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LOSS-OF-COOLANT ACCIDENT EXPERIMENT AT THE AVR GAS-COOLED REACTOR

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ABSTRACT

Loss-of-coolant is one of the most severe accidents for a nuclear power plant. To demonstrate inherent safety characteristics incorporated into small High-Temperature Gas-Cooled Reactor (HTGR) designs, loss-of-coolant accident (LOCA) simulation tests have been conducted with the German pebble-bed High-Temperature Reactor AVR. The AVR is the only nuclear power plant ever to have been intentionally subjected to LOCA conditions.

The LOCA test was planned to create conditions that would exist if a rapid LOCA occurred with the reactor operating at full power. The test demonstrated this reactor's safe response to an accident in which the coolant escapes from the reactor core and no emergency system is available to provide coolant flow to the core. The test is of special interest because it demonstrates the inherent safety features incorporated into modular HTGR designs.

The main LOCA test lasted for 5 d. After the test began, core temperatures increased for ~13 h and then gradually and continually decreased as the rate of heat dissipation from the core exceeded accident levels of decay power. Throughout the test, temperatures remained below limiting values for the core and other reactor components.

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INTRODUCTION

Relative to earlier nuclear power development, both in the United States and in the Federal Republic of Germany (FRG), a fundamentally different approach is taken by modular gas-cooled reactors for ensuring economical and safe power generation. This new approach is to design the reactor system to rely primarily on its inherent characteristics for safety and thus to reduce the number of active engineered-safety systems that are needed. Such characteristics can be confirmed by test.

In the FRG, the 200-MW(t), MODUL high-temperature reactor (HTR) has been developed by Interatom (Siemens). The Hochtemperatur-Reaktorbau GmbH (HRB) has developed a 258-MW(t) design. Research and development activities are conducted at Kernforschungsanlage-Jülich GmbH (KFA). In the United States, the Department of Energy HTGR Program is developing a 350-MW(t), Modular High-Temperature Gas-Cooled Reactor (MHTGR). The U.S. development team consists of General Atomics, Combustion Engineering Inc., Bechtel, Stone and Webster, Oak Ridge National Laboratory (ORNL) and Gas-Cooled Reactor Associates.

While the AVR is a relatively low-power-level reactor [46 MW(t)], it possesses inherent safety characteristics also incorporated into modular gas-cooled reactors. These are:

- A chemically inert coolant (helium) that will not react with fuel under any circumstances.
- Refractory-coated particle fuel that is capable of withstanding very high temperatures (1600 to 1800°C). Modular HTR concepts are designed to remain below these temperatures in case of LOCA.
- A negative temperature coefficient of reactivity that results in a shutdown of the nuclear fission chain reaction as core temperatures increase.
- A low power density and a high heat capacity leading to very slow thermal response during transients. Because this high heat capacity is a property of the core material and not the coolant, this inherent feature is retained with loss-of-coolant.
- The ability to passively dissipate decay heat from the core.

The LOCA tests were performed to

1. demonstrate that temperatures in the fuel and the reactor remain well below limiting values for fission product release and component damage without active components operating in the primary system,

2. confirm that the temperature coefficient of reactivity is negative at temperatures that exist during accident conditions, and

3. obtain data at accident conditions to be used by gas-cooled reactor designers to help validate computer codes used for safety analyses.

ORNL participation in the preparation and conduct of the tests was carried out within the U.S./FRG Agreement for Cooperation in Gas-Cooled Reactor Development.

SUMMARY DESCRIPTION OF THE AVR REACTOR

The AVR is a pebble bed demonstration HTGR plant in Jülich, West Germany that began generating electricity in December 1967. Its purpose is to demonstrate the feasibility of an HTGR with pebble fuel elements and high operating temperatures. The operating utility group is Arbeitsgemeinschaft Versuchsreaktor (AVR) GmbH. The lead constructor was Brown-Boveri-Krupp Reaktorbau GmbH.

The AVR is shown schematically in Fig. 1, and technical information is summarized in Table 1. The core is fueled with about 100,000 graphite pebbles containing coated fuel particles. During operation, the 6-cm-diam pebbles are continuously withdrawn from the bottom of the reactor core and their burnup is measured individually. Those with the highest burnup are removed from the cycle. The others are filled back on to the core together with some fresh fuel elements. Every element passes through the core several times until it reaches the target burnup.

Helium flows upward through the pebble bed and then across the steam generator tubes, which are located above the core in the steel reactor vessel. The steam generator is shielded from core radiation by a 50-cm-thick graphite top reflector and two additional 50-cm-thick carbon brick layers. The helium is circulated by two blowers located in the lower part of the vessel.

The AVR has four control rods in reflector "noses," which protrude into the core region (Fig. 2). The rods enter from below and in the event of a scram are driven upward by a counterweight device.

The inner vessel is enclosed concentrically in a second reactor vessel. A biological shield is located between the vessels. The space between the vessels is cooled by natural convection of helium at a slightly higher pressure than the primary coolant. Heat is removed from the interspace helium by a water-cooled heat exchanger (Fig. 1). The reactor is surrounded by a steel containment vessel and by a 1.5-m-thick concrete building.

DEVELOPMENT OF THE LOCA TEST PLAN

The LOCA test was planned to create conditions that would exist if the accident occurred with the reactor initially operating at full power. This test was planned to simulate a very rapid LOCA during which the primary system pressure approaches 1 bar essentially instantaneously. The AVR was not designed so that the helium could be removed rapidly for a test. To remove the coolant for the test by normal depressurization (pumping helium into storage tanks) requires about 3 d. During this time the decay heat generation rate decreases to levels that are much lower than would exist immediately following an actual rapid LOCA – making a test at these conditions uninteresting and unrealistic.

Thus, a test plan was devised to establish steady-state full-power operating temperatures using fission power following a normal shutdown, cooldown, and depressurization. Then, during the accident simulation phase of the test, fission power would be generated so that the total power during the accident simulation matches realistic accident levels of decay power. In this way the thermal response of the reactor would be similar to that occurring in an actual LOCA.

The test plan consisted of these steps:

- For several weeks before a planned shutdown, core reactivity would be decreased by not adding fuel. By this technique the fission power during the experiment could be controlled to the desired levels with the control rods nearly withdrawn from the core. This step allows accurate determination of fission power in the 1 to 10% range during the test. With the core at high temperatures and the control rods mostly or completely withdrawn, a proportionality between the signal of the flux detectors and the fission power was found. In this way the control rods could be in a relatively cool core position so that their temperatures would not approach the normal licensed limit during the test.
- Shut down the reactor to cold shutdown conditions (130°C) by fully inserting the control rods and cooling the core with forced convection of helium.
- Depressurize the reactor to 1 bar by pumping the helium to storage tanks.

- Establish temperatures representative of full-power operation by heating the core with fission heat.
- Simulate the accident by varying the nuclear fission power with time so that the total power matches accident levels of decay power. Water flow to the steam generator is continued during the LOCA test to protect the tubes from possible damage by overheating.

The goal was to continue the accident simulation until all measured temperatures (in the top, bottom, and side reflectors, in the reflector noses, and reactor vessel) had been decreasing for quite some time.

To limit the effort required to license the test, component temperature limits approved for normal operation also applied for the test, for example:

- fuel – 1250°C,
- control rods – 700°C (900°C for short time in case of a scram), and
- inner vessel – 325°C.

PREPARATIONS AND PRETESTS

Dynamic heat transport analyses were performed for the licensing effort to predict the temperature distribution throughout the reactor during the LOCA test. Two-dimensional analyses were performed with the THERMIX-KONVEK code,[1] developed by KFA, to predict temperatures throughout the system. THERMIX-KONVEK performs a 2-D calculation of solid body temperatures which is coupled to a 2-D calculation of the gas flow, pressure, and fluid-temperature distribution.

The THERMIX-KONVEK model included the core and surrounding graphite structure, the steam generator, the thermal shield, inner vessel, biological shield and interspace cooling system, and the outer vessel. For the licensing process, certain conservative assumptions were employed. Adiabatic boundary conditions were applied to the bottom of the gas inlet plenum and at the gas outlet position of the steam generator. Also, feedwater temperature to the steam generator was assumed to remain at 130°C throughout the accident simulation, while in fact the feedwater temperature decreased to 60°C during the test.

Three dimensional analyses were performed with the ORNL HEATING-6 heat transport code[2] to predict temperatures in the core, reflectors, and reflector noses during the test, and in the control rods in the event of a scram. Time-dependent temperatures at the outer surfaces of the reflectors determined by the

THERMIX-KONVEK computations were applied as boundary conditions in the HEATING-6 analyses. Conservatisms discussed above for the THERMIX-KONVEK analyses were transferred to HEATING-6 through these boundary conditions.

Results of the pretest analyses (Figs. 3 and 4) showed that the highest control rod temperatures would be reached if the scram occurred at the beginning of the accident simulation and that these temperatures would remain below 900°C. THERMIX-KONVEK and HEATING-6 analyses corroborated that neither the fuel nor any other component would reach its temperature limit under the conditions planned for the test. A more detailed discussion of the analytical models and predictions is presented in Ref. 3.

Pretests were conducted in May 1986 and January 1987 to carry out portions of the LOCA test plan.

Conclusions drawn from these pretests were:

- Simulation of the time dependence of decay power with fission power could be performed without difficulty.
- An initial temperature distribution representative of full power operation could be established by heating the core with about 4 MW(t) of fission power and with forced convection of helium at depressurized conditions. During this heatup, the circulators were operated at ~85% speed (3000 rpm) and heat was removed from the system by the steam generator. Heatup by this technique required ~3 d.

Other preparations for the LOCA test involved installing additional thermocouples on the outer surface of the inner vessel. Also, a small gas circulator was added in the interspace cooling loop. This circulator could be used to provide a well-defined temperature boundary condition on the outer surface of the inner vessel (it was later determined not to use this option) and to provide a means of cooling the inner vessel if its temperature approached its licensed limit (325°C) during the test.

CONDUCT OF TEST AND RESULTS

Two LOCA tests were conducted in May 1988. The first was conducted with the main circuit valves closed, and the second with these valves open. With all other conditions the same, results would indicate the effectiveness of natural convection of helium through the main loop. In both tests, nose temperatures peaked at ~14 h and then slowly decreased (see Fig. 5). Measured temperatures in the reflector nose and in the side reflector during the two tests differed by <10 K, indicating that heat transport by natural convection through the main loop was insignificant. To maintain criticality, the control rods were continually withdrawn during the initial 12 h as the core heated up. After that time they were driven farther into the core as its temperature began a slow and gradual decrease. After 12 h, control rods were therefore moved into a hotter region of the core. A conservative correlation was used to infer control rod temperatures from control rod position, from measured temperatures in the reflector noses and in the bottom reflector, and from computed axial temperature profiles. It was necessary to terminate both tests as this inferred temperature approached the control rod temperature limit for normal operation of 700°C. In the first test, the point at which forced convection cooling was necessary to maintain control rod temperatures below 700°C was reached after 28 h. This required stopping the experiment by fully inserting the control rods into the core and starting forced convection cooling. To extend the test duration for the second test, a plan was devised to inject nitrogen into the primary system to introduce negative reactivity requiring less insertion of the control rods in the core. The test duration was extended to 37 h by injection of 0.5 bar of nitrogen.

Experience gained during these two tests conducted in May 1988 contributed to the success of later tests. The following approaches were identified as having potential for extending test duration well beyond 37 h:

- injection of more nitrogen,
- development of a more accurate method for inferring control rod temperature, and
- requesting licensing approval for higher control rod temperatures during the test.

The goal was to find a way to extend the test until all temperatures had been decreasing for a significant period. The approach taken was to develop a more realistic method for predicting control rod temperature while maintaining the option of nitrogen injection as a backup. Additional main tests were planned for October 1988.

The most significant test with the maximum temperatures was initiated on October 14 and lasted through October 19. Preparation for this test began in July 1988, when 110 temperature monitor elements were loaded into the core during normal operation. Monitor elements are graphite pebbles that are the same size as normal fuel pebbles. Each element contains 20 quartz capsules with wires of different metal alloys with melting temperatures ranging from 655°C to 1280°C. The elements record the highest temperature reached while they pass through the core. Upon discharge the monitor elements are distinguished from other fuel-free elements by their high silver activity (many of the metal alloys contain silver). In the hot cells they are x-rayed to determine the number of wires that melted, so the maximum temperature that they reached can be inferred.

From July 1988 to October 1988 the fuel was cycled as normal, and by October 2 some of the monitor elements were near core midheight. Beginning in mid-September the reactor was operated without loading fresh fuel to reduce excess core reactivity, so the control rods could be nearly withdrawn during the LOCA test.

The test was initiated as planned following a 4-MW(t) heatup phase with fission heat at depressurized conditions and with forced convection to establish a proper initial temperature distribution. To initiate the accident simulation phase of the test, the gas circulators were stopped and the fission heat was varied so that the total heat matched accident levels of decay heat. The fission power generation and total power simulating accident levels of decay power are shown in Fig. 6.

Figure 7 shows the starting temperature level and the temperature development during the test for a number of thermocouples at key positions in the reactor structure. The thermocouples in the reflector noses at core midheight, which are the nearest to the core center, showed a maximum rise of 300 K. They reached their maximum of 860°C at ~12 to 13 h after accident initiation. Further to the outside, at core midheight, peak temperatures occurred later; in the side reflector after 20 h and at the reactor shroud after 34 h. The inner reactor vessel temperature showed a rise of only 15 K at that height.

In the axial direction an enormous temperature shift took place. Whereas temperatures in the upper core region decreased rapidly from the very beginning (see nose temperature above pebble bed), the bottom reflector experienced a steady long-term temperature rise that peaked at 500°C 65 h after accident initiation. Thus the initially cold lower part of the core functions as a longer-term intermediate storage for a considerable amount of heat.

Measured temperature results have been obtained for eight monitor elements which, during the test, were on or near the core axis and only 20 to 25 cm above the core center. These elements all recorded 1080°C ± 10°C. Because of the predicted flat temperature profile near the core center at the time of maximum core temperature, these measured values should be within 5 to 10°C of the maximum core temperature.

The amount of reactivity balanced by the control rods during the test (Fig. 8) gives, due to the negative temperature coefficient of reactivity, a rough image of the inverse course of the average moderator temperature. Small reactivity changes due to xenon decay and to the temperature dependence of the temperature coefficient are of minor importance. The point of greatest reactivity demand with the rods mostly withdrawn occurred 8 h after accident initiation. Thus, with regard to the xenon effect, the maximum of the average moderator temperature is estimated to have occurred after 10 to 12 h, which is not much earlier than that of the previously noted midheight nose temperature.

Figure 9 shows a first comparison between measured temperatures during the October test and computed predictions for the licensing effort. It is obvious that the previously discussed conservative assumptions lead to higher predicted temperatures at these measurement locations in the longer term. More sophisticated computational analysis has to be done to quantify the heat transport in terms of a best-estimate.

CONCLUSIONS

LOCA tests performed with the German pebble-bed high-temperature reactor AVR have demonstrated that temperatures in the fuel and the reactor components remain well below limiting values. The tests confirmed that the temperature coefficient of reactivity is negative at all temperatures during LOCA conditions.

In this test heat was removed from the core by passive means. Heat was transported to the steam generator by natural convection and radiation. In the German MODUL design, the heat would be dissipated from the core through the reactor vessel to natural circulation, water-cooled coils and then to the environment. In the U.S. MHTGR design the heat would be dissipated from the core through the reactor vessel to natural circulation, air-cooled panels and then to the environment.

The significance of this AVR test is that inherent safety characteristics have now actually been demonstrated for a LOCA — this is no longer just a computer prediction. Important data were obtained that now can be used by gas-cooled reactor designers to help validate computer codes used for safety analyses.

Experience gained through the AVR LOCA test can be useful in establishing a testing approach for future modular gas-cooled reactors so that their safe response to accidents can be also demonstrated. Such a demonstration of reactor safety – showing that the reactor can survive very unlikely accident conditions without release of radioactivity and without damage – should instill a high degree of public confidence in this fundamentally new approach to safe and economical nuclear power.

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Table 1. Technical data for AVR

Thermal power rating	46 MW
Electric power	15 MW
Average core power density	2.6 MW/m^3
Core inlet temperature	275°C
Average core outlet temperature	950°C
Mass flow	13.1 kg/s
Primary system pressure	10.8 bar
Core diameter	3.0 m
Average core height	2.8 m
Steam pressure	73 bar
Steam temperature	505°C
Absorber rods	4

FIGURE CAPTIONS

Fig. 1. AVR reactor arrangement.

Fig. 2. View upward in empty core of AVR showing graphite reflector and reflector noses.

Fig. 3. AVR/ORNL pretest prediction of maximum fuel and nose temperatures during LOCA test.

Fig. 4. AVR/ORNL prediction of control rod temperatures in event of SCRAM at initiation of LOCA test.

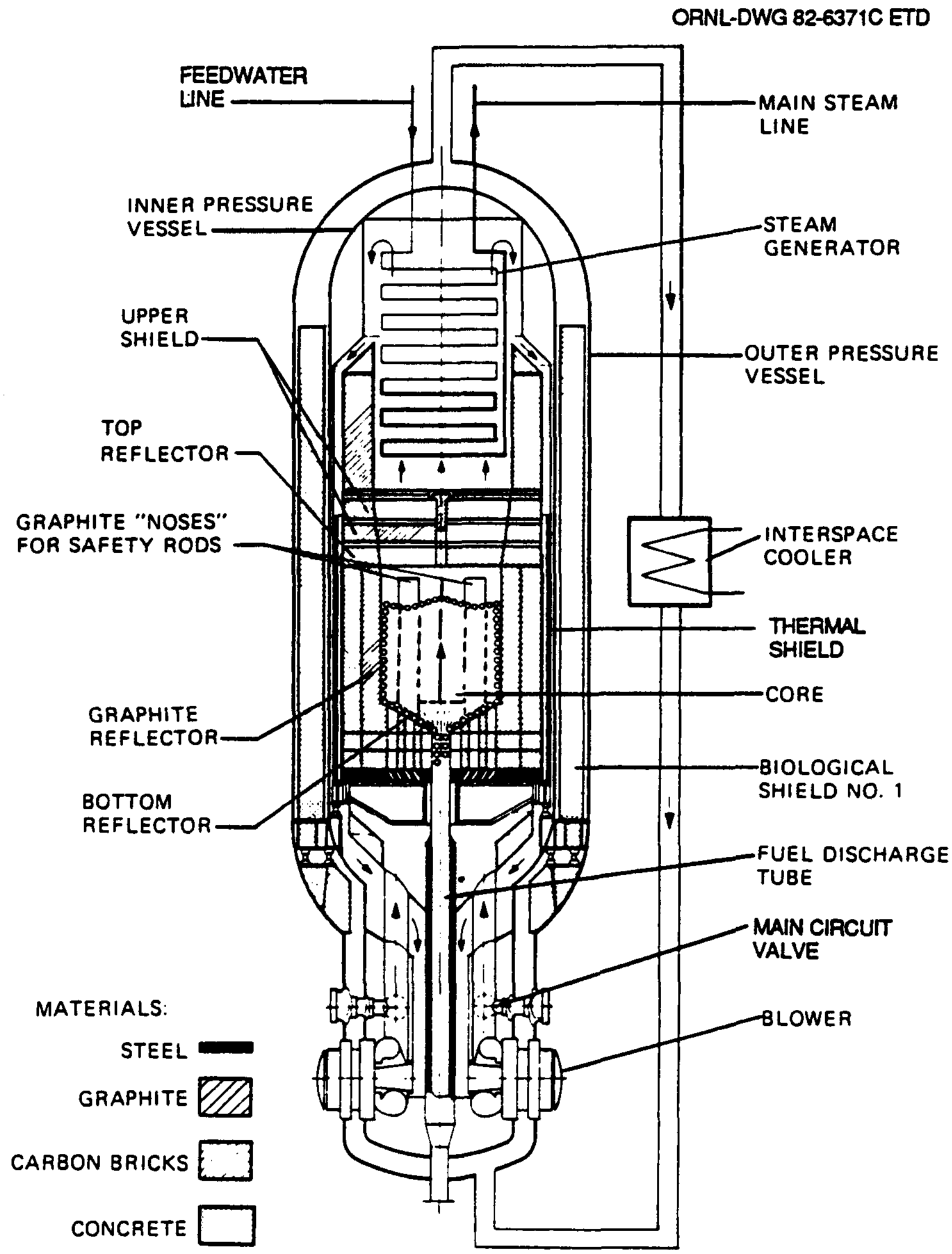
Fig. 5. Temperature measurement results of LOCA test conducted May 27-28, 1988.

Fig. 6. Simulation of accident levels of decay power (heat) with fission power.

Fig. 7. Temperature measurement results of LOCA test conducted Oct. 14-19, 1988.

Fig. 8. Measured reactivity held by control rods during LOCA test conducted Oct. 14-19, 1988.

Fig. 9. Comparison of AVR/ORNL pretest predictions with measured results of LOCA test.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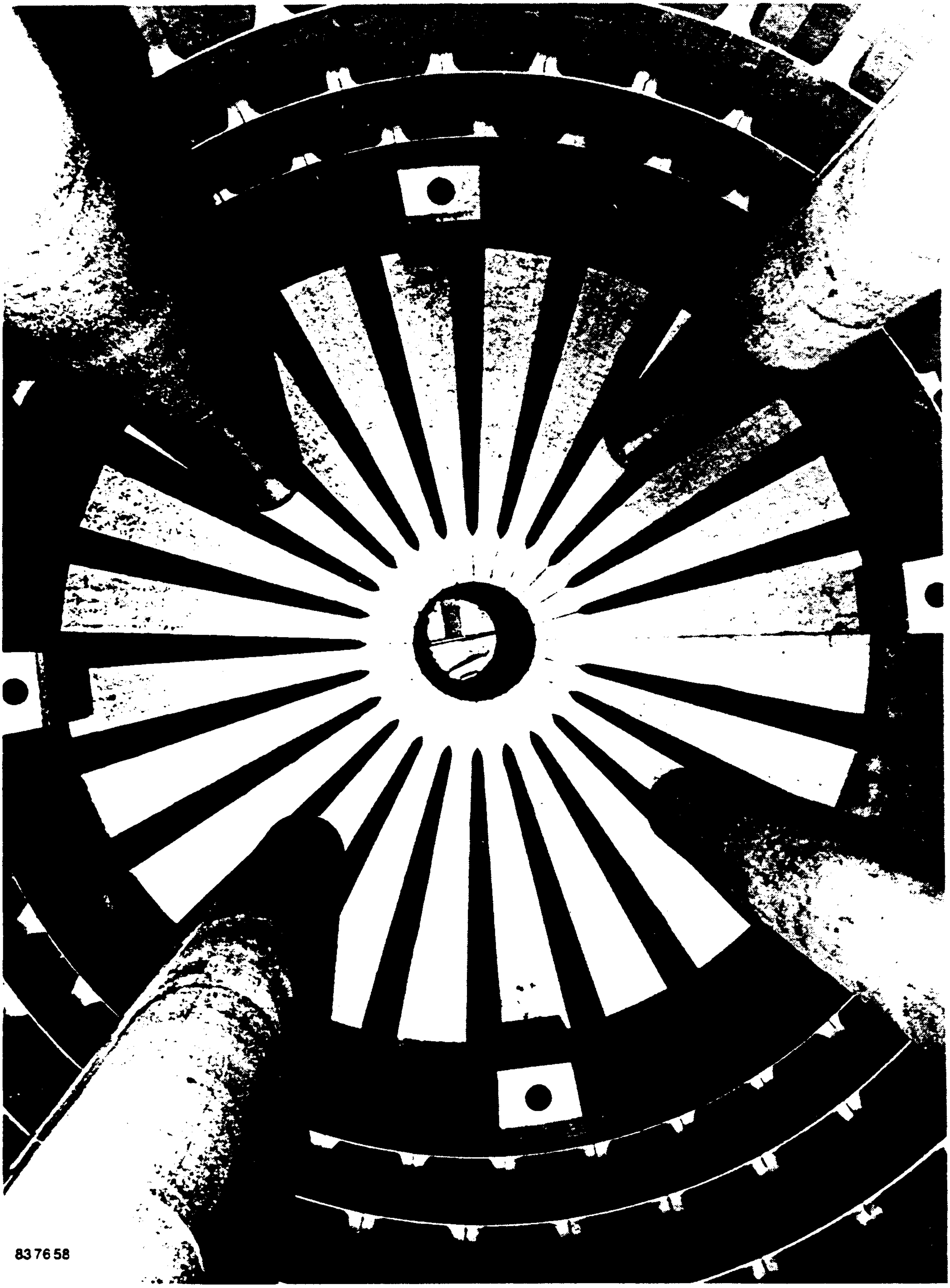
MAIN STEAM LINE
INNER PRESSURE VESSEL
STEAM GENERATOR
UPPER SHIELD
OUTER PRESSURE VESSEL
TOP REFLECTOR
GRAPHITE "NOSES" FOR SAFETY RODS
INTERSPACE COOLER
THERMAL SHIELD
CORE
GRAPHITE REFLECTOR
BIOLOGICAL SHIELD NO. 1
BOTTOM REFLECTOR
FUEL DISCHARGE TUBE
MAIN CIRCUIT VALVE
BLOWER
MATERIALS:
STEEL
GRAPHITE
CARBON BRICKS
CONCRETE



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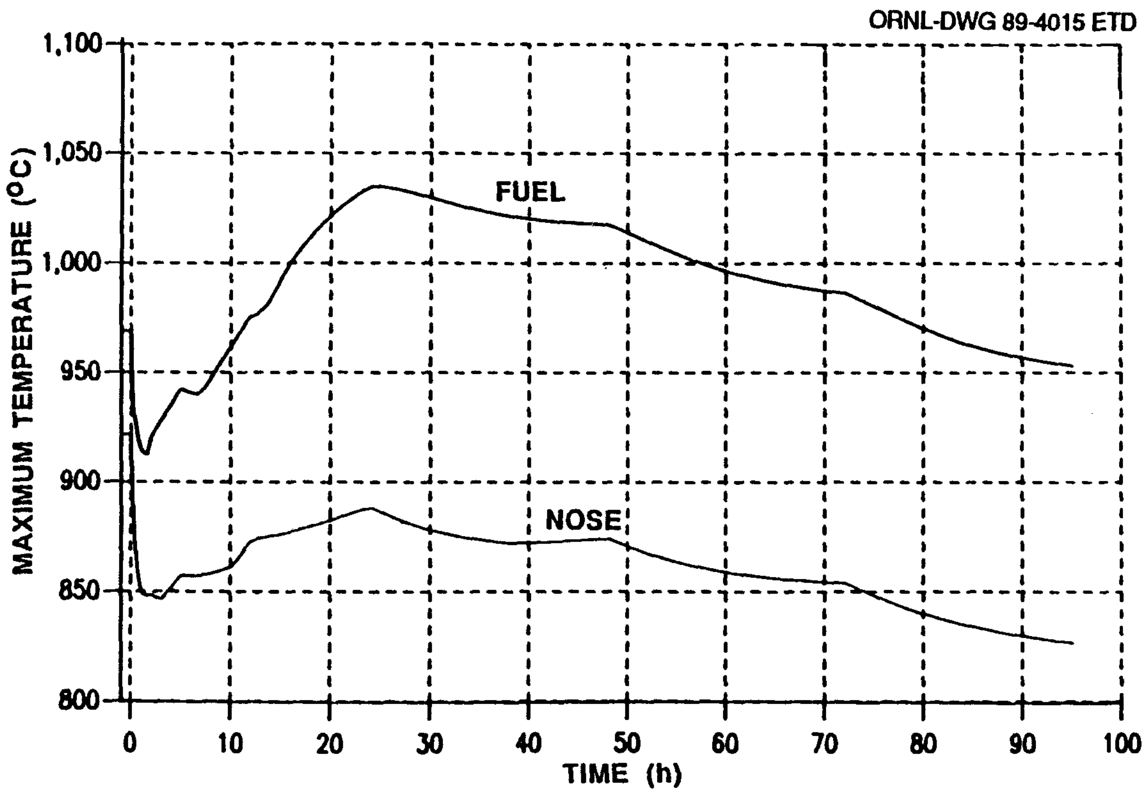
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1,050
1,000
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900
850
800
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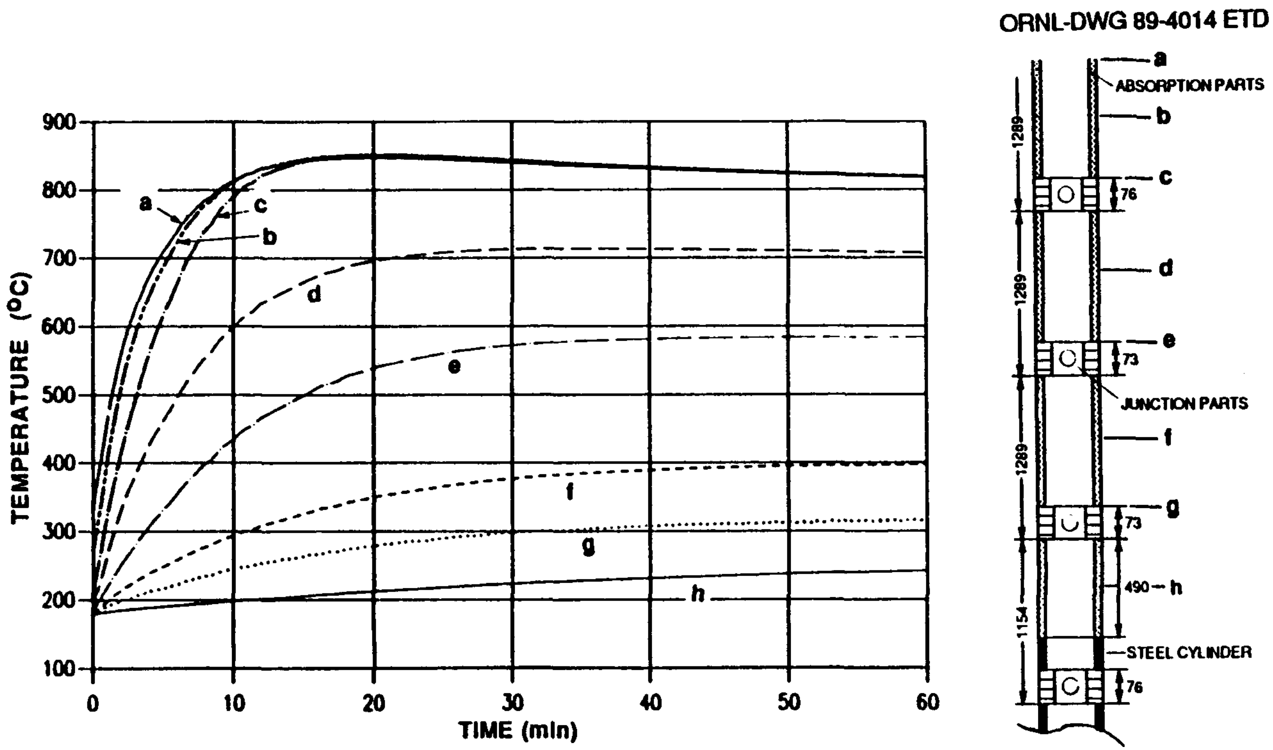
0
10
20
30
40
50
60
70
80
90
100
TIME (h)



ORNL-DWG 89-4014 ETD
TEMPERATURE (°C)
TIME (min)
a
b
c
d
e
f
g
h
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JUNCTION PARTS
STEEL CYLINDER
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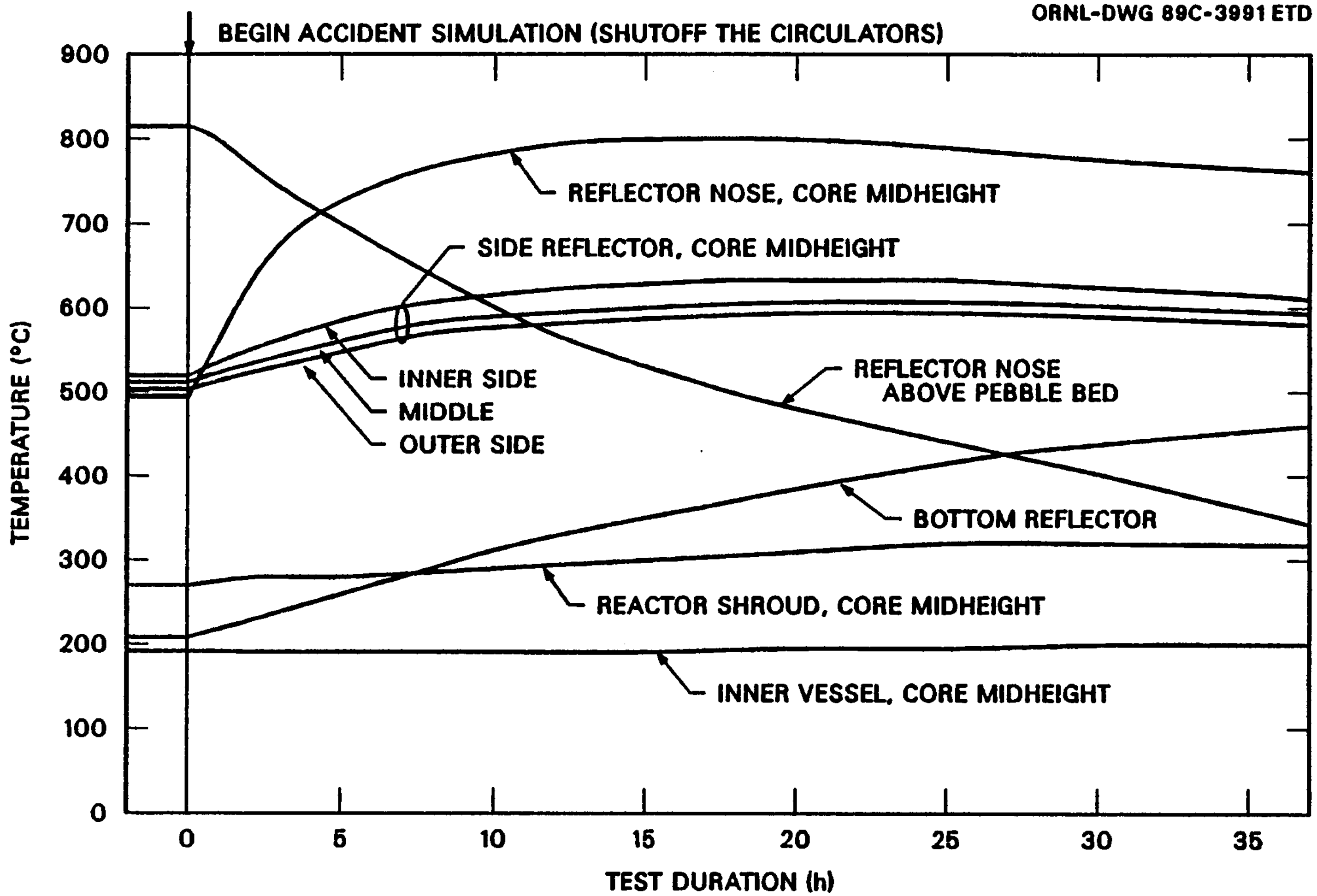
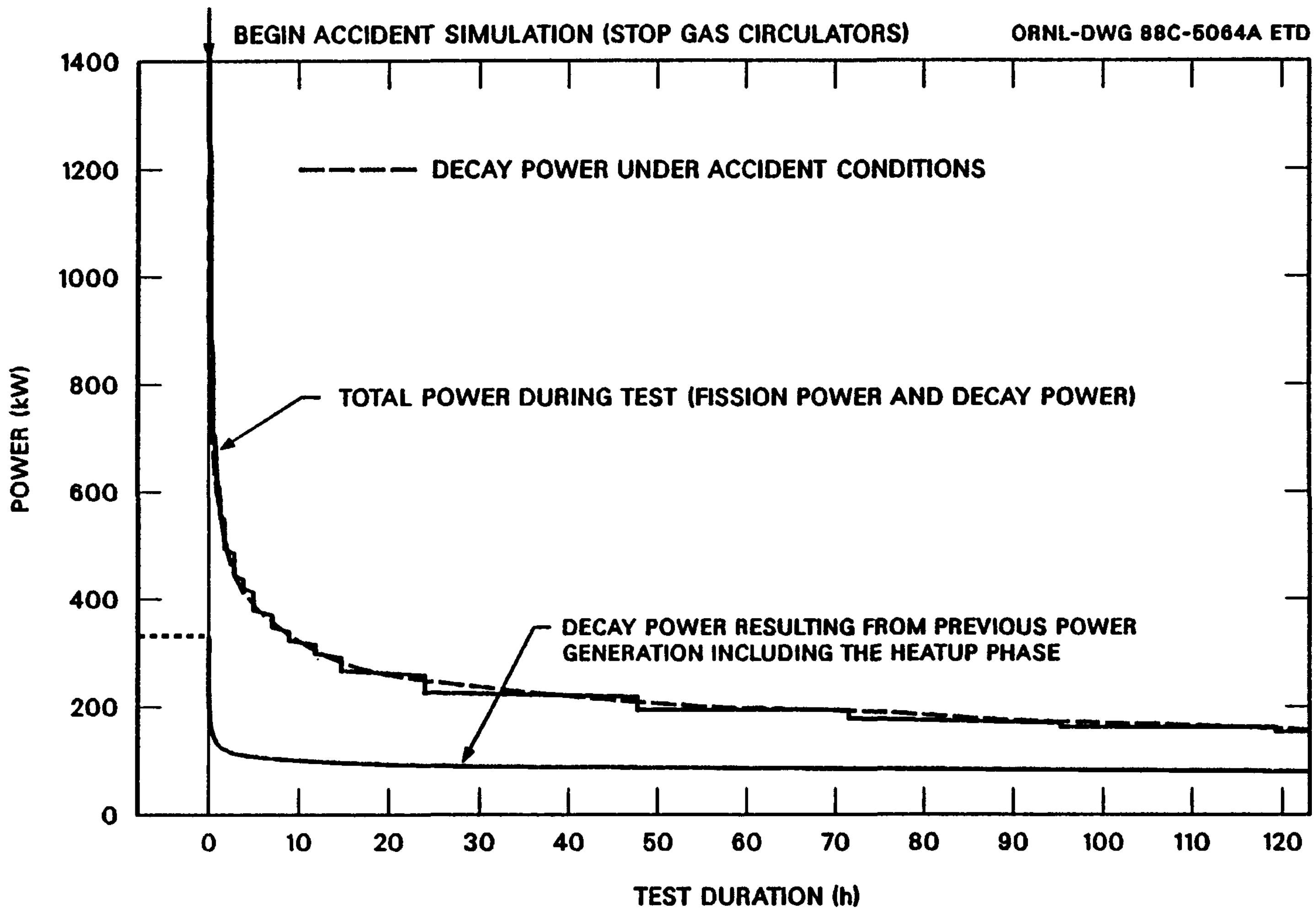
76
73
490
1154


ORNL-DWG 89C-3991 ETD
BEGIN ACCIDENT SIMULATION (SHUTOFF THE CIRCULATORS)
900
800
700
600
500
400
300
200
100
0
TEMPERATURE (°C)
0
5
10
15
20
25
30
35
TEST DURATION (h)
REFLECTOR NOSE, CORE MIDHEIGHT
SIDE REFLECTOR, CORE MIDHEIGHT
INNER SIDE
MIDDLE
OUTER SIDE
REFLECTOR NOSE ABOVE PEBBLE BED
BOTTOM REFLECTOR
REACTOR SHROUD, CORE MIDHEIGHT
INNER VESSEL, CORE MIDHEIGHT



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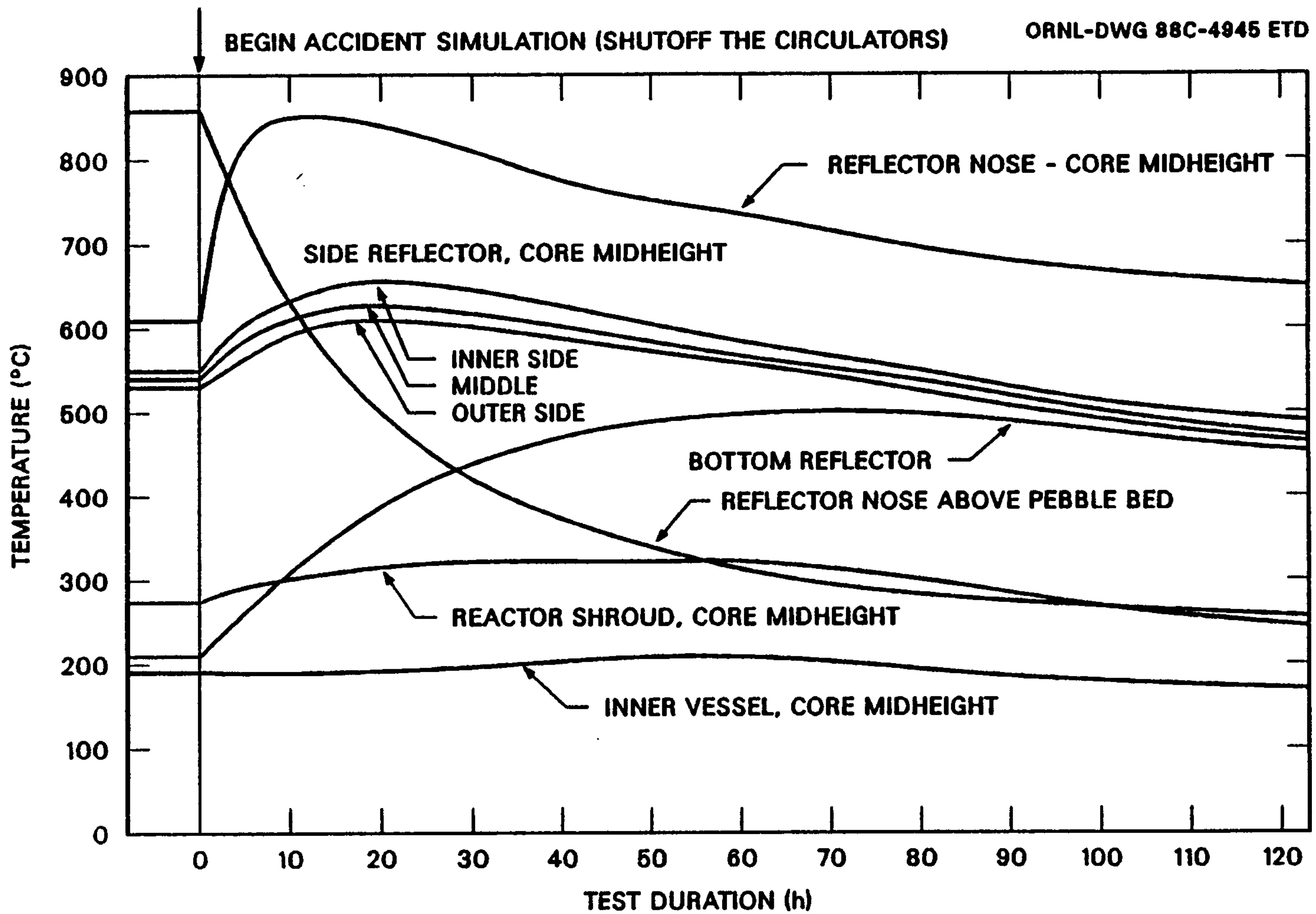
DECAY POWER UNDER ACCIDENT CONDITIONS
TOTAL POWER DURING TEST (FISSION POWER AND DECAY POWER)
DECAY POWER RESULTING FROM PREVIOUS POWER GENERATION INCLUDING THE HEATUP PHASE
POWER (kW)
TEST DURATION (h)
0
200
400
600
800
1000
1200
1400
0
10
20
30
40
50
60
70
80
90
100
110
120



ORNL-DWG 88C-4945 ETD
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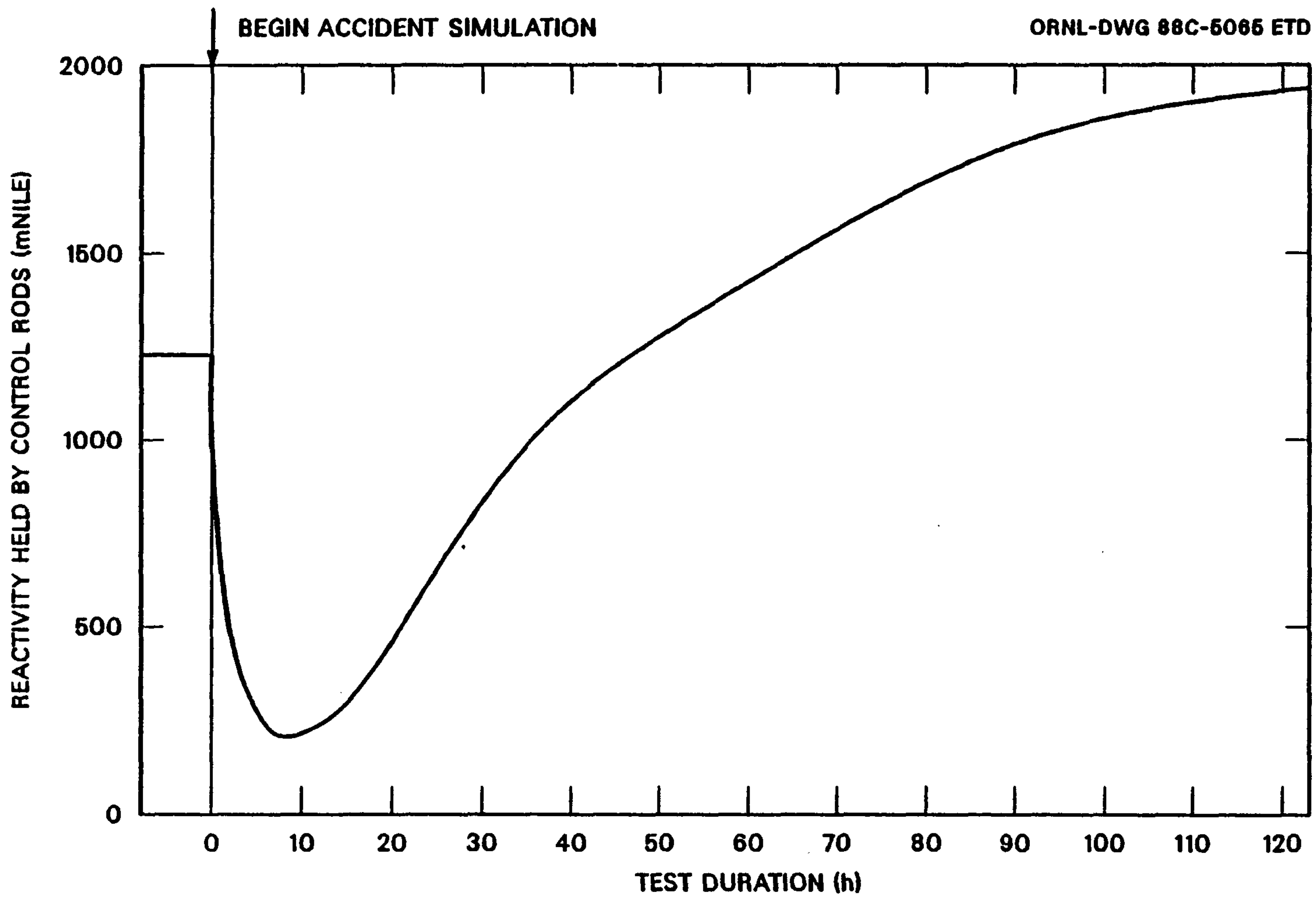
REFLECTOR NOSE - CORE MIDHEIGHT
SIDE REFLECTOR, CORE MIDHEIGHT
INNER SIDE
MIDDLE
OUTER SIDE
BOTTOM REFLECTOR
REFLECTOR NOSE ABOVE PEBBLE BED
REACTOR SHROUD, CORE MIDHEIGHT
INNER VESSEL, CORE MIDHEIGHT
TEMPERATURE (°C)
TEST DURATION (h)
0
100
200
300
400
500
600
700
800
900
0
10
20
30
40
50
60
70
80
90
100
110
120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REACTIVITY HELD BY CONTROL RODS (mNILE)
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1500
1000
500
0
0
10
20
30
40
50
60
70
80
90
100
110
120
TEST DURATION (h)

ORNL-DWG 88C-5069A ETD