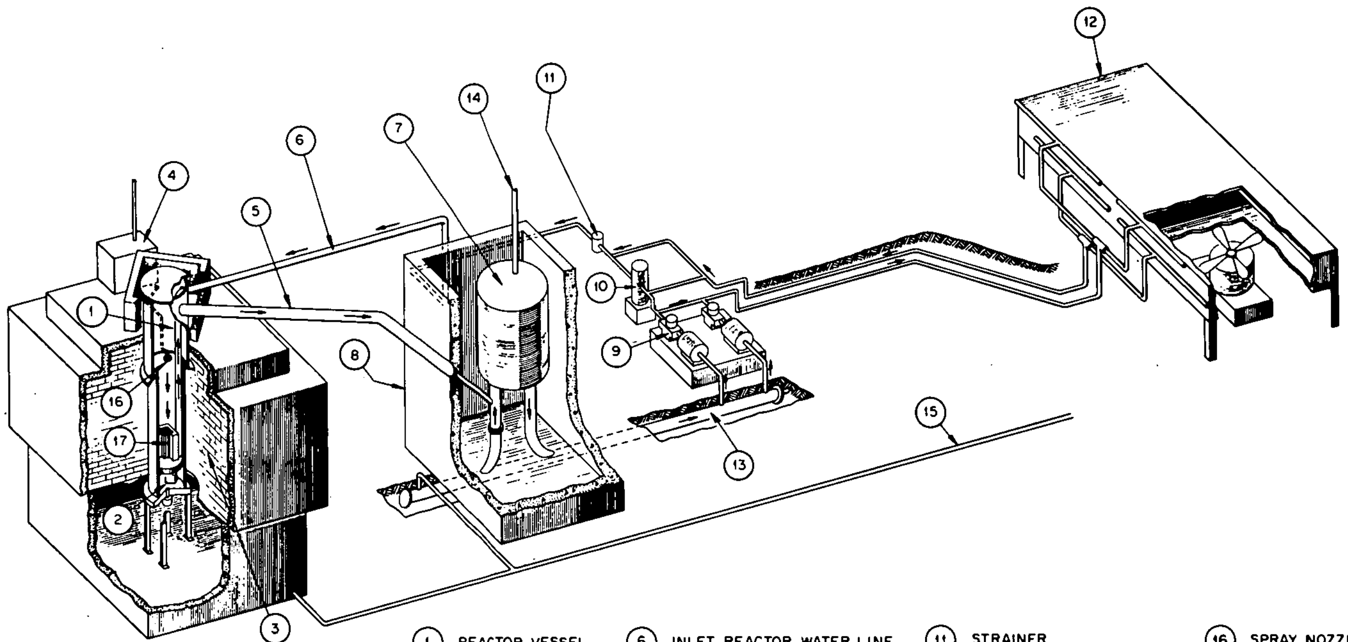

(1) REACTOR VESSEL
(2) EXIT WATER LINES
(3) REACTOR SHIELD
(4) SPRAY TANK
(5) EXIT REACTOR WATER LINE
(6) INLET REACTOR WATER LINE
(7) SEAL TANK
(8) SHIELD FOR SEAL TANK
(9) WATER CIRCULATING PUMP
(10) SHELL AND TUBE HEAT EXCHANGER
(11) STRAINER
(12) FLUID COOLER
(13) REACTOR WATER LINE
(14) VENT TO OFF-GAS
(15) CONTAMINATED DRAIN TO LITR RETENTION POND
(16) SPRAY NOZZLE FOR SPRAY TANK
(17) LITR CORE

Fig. 2. LITR Cooling Water System

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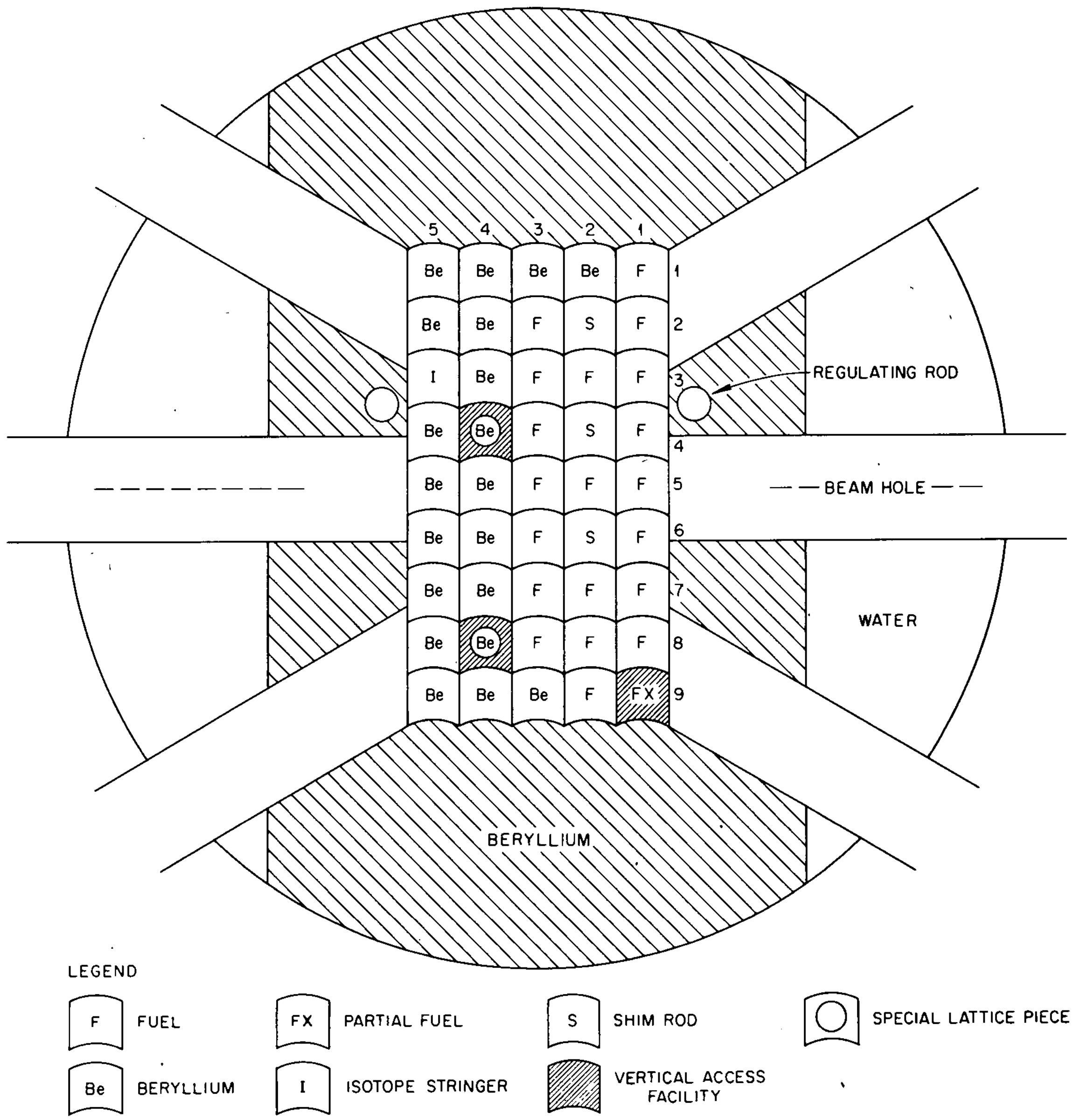

Fig. 3. LITR Core Arrangement on 5/12/52

reactor tank; lead wires from thermocouples, positioned in specific fuel elements to monitor the temperature during the tests described below, passed through one of these ports.

TESTS AND RESULTS

Several loss-of-coolant tests were conducted during the early performance runs made with the reactor. These early tests are listed in Table 1 and are described in reports ORNL-1075 and ORNL CF-52-2-158. Three tests will be described in this report: one performed on May 12, 1952, at 1000 kw without auxiliary cooling; one on May 19, 1952, at 1250 kw without auxiliary cooling; and one on August 31, 1953, at 2300 kw with auxiliary cooling.

May 12, 1952, Loss-of-Water Test

After the reactor had been operated for 142 hr at 1000 kw, the water was suddenly drained from the reactor tank; and the temperatures at two locations in one of the fuel elements were monitored by means of thermocouples. Before the start of this particular reactor cycle, two thermocouples were placed in the fuel element then located in core-position C-25. The thermocouples were inserted into an aluminum tube with their junctions positioned 6 in. apart in the axial direction. The aluminum tube was about 10 ft long with Tygon tubing attached to the upper end and extending out of the reactor tank. The aluminum tubing was flattened over the lower portion of its length to about 0.050 in. thick so that it would fit into the coolant channel.

Because the flow channels are somewhat greater than 0.10 in. wide, the thermocouple tube cannot be considered to have been in contact with the fuel plates. Therefore, one can expect that the fuel-plate temperature was somewhat higher than the recorded temperature until thermal equilibrium was achieved.

Figure 4 is a plot of the temperature versus time for the thermocouples No. 1 and 2 in the fuel element in position C-25 located at 12 5/16 and 6 5/16 in., respectively, below the top of the core. The time scale is translated horizontally so that zero time is indicated when a drop in the reactor-water temperature was observed. For this particular test, the

Table 1. Water-Loss Test Data

The results listed below are discussed in ORNL-1075 and ORNL CF-52-2-158

Test Number	Power Level (kw)	Time at Power (hr)	Time to Reach T_{max} (sec)	ΔT_{max} (°F)	ΔT_{max}[1]
3	22.5	2.13	2100	7.5	16.3
4	60	2.5	2100	16.5	35.0
5	90	2.5	2400	22.0	46.6
6	112	2.25	2400	26.0	56.7
7	112	2.17		28.5	62.3
8	135	2.08	2400	31.0	69.7
9	150	2.2	2700	33.5	73.2
10	150	6.5	2400	40	72.0
11	150	24.5	6000	53	76.0
12	300	21.0	3100	62.5	136.5
13	150	24.5	9000	47.0	53.0
14	150	117.0	9000	60.0	68.0
15	350	129.0	9000	126.0	140.0
16	770	131.6	6000	180.0	215.0[2]

[1] Normalized to 120 hours level power.

[2] Reactor shut down 2 minutes early.

Table 2. Water-Loss Test Data

Test Number	Date	Power Level (kw)	Time at Power (hr)	Temperature at Start (°C)	T_{max} (°C)	Time to Reach T_{max} (sec)	ΔT_{max} (°F)
17	5-12-52	1000	142	50	248	6420	360
18	5-19-52	1250	138	50	254	8100	376[1]
19	2-23-53	1250	115		204	6000	
20*	3-2-53	1250	141		86	1320	187[2]
21*	3-9-53	1500	152	45	86	1020	185
22*	6-22-53	1500	143	40	84	2160	183
23*	8-24-53	1900	134	46	79	1600	174
24*	8-31-53	2300	114	52	92	1080	198

(1) Might have been 40°C higher if the reactor had not shut down prematurely.

(2) Spray tank went dry at 2 hr 35 min. At 3 hr 42 min, temperature was 160°C in C-36.

*Tests performed using auxiliary spray.

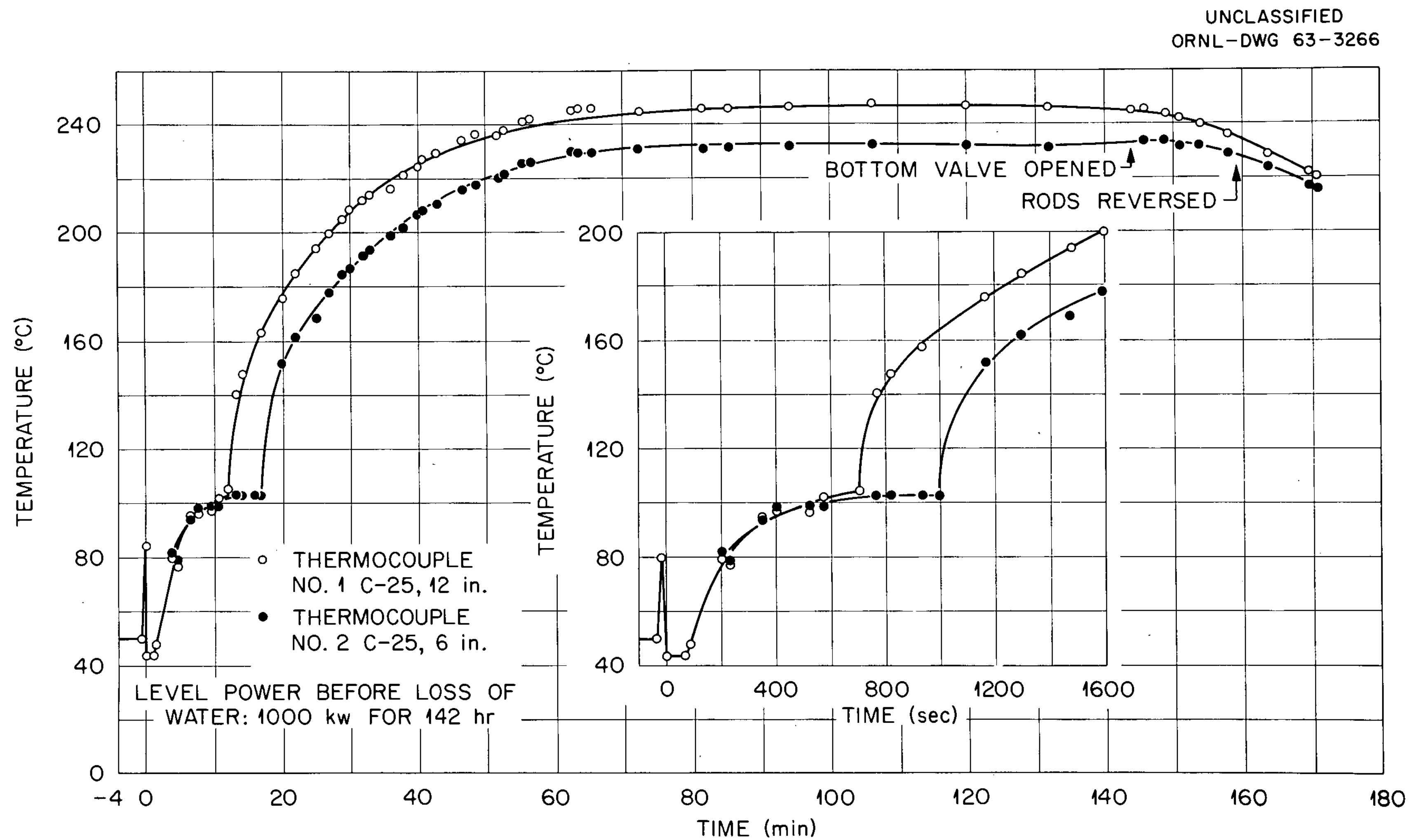

Fig. 4. Water Loss Tests at LITR; 5/12/52

safety system was interceded and the reactor was operated manually so that the reactor was shut down, not by insertion of poison control elements but by loss of reflector and moderator. The control rods were not inserted into the core until more than 2 1/2 hr later when the thermocouples indicated a decreasing temperature.

The thermocouple No. 1 indicated an increase in the temperature of the water adjacent to it when the flow of water through the core was reduced by stoppage of flow through the normal coolant outlet line with the reactor still operating at level power. When the reactor shut down due to a loss of reflector, the temperature dropped below the earlier equilibrium value. The temperature remained at this lower value until the water drained out of the core. The bottom valve was then closed which prevented any stack cooling effect of the fuel. The temperature then rose to about 100°C and remained at this value until all the water had evaporated from the surface of the thermocouple tube and the adjacent fuel plate surfaces. The temperature then rose to a maximum value of 248°C when the heat-removal rate from the fuel plate equaled the generation rate.

The bottom valve and top manhole were opened simultaneously 142 min after reactor shutdown so that a chimney or stack effect could be used to help cool the fuel elements. The effect of this is immediately noticeable. The temperature continued to drop and 12 hr later had dropped to ~150°C. At this time, the water spray, which had been previously made ready at the top of the tank, was opened and the temperature registered by the thermocouples dropped sharply to the boiling temperature of water.

The heat-generation rate due to fission products is:

$$\frac{dq}{dt} = K\,P_\circ\,[t^{-0.2} - (t + T_\circ)^{-0.2}]$$

where $P_\circ$ is the level reactor power before shutdown, t is the time (sec) after shutdown, and $T_\circ$ is the time (sec) the reactor is at power just prior to shutdown.

The heat removal rate is:

$\frac{dq}{dt} = H\,\frac{d\theta_1}{dt}$ where H includes both the heat-conduction and heat-convection constants, and θ_1 is the temperature difference between that portion of the fuel plate under consideration and its environment. Because of the low temperature difference involved, we will neglect the heat transfer by radiation. Above 100°C the heat is removed by the natural convection of the air passing through the coolant channels and by conduction along the fuel elements to structural parts of the core.

The heat absorption rate is:

$\frac{dq}{dt} = MC_p \frac{d\theta}{dt}$ where M is the mass of that portion of the fuel plate being considered, C_p is its heat capacity, and θ is the instantaneous temperature of the plate.

The heat absorption rate = the heat generation - the heat-removal rate. If the heat-removal rate could be determined, it would be easy to calculate the maximum temperature which the hottest section of a fuel element would reach if a loss of the coolant water occurred.

May 19, 1952, Loss-of-Water Test

The reactor had been operated for 138 hr at the 1250-kw power level when the water was suddenly drained from the reactor by opening the bottom valve and shutting off the pump. At the same time, the reactor scrammed due to the dropping of one of the shim control rods apparently caused by the change in water flow. Because of this, the reactor was shut down for ~2 min while the water was draining from the top portion of the vessel before the water level reached the top of the core. This meant that the fission-product heat generated during that first two mintues was carried away by the reactor water and did not contribute to the heating of the fuel plate. In this test there were two additional thermocouples, prepared similar to those in C-25, inserted into the fuel element in position C-28; these were designated No. 3 and 4 and positioned 12 5/16 in. and 6 5/16 in., respectively, from the top of the element. Part (a) of Figure 5 shows the thermocouple positions in the fuel elements.

The temperature data are plotted in Figures 6 and 7. Initially, No. 3 in the center of C-28 indicated a higher temperature than No. 1 in the center of C-25. This has been attributed to the fact that the one in C-28 was in a tube which had been made ~1/10 in. thick so that it fitted more closely between the fuel plates. The one in C-25 was much thinner, ~0.050 in. thick; and there was some question that the thermal contact between it and the fuel plates might be poor. At the beginning, the thermocouples in C-28 were about 8°C hotter than the ones in C-25, although the maximum temperature of No. 1 in C-25 was 24°C hotter than No. 3 in C-28.

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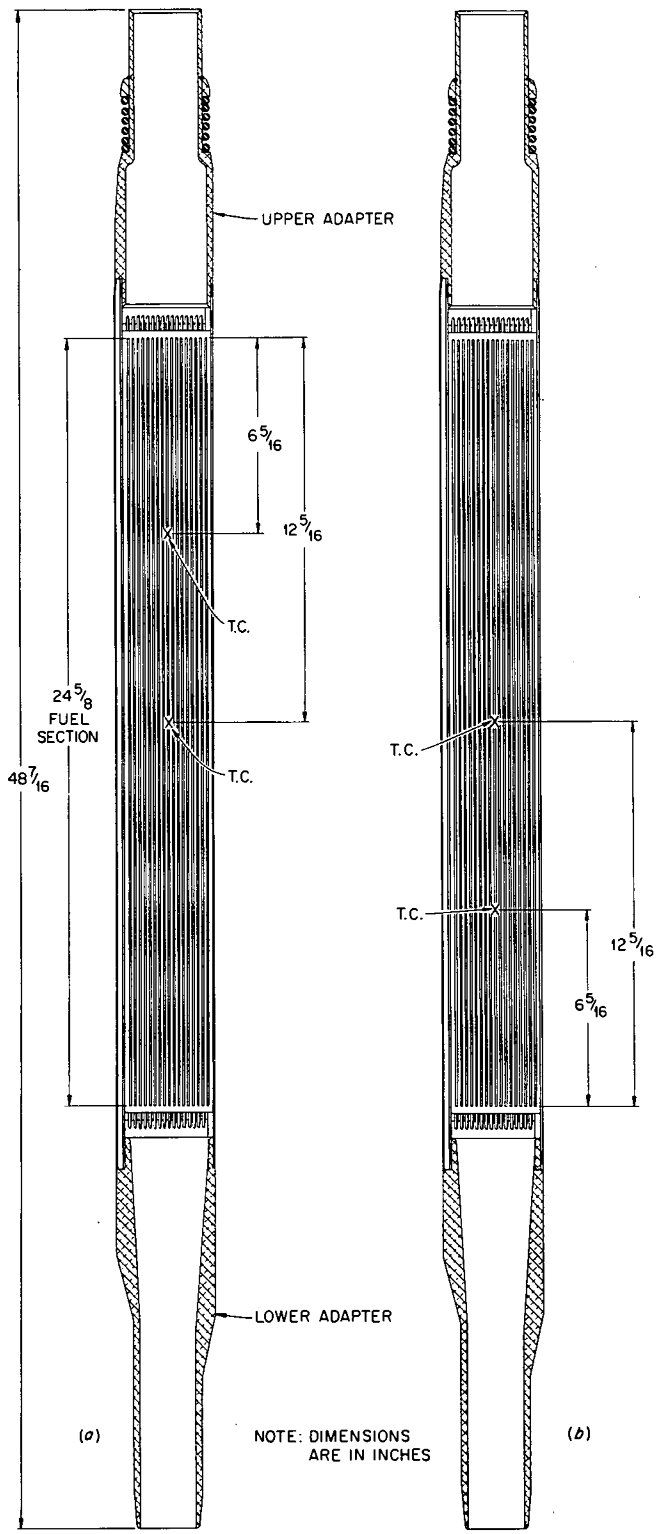

Fig. 5. Fuel Elements Showing Thermocouple Locations

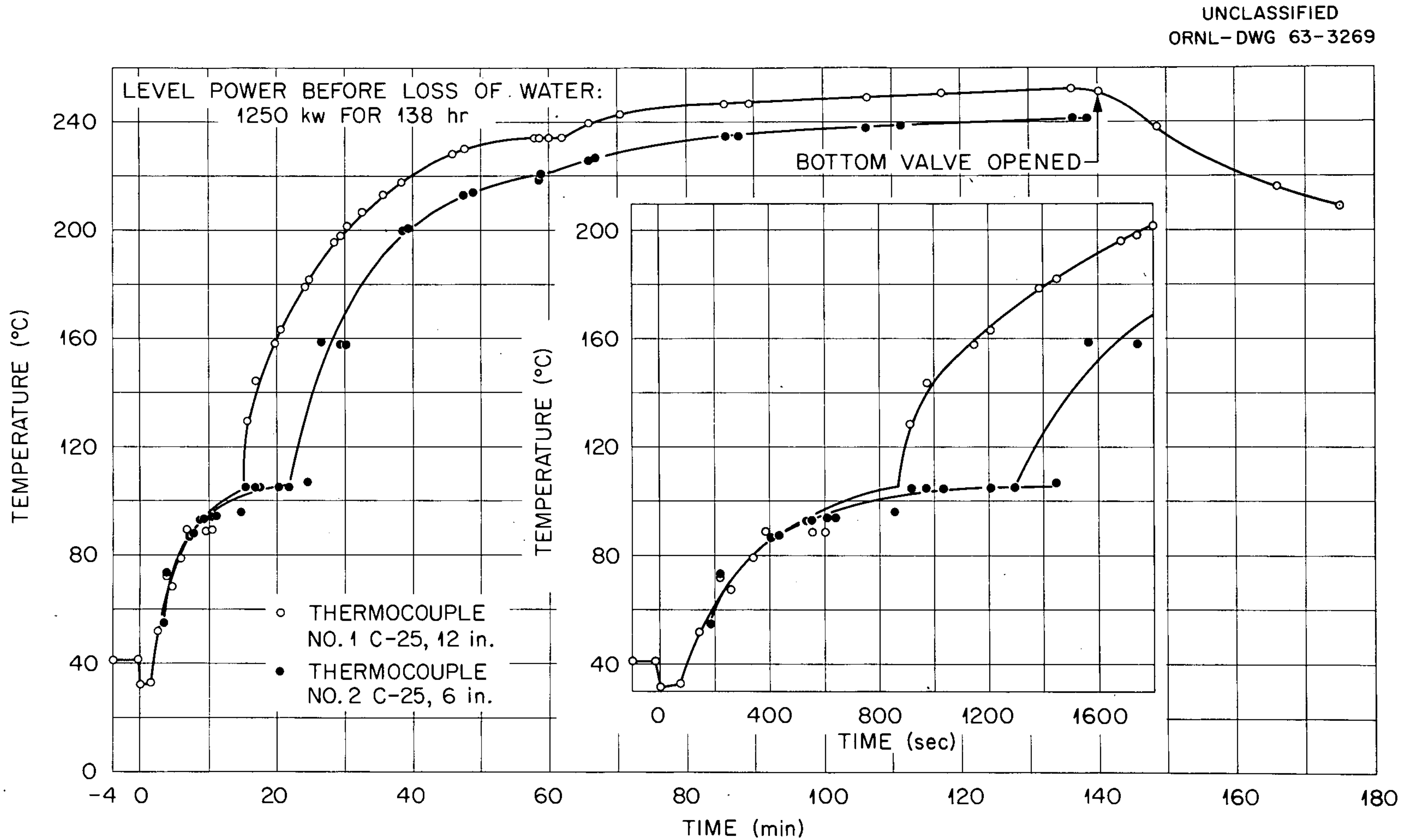

Fig. 6. Water Loss Tests at LITR; 5/19/52

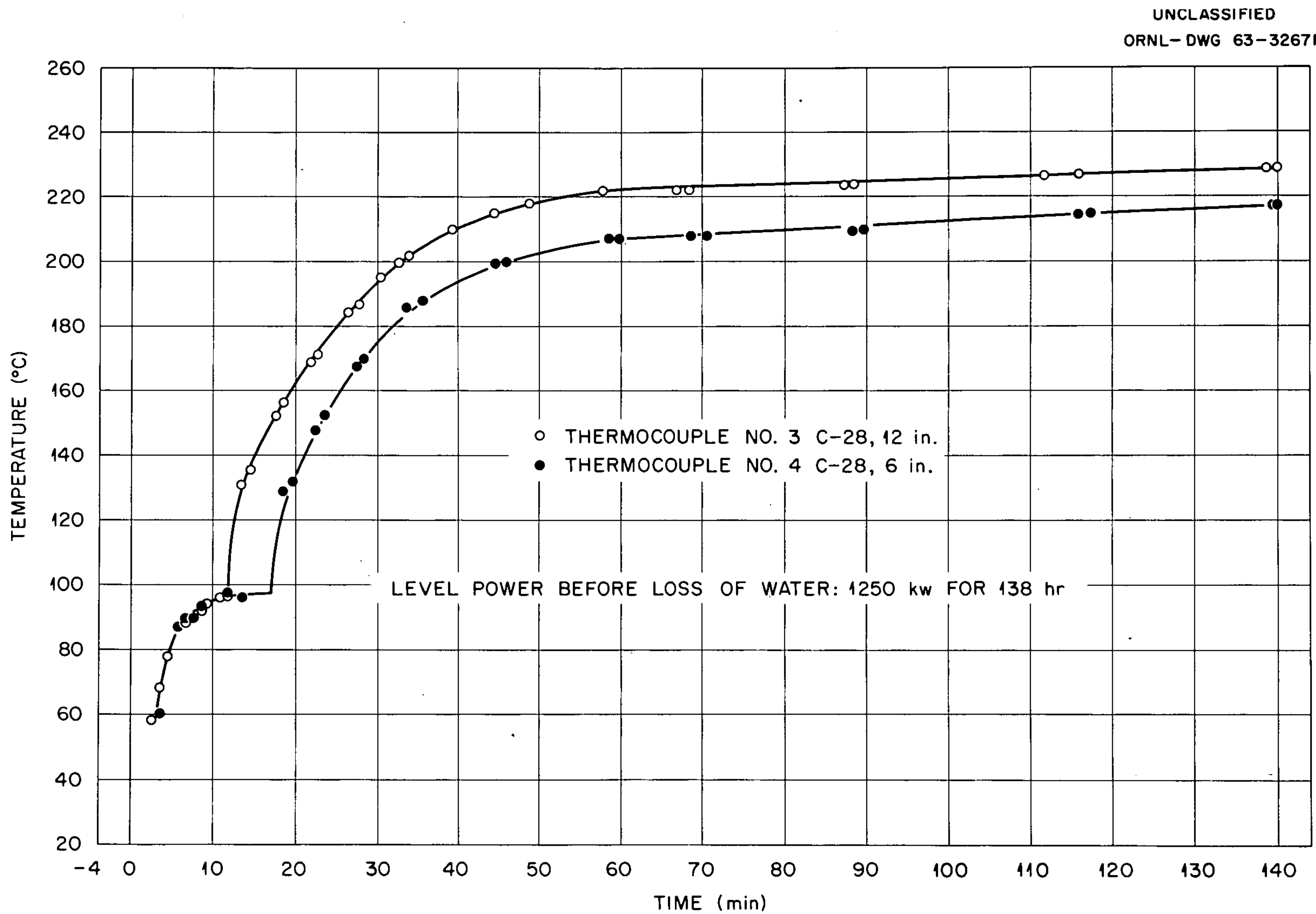

Fig. 7. Water Loss Tests at LITR; 5/19/52

This suggests that at the maximum temperature, equilibrium conditions prevailed so that the thermocouples and the fuel plates were at approximately the same temperature.

The thermocouples in C-25 indicated the maximum temperature of 253°C was attained in this test 135 min after the temperature started rising; whereas, in the earlier test, the maximum indicated temperature of 248°C was reached 106 min after the temperature rise started. The power level of the second test was 1250 kw, while the earlier test performed on May 12, 1952, had been at 1000 kw. The higher-power run took 27% longer time to reach its maximum temperature. It should be noted that 105 min after the temperature rise started, the No. 1 thermocouple, located near the center of the fuel element in position C-25, reached the same temperature in both tests. This suggests that the maximum temperature achieved is quite dependent upon the mechanism of heat removal. Since the heat removal is dependent upon both the heat generation rate and the maximum temperature of the fuel plates, one would expect a longer time for the fission-product heat generation of a higher power level of operation to raise the fuel plates to an equilibrium condition between generation rate and removal rate.

At this stage of the test, it was necessary to determine the amount of auxiliary cooling required to remove the fission-product heat after higher-power operation.

Assuming the same heat-transfer conditions to exist at higher heat-generation rates, one can conclude that a loss of water after operating the reactor at a power level above 2000 kw could result in some melting of the fuel. What rate and quantity of auxiliary cooling is necessary to assure that a safe temperature would be maintained following a loss of reactor coolant water? The following equation can be used to determine the rate of energy release by β and γ rays emitted by the fission products as they decay toward stability for times greater than 10 sec after reactor shutdown. For times less than 10 sec, the rate of energy release is somewhat greater.

$$R_t \cong 6.4 \times 10^{-2} \, P \left[t^{-0.2} - (t + T_o)^{-0.2} \right] \text{ watts}$$

where P is expressed in watts of thermal power, T_o equals time reactor was operating, and t equals time after reactor shutdown (see Appendix A).

Assume that the reactor shuts down as the water leaves the core and that emergency cooling is not required until 20 sec after shutdown from 3000-kw operation to determine the amount of cooling water required.

$t = 20$ sec

$T_\circ = 114 \text{ hr} = 4.1 \times 10^5$ sec

$R_t \cong 6.4 \times 10^{-2} \times 3 \times 10^6 \left[(20)^{-0.2} - (4.1 \times 10^5)^{-0.2} \right]$

$\cong 19.2 \times 10^4 \left[0.55 - 0.075 \right] \cong 19.2 \times 10^4 \times 0.475$

$\cong 91.0 \times 10^3$ watts $\cong 91$ kw.

Assuming that the water from the auxiliary cooling system is at 100°F when sprayed onto the fuel elements, what flow rate of water is required to maintain a safe fuel-element temperature?

1 kw = 3412 Btu/hr = 57 Btu/min

$R_t \cong 91 \times 57 = 5180$ Btu/min

$Q = R_t = M\, Cp\, \Delta t + M\, L_V$

where M = V (gal/min) x 8 lbs/gal

5180 Btu/min = 8 V [112 + 970]

$V = \dfrac{5180}{8\,(112 + 970)} = 0.615$ gal/min

if all the water were to evaporate and if the heating rate were uniform throughout the core. However, the fission products may reach a local peak concentration value between 1.5 and 2 times the average value. Therefore, a coolant flow rate of about 3 gal/min will give sufficient flow with a safety margin.

August 31, 1953, Test with Auxiliary Cooling

Tests were carried out at the LITR during 1951 and 1952 to determine what temperature the fuel elements might reach if the cooling water should suddenly be lost from the reactor tank during operation. After a series of these tests, some of which are described in reports ORNL-1075 and ORNL CF-52-2-158 and those described earlier in this report, it was determined that auxiliary cooling shouldbe required for operation above 1500 kw. The system finally adopted was an auxiliary tank of ~500-gal capacity so arranged as to be kept filled automatically by the circulating water. A line from this tank was brought through the wall of the reactor tank and terminated in two nozzles about 7 ft above the active lattice. These nozzles were so sized that they would deliver the water from the auxiliary tank at about 3 gal/min in a fine spray over the top of the active lattice. The normal water path and the location of the sprays is shown in Figures 1 and 2.

Table 3. Water-Loss Tests with Spray Tank Cooling

Date	Power kw	Temperature °C Start	°C Maximum	°F Maximum	Remarks
3/2/53	1250		86	187	Core allowed to heat up for 62 min after spray tank ran dry without water cooling. Temperature increased from 65°C to 156°C.
3/9/53	1500	65	85	185	Apparatus prevented closing the Lucite cover at the top of the tank and created more stack effect.
6/22/53	1500	46	84	183	Same as above.
8/24/53	1900	46	79	174	No. 2 shim rod dropped causing lower temperature.
8/31/53	2300	52	92	198	Results as expected.

The test at the highest power (2300 kw) was made on August 31, 1953, after the reactor had operated for 114 hr at 2300 kw. Following is a list of preparations made for this test:

1. During the previous week, thermocouples were inserted between fuel plates of the fuel elements No. C-36, C-32, and C-25. In addition, thermocouples were installed adjacent to the beryllium between an aluminum plate and the stacked beryllium next to the fuel; and a thermocouple was placed in V-4, one of the four inclined holes attached to the outside of the tank wall at the same elevation as the center plane of the core. All of the thermocouples in the fuel and beryllium were prepared by placing two thermocouples, one 6 in. above the other, in an aluminum tube which was then rolled flat to a thickness of ~0.050 in. The aluminum tube was ~10 ft long so that the upper portion of it extended out of the zone of radiation, and a Tygon tube was used to protect the

thermocouple wires from the water the rest of the way out of the reactor tank. In the case of the thermocouples which were placed in the fuel elements, an aluminum stop was welded to the side of the tube so that the higher of the two thermocouples in the tube would be located in the center of the uranium fuel while the lower one would be situated 6 5/16 in. above the bottom of the uranium fuel. The arrangement of a thermocouple in a fuel element and the general arrangement in the reactor tank are shown in Figure 3 and part (b) of Figure 5.

2. The water level in the seal or surge tank was lowered to 3 1/2 ft so that the water level in the reactor tank would be below the top plug and would permit removal of the manhole cover. Previous tests had shown that as the water level dropped in the reactor tank, considerable gaseous activity came out. A 2-in. suction line was inserted in the manhole to carry the gaseous activity to the stack. An emergency spray was also installed in the open manhole so that it could be turned on if any failure of the auxiliary spray system should occur or if the need for additional cooling was indicated after the auxiliary tank ran dry.
3. To insure that the reactor would continue to operate while the water was draining from the reactor tank, the scram circuits, normally activated by reduction of water flow and exit-valve closure, were interceded so that stopping the pump and closing the exit valve would not terminate the reactor operation.
4. The suction from the pit, into which the reactor water drained, was increased so that radioactive gas would not escape into the control room.
5. The currents to the magnets holding up the shim rods were increased 10-20 ma. Previous experience had shown that one rod often dropped because of the power fluctuations encountered during the test. In normal operation the magnet currents are set rather close to the drop point, and a short period fluctuation would sometimes cause one rod to drop.
6. The reactor control was changed from servo to manual.

7. The bottom drain valve was opened until flow into the pit was established.
8. The circulating pump was stopped and the exit valve was closed to prevent water siphoning back into the reactor tank from the seal tank.
9. As the water drained from the tank, the power fluctuated somewhat and the operator attempted to hold it at 2300 kw by withdrawing or inserting a shim rod.

As the water drained from the reactor tank in the manner described, temperature readings (shown in Table 4) were taken at regular time intervals on all of the thermocouples.

Figure 8 is a plot of the points taken from a recorder trace of the thermocouple located in the middle of the element positioned in C-36. The readings from the thermocouple located in the center of the element in position C-25 are shown for comparison. Assuming that the reactor water inlet temperature was about 100°F and that an equal amount of auxiliary cooling water passed through each element, the element in position C-36 was generating ~25% more heat than the one in position C-25.

The highest temperature observed was 92°C showing that the auxiliary spray tank offered ample cooling capacity to protect the fuel elements from excessive overheating following loss of water. The cooling effect of air entering an open drain valve and leaving by the vent at the top of the tank is demonstrated on the chart following the closing of the bottom valve. At this point, the thermocouple shows a temperature increase from ~85°C to ~92°C in about 10 min. The effect of the fuel and the shim rods being removed from the lattice is also demonstrated when the rods were reversed, as indicated on the chart, following which the temperature decreased from 92°C to ~85°C in ~5 min. These three shim rods carried ~10% of the fuel in the reactor.

Radiation-Level Measurements on August 31, 1953

A number of other observations were made during the test including the radiation level at various times. For example, in the control room situated at ground level, approximately the same elevation as the reactor core and about 20 ft north (separated, of course, by ~11 ft of concrete shielding), the following radiation readings were obtained.

Table 4. Loss-of-Water Test

August 31, 1953

Time (PM)	No. 1 Be (Middle)	No. 2 Be (Bottom)	No. 3 C-36 (Middle)	No. 4 C-36 (Bottom)	No. 5 C-25 (Middle)	No. 6 C-25 (Bottom)	No. 7 V-4	No. 8 C-32 (Middle)	No. 9 C-32 (Bottom)
	Thermocouple Positions and Locations — Temperature Reading °C								
4:45	35	41	52	48	37	44	86	46	47
4:52	Down								
4:53	38	43	70	49	63	64	85	62	66
4:55	50	53	80	92	77	70	85	76	78
4:57	67	68	90	90	79	74	85	80	80
5:00	70	70	91	91	83	82	84	79	80
5:05	67	66	92	91	84	81	83	78	78
5:10	68	67	92	92	85	81	82	78	78
5:15	69	67	90	88	84	83	82	76	76
5:20	66	68	89	86	84	82	82	75	76
5:25	65	67	88	85	84	81	81	74	74
5:30	64	66	88	84	82	81	81	73	73
5:35	64	66	87	84	82	80	81	72	72
5:40	65	66	87	83	80	80	81	72	72
5:45	64	66	86	83	80	78	81	72	72
5:50	63	66	86	83	79	78	80	71	72
5:55	63	66	85	83	76	75	80	71	72
5:56	Closed Bottom Valve								
6:01	66	66	92	88	92(?)	87	80	70	72
6:05	69	68	93	90	81	77	80	70	72
6:07	Reversed Rods								
6:10	67	66	86	76	80	76	79	62	63

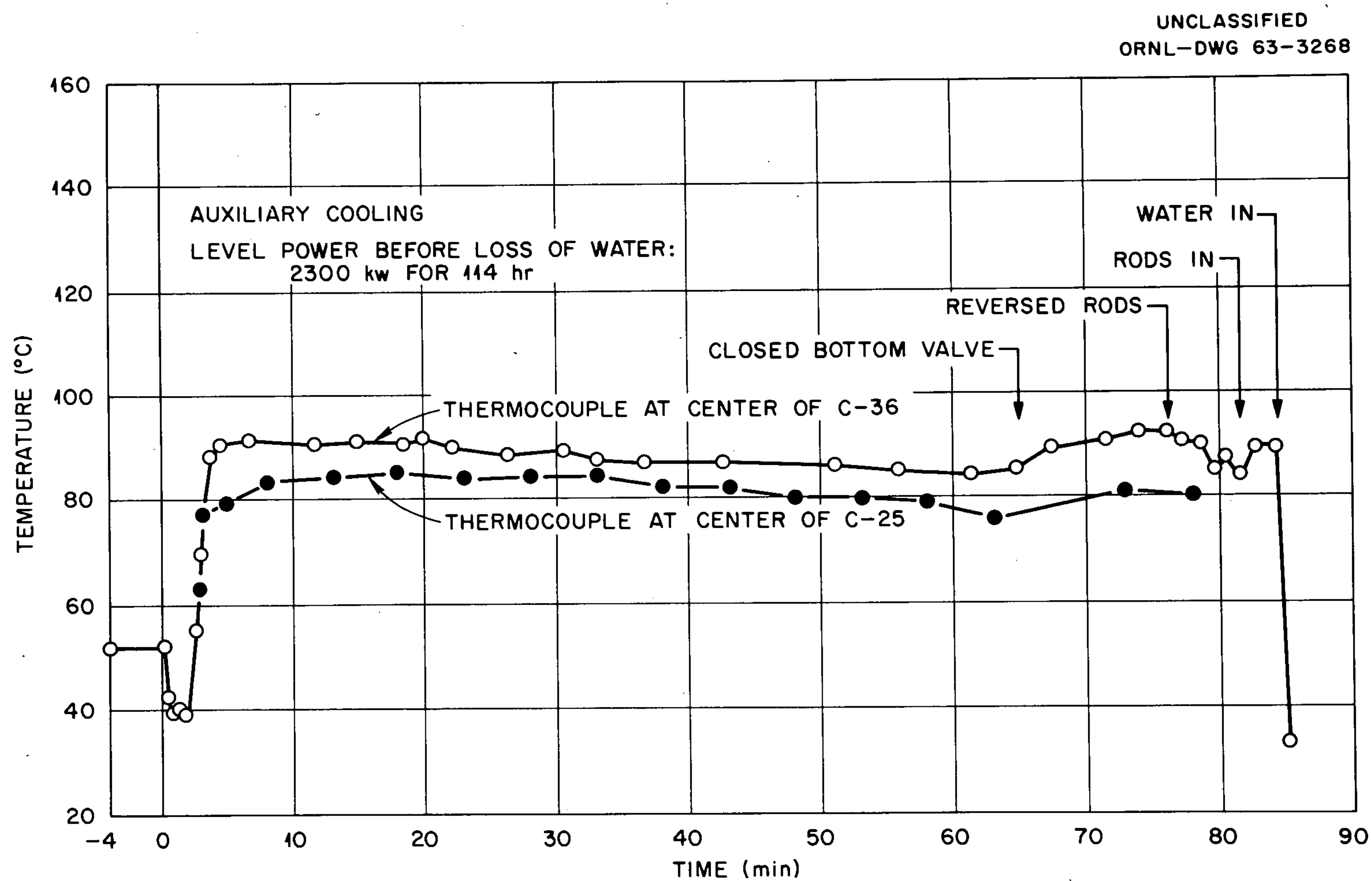

Fig. 8. Water Loss Test at LITR; 8/31/53

Time after uncovering core (min)	mr/hr
0	180
1	50
2	25
3	17
4	15
6	10

This radiation was a result of scattering from the top of the reactor back to ground level.

Envelopes containing sheets of film were placed in a number of locations around the reactor at 4:45 p.m. and removed by 6:30 p.m. Film densities were interpreted from unshielded radium gamma calibrations and indicated the following radiation values (integrated).

Radiation Dose as Determined from Film Densities

	Location	Dose Reading (mr)
1.	Control desk in front of operator (3 ft above ground level ~20 ft north of reactor).	30
2.	Center of door, south wall of control room (3 1/2 ft above ground level ~11 ft north of reactor).	30
3.	At door facing north wall of control room (3 1/2 ft above floor ~25 ft north of reactor).	30
4.	At door facing north wall of control room annex (3 1/2 ft above floor ~40 ft north of reactor).	30
5.	West side of north post at top of short stairs, midriff (~5 ft below top ~10 ft northeast of reactor).	395
6.	West side of post at southeast corner of walkway, midriff (~10 ft below top ~15 ft east of reactor).	60
7.	Door to top room (~10 ft southeast of top of reactor tank).	395
8.	South manhole cover, top plug.	455,000
9.	Under side of upper plate, top plug.	340,000
10.	North window, top room (~25 ft north and 4 ft above top of water tank).	215

A Victoreen probe with a maximum range of 6,000 r/hr was placed over the top of the north manhole cover which consists of ~1 in. of Lucite. This was connected to a recorder in the control room, and data from the chart tracing are plotted in Figure 9. However, since the instrument was not calibrated over different energy levels or to its maximum range, the results from the film located in a similar spot on the top of the reactor were used to normalize the area under the curve and to determine the maximum radiation level. The maximum radiation level measured at the top of the reactor was 1300 r/hr. This occurred immediately after dropping the water level below the level of the core. This radiation level does not represent what the reading would be from the bare fuel elements if they were completely exposed since an aluminum grid with a total thickness of about 10 in. of aluminum (but with a number of holes in it) is interposed between the fuel and the top of the reactor. A number of other mechanical parts in the reactor tank also, undoubtedly, absorbed a large portion of the radiation.

CONCLUSIONS

From the information obtained during the water-loss tests described above, it was concluded that:

1. The LITR should have the safety factor of an auxiliary cooling system in case of a rupture of the reactor tank causing the sudden loss of water in the tank when the reactor is being operated near 3000-kw power level.
2. The auxiliary spray cooling system, as used in some of the tests described above, is adequate to prevent melting of the fuel elements if there were a sudden loss of cooling water.
3. The radiation protection to personnel is adequate for any foreseeable situation.
4. An important factor in evaluating the need for auxiliary cooling is the time required to evaporate the water entrained within the reactor core when it is drained.
5. The time required to lose the water after reactor scram significantly affects the maximum temperature.

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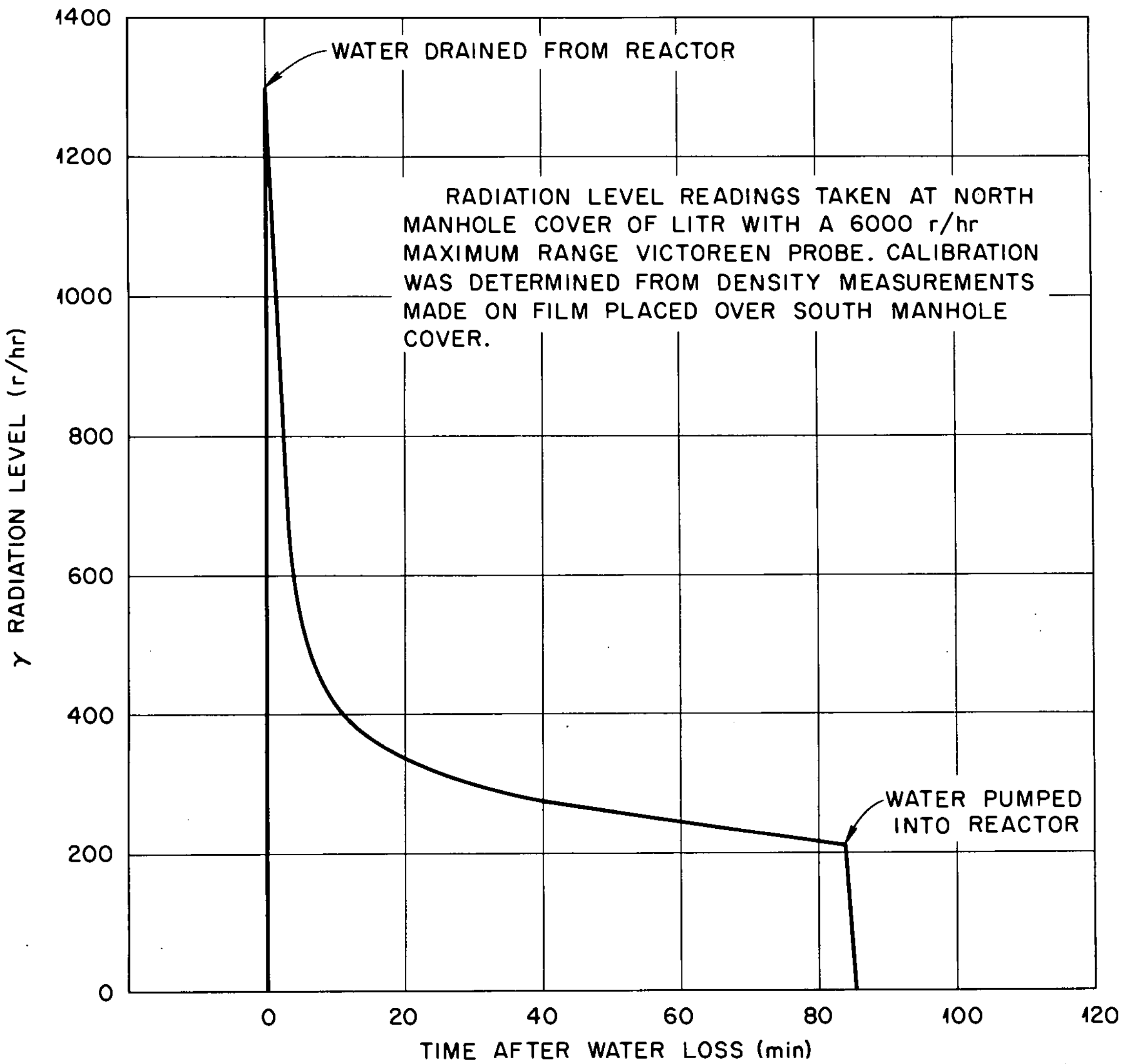

Fig. 9. Radiation Level vs Time at Top of Reactor Tank

APPENDIX A

After-Heat Generation Rate

To determine the cooling requirements when a reactor is shut down, it is necessary to know the energy released by the fission products as a function of time. The development given here is based on experimental data quoted in many reference books and handbooks.

About 6 Mev of γ-ray energy is emitted by the fission products as they decay by β emission toward stability. The rate of release of this 6 Mev for times greater than 10 sec after fission has occurred may be represented by:

$$\gamma \text{ energy} = 1.3\ t^{-1.2} \text{ Mev/sec fission} \tag{1}$$

where t is expressed in seconds. After 10 sec, about the same amount of energy is released by the β rays.

$$\gamma + \beta \text{ energy} = 2.6\ t^{-1.2} \text{ Mev/sec fission.} \tag{2}$$

The number of fissions occurring in the reactor operating at steady power P (expressed in watts) during a time interval dT (expressed in seconds) is 3.1×10^{10} P dT. The rate of release of β and γ energy at time $\boldsymbol{\tau}$ due to fissions which occurred during time interval dT where $\boldsymbol{\tau} = t + T$

$$R \cong 2.6 \times 3.1 \times 10^{10}\ P\ \boldsymbol{\tau}^{-1.2}\ dT \text{ Mev/sec watt}$$

$$\cong 8 \times 10^{10}\ P\ \boldsymbol{\tau}^{-1.2}\ dT. \tag{3}$$

Rate of emission of β and γ energy at time $\boldsymbol{\tau}$ due to fissions during reactor operation for time $T_\circ$ at fixed power P

$$R_t \cong 8 \times 10^{10}\ P \int_\circ^{T_\circ} (t + T)^{-1.2}\ dT$$

$$R_t \cong 4 \times 10^{11}\ P\ [t^{-0.2} - (t + T_\circ)^{-0.2}] \text{ Mev/sec} \tag{4}$$

where P is expressed in watts of thermal power, $T_\circ$ equals time reactor was operating, and t equals time after reactor shut down.

$$1 \text{ Mev} = 1.5 \times 10^{-13} \text{ watt-sec}$$

$$R_t \cong 6.4 \times 10^{-2}\ P\ [t^{-0.2} - (t + T_\circ)^{-0.2}] \text{ watts,} \tag{5}$$

where $t > 10$ sec.

At times less than 10 sec, the rate of γ energy release is somewhat greater than at 10 sec.

Equation (5) represents the fission-product heat-generation rate after reactor shutdown for times greater than 10 sec.

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