DP - 465

Reactors - Power

AEC Research and Development Report

HEAVY WATER MODERATED POWER REACTORS

PROGRESS REPORT
January 1960

Technical Division
Wilmington, Delaware

February 1960

E. I. du Pont de Nemours & Co.
Explosives Department - Atomic Energy Division
Technical Division - Wilmington, Delaware

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Progress Report
January 1960

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Wilmington, Delaware

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ABSTRACT

Approximately one-quarter of the construction and 85% of the firm design of the Heavy Water Components Test Reactor (HWCTR) were complete at the end of January 1960. Safeguards analyses of the liquid-D_2O-cooled loop of the HWCTR showed that none of the accidents considered to date have serious potential. Exploratory tests of a device for quenching the steam that would be generated in the boiling-D_2O-cooled loop of the HWCTR showed that a quencher could be designed to operate satisfactorily without excessive accompanying noise or vibration. Two Zircaloy-clad tubes of crushed, fused uranium oxide were cold swaged to a density of greater than 90% of theoretical. Several other cold-swaged oxide tubes clad with stainless steel were fabricated for irradiation specimens. Mechanical, hydraulic pressure, thermal- and pressure-cycling tests of tubular metallurgical joints between Zircaloy and stainless steel, fabricated by Nuclear Metals, Inc., continued to show excellent properties.

CONTENTS

LIST OF FIGURES

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INTRODUCTION

This report is one of a series that records the progress of the du Pont study of heavy-water-moderated, natural-uranium-fueled power reactors. The current effort is divided into two main categories: (1) the research and development required for the successful design, construction, and operation of the Heavy Water Components Test Reactor (HWCTR), a high temperature fuel irradiation facility, and (2) the experimental and theoretical studies required for developing the technology of a full-scale D_2O-moderated power reactor plant. Earlier reports on this study are:

DP-232	DP-315	DP-405
DP-245	DP-345	DP-415
DP-265	DP-375	DP-425
DP-285	DP-385	DP-435
DP-295	DP-395	DP-445
		DP-455

Progress for the month of February 1960 will be reported in DP-475.

SUMMARY

At the end of January 1960, approximately one-quarter of the construction and 85% of the firm design of the Heavy Water Components Test Reactor (HWCTR) were complete.

Safeguards analyses of the liquid-D_2O-cooled loop of the HWCTR showed that none of the accidents considered to date have serious potential. Among the accidents considered were (1) the cessation of H_2O flow on the shell side of the loop heat exchanger, (2) the failure of the AC power supplied to the loop D_2O coolant pump, and (3) inadvertent changes in the amount of coolant D_2O bypassing the loop heat exchanger. The results of the computations are presented in Figures 1 through 9.

Exploratory tests to provide preliminary information for the design of a steam quencher for the boiling-D_2O-cooled-loop of the HWCTR showed that a perforated sparger will probably perform satisfactorily in the loop without excessive noise or vibration accompanying the quenching.

Further analysis was made of the measurements of the effects of moderator temperature on a mockup of the HWCTR in the pressurized subcritical facility (PSE). This analysis indicated that a hot-and-dirty HWCTR lattice will have a slightly longer reactivity lifetime than had been calculated previously.

The program to determine the leakage of D_2O from typical reactor components, such as pump seals and valve stem closures, continued at the Savannah River Laboratory. Unrecoverable losses from these components could have considerable effect on the economics of D_2O-moderated power. To date, however, results of SRL tests and of tests conducted by Sargent & Lundy have shown that it is possible to limit the unrecoverable losses of D_2O from these sources to a tolerable level.

Additional tubular specimens of stainless-steel-clad uranium oxide for irradiation testing were fabricated at the Savannah River Laboratory by cold swaging. The tubes are approximately 1-1/2 inches in ID and 2.0 inches in OD and the cladding thickness is 0.022 inch. Irradiation of an assembly with five such tubes, each 2 feet long, was begun this month in a Savannah River reactor. In addition, two 5-foot-long, Zircaloy-clad tubes of uranium oxide of the same diameters as the stainless steel specimens were swaged successfully through a 37% area reduction. This reduction in area produced an average oxide density of 91.3% of theoretical density. The fabrication behavior of the Zircaloy-clad tubes was quite similar to that of the stainless-steel-clad tubes. However, the two Zircaloy-clad tubes stuck to the mandrels during swaging because of insufficient lubrication of the mandrel. No evidence of cracks was found on the inside surfaces of the sheaths of one of the tubes that had been cut from the mandrel. The other tube was removed from the mandrel by means of special tools attached to a drawbench.

Mechanical, hydraulic pressure, and thermal and pressure cycling tests of tubular metallurgical joints between Zircaloy and stainless steel continue to show promising results. Fabrication development of these pressure tube connections at Nuclear Metals, Inc., is now being concentrated on the joints that to date have exhibited the best corrosion resistance. The mechanical properties of all the NMI-fabricated joints tested thus far have been satisfactory.

DISCUSSION

I. HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

The HWCTR is a test reactor in which numerous fuel elements will be irradiated under conditions of temperature, pressure, and neutron flux that are typical of those expected in D_2O-moderated power reactors. A description of the reactor was presented in DP-383[1] and in earlier progress reports. Construction of the facility was authorized by the Atomic Energy Commission in November 1958. The goal for startup is mid-1961. The total cost of the test facility, which is designed for a thermal output of 61 MW, is estimated to be about $8 million, plus an additional $1 million for two isolated coolant loops in which special fuel assemblies will be irradiated. Progress during the month of January on the HWCTR design, construction, and supporting experimental work is summarized in this section.

A. STATUS

1. Construction

At the end of January 1960, approximately one-quarter of the HWCTR construction was complete. The zero-level concrete slab for the reactor containment building was poured on January 6 and all external forms were removed. Erection of scaffolding for installation of post-tensioning strands around the concrete structure was completed. Approximately one-half of the required strands were received at the Savannah River Plant. Installation of the permanent interior stairway for the building began, and preparation of interior walls for application of "Liquid Tile" was started.

Work also began on the conversion of Wing D of the TC-1 building into the Office and Shops Building for the HWCTR. The foundations were poured for the remote bunker from which the deluge system of the reactor containment building can be operated.

Delivery of stainless-steel-clad steel plate to Paceco for fabrication of the reactor vessel began during January. Faults detected in the upper head of the reactor vessel and in one clad plate intended for the wall of the lower portion of the vessel will cause a delay in the delivery of critical reactor materials to Paceco. Shipment of a new head and wall plate is expected in March.

2. Design

At the end of January 1960, about 15% of the firm design remained to be completed.

(1) DP-383, Preliminary Hazards Evaluation of the Heavy Water Components Test Reactor, D. S. St. John, et al., May 1959.

3. HWCTR Model

The model of the HWCTR reactor containment building was shipped to the Savannah River Plant. Some minor changes that reflect recent design modifications are being made to the model at SRP. Installation of isolated coolant loop piping and equipment in the model is also to be completed at the plant.

4. Isolated Coolant Loops

Approximately 20% of the firm design of the liquid-D_2O-cooled loop and the boiling-D_2O-cooled loop was complete at the end of January.

B. PHYSICS

1. Transient Behavior of the Liquid-D_2O-Cooled Loop

The transients that follow coolant flow accidents in the liquid-D_2O-cooled loop of the HWCTR were analyzed with the aid of the Savannah River Laboratory's IBM 650 digital computer. The calculations show that none of the accidents considered have serious potential, if the reactor scram circuits are operative.

a. Transients Following a Loss of H_2O Flow

In DP-445, the progress report for November 1959, it was shown that 87.35% of the D_2O flow in the test loop must bypass the heat exchanger in order to attain the design temperatures of about 274°C for the D_2O effluent from the test assembly and about 250°C for the D_2O entering the assembly. These conditions would exist if the HWCTR were operating at 50 MW and the uranium metal test assemblies described in DP-445 were each generating 1.8 MW. In DP-445, a surge tank with a gas volume of about 55 gallons was recommended in order to obtain a reasonably constant pressure differential between the reactor and the loop following a scram. The seals between the gas pressurizing systems of the loop and the reactor are designed to rupture (a) when the pressure in the loop is 700 psi higher than the pressure in the reactor, or (b) when the pressure in the reactor exceeds the pressure in the loop by 200 psi. In the present calculations, the reactor pressure is assumed to be 1000 psi, the loop pressure 1485 psi, and the reactor operating power 50 MW.

Typical transients in loop temperatures and pressures following a sudden and complete loss of H_2O coolant flow to the loop heat exchanger are shown in Figure 1. For the curves shown in Figure 1, it was assumed that the scram circuits failed, that 87.35% of the D_2O flow through the loop was bypassing the loop heat exchanger, and that the reactor continued to operate at 50 MW. The figure shows that upon cessation of H_2O flow, the temperatures around the loop rise steadily, and that the rate of rise of the loop pressure depends strongly upon the initial gas volume in the surge tank. The calculations covered a period of 35 seconds after the accident.

The pressure differentials that are reached 35 seconds after cessation of H_2O flow are listed in the table below for several combinations of loop heat exchanger bypassing and initial surge tank gas volume.

Pressure Differentials Following Loss of H_2O Flow

D_2O Bypassing Loop Heat Exchanger, % of Total Flow	Initial Gas Volume in Surge Tank, gal.	Reactor Power, MW	Loop Pressure after 35 sec, psia	Reactor Pressure after 35 sec, psia	Pressure Differential after 35 sec, psi
0	110	50	1559	1000	559
"	55	"	1629	"	629
"	22	"	1884	"	884
87.35	110	"	1586	"	586
"	55	"	1697	"	697
"	22	"	2144	"	1144
"	110	scram	1458	816	643
"	55	"	1441	"	625
"	22	"	1392	"	576

At a reactor power of 50 MW, with 87.35% of the D_2O flow in the test loop bypassing the heat exchanger, and with a surge tank gas volume of 55 gallons (the operating conditions proposed in DP-445), a sudden loss of H_2O coolant to the loop heat exchanger causes the pressure differential to rise to about 700 psi in 35 seconds, if the reactor is not scrammed. The seal between the reactor gas system and the loop gas system is ruptured at that time, and the pressures in the combined gas systems quickly become balanced at about 1045 psia. Within these 35 seconds, the temperature of the D_2O emerging from the test assembly (the hottest D_2O in the loop) has increased to 298°C; at this temperature, D_2O exerts a vapor pressure of about 1249 psia, considerably in excess of the pressure of 1045 psia of the gas phase over the D_2O. Consequently, if the flow of H_2O coolant is stopped and the reactor is not scrammed, rupture of the seal between the loop and the reactor occurs within about 35 seconds, and sudden boiling of the D_2O in the loop follows. Even if the "very-low-H_2O-flow" signal or "effluent-D_2O-temperature-too-high" signal fails to scram the reactor, about half a minute is available in which to scram the reactor manually, thus avoiding seal rupture and D_2O boiling in the loop.

Instead of assuming that the reactor scram circuits are inoperative, one would be more realistic to assume that the scram circuits will not fail. Figure 2 shows that if the reactor is scrammed when the H_2O

flow stops, the loop temperatures rapidly approach new levels about 5°C above the initial test assembly inlet D_2O temperature and the loop pressure decreases somewhat to a steady value. The reactor pressure decreases a bit more rapidly than the loop pressure, but the excess of loop pressure over the reactor pressure rises to only 625 psi after 35 seconds, well below the seal rupture pressure differential of 700 psi. There is ample time to adjust both the reactor pressure and the loop pressure as the reactor and the loop cool.

Figure 3 shows the temperature of the test assembly effluent D_2O as a function of time for three possible conditions during H_2O-loss accidents: (1) constant reactor power of 50 MW, no D_2O bypassed around the loop heat exchanger; (2) constant reactor power of 50 MW, 87.35% of D_2O flow bypassed; and (3) reactor scrammed from 50 MW, 87.35% of D_2O flow bypassed. Initial values and values 35 seconds after the start of the accident are listed below for the test assembly fuel temperature and for the D_2O temperature at the inlet and outlet of the test assembly.

Loop Temperatures Following Loss of H_2O Flow

D_2O By-passing Loop Heat Exchanger, % of Total Flow	Reactor Power, MW	Initial			After 35 seconds		
		Fuel Temp, °C	Inlet D_2O Temp, °C	Outlet D_2O Temp, °C	Fuel Temp, °C	Inlet D_2O Temp, °C	Outlet D_2O Temp, °C
0	50	235	173	192	262	205	224
87.35	50	303	250	273	325	276	298
87.35	scram	303	250	273	255	255	255

b. Transients Following an AC Power Failure

When the AC power supply to the loop circulating pump fails, coolant circulation decreases relatively slowly, because of the action of the pump flywheel; the circulation continues at this lower rate when the emergency DC power is supplied to the pump. For the calculations, it was assumed that the coolant D_2O flow decreases linearly to one-third of its initial value within 30 seconds and then remains constant at this level.

The loop temperature and pressure transients that follow such a reduction in D_2O flow are shown in Figure 4. In this typical case, it was assumed that the reactor continues to operate at 50 MW and that 87.35% of the loop D_2O flow continues to bypass the loop heat exchanger. As shown in Figure 4, the temperature of the D_2O at the test assembly inlet decreases slightly because of the increased residence time in the heat exchanger. The test assembly fuel temperature and the test assembly effluent D_2O temperature rise to higher steady values. The rate of rise of the loop pressure depends upon the initial gas volume in the surge tank, but the rise would be smaller than in the case of the loss-of-H_2O-coolant accident. Only in the case of a 22-gallon initial gas volume in the surge tank would there be the possibility of rupture of the seal between loop and reactor.

It is more realistic to assume that when AC power fails, the "AC power failure", the "effluent-temperature-too-high" or the "very-low-D_2O-flow" signals would scram the reactor, rather than that the scram circuits would be inoperative. Figure 5 shows that when the AC power failure is accompanied by a reactor scram, the loop temperatures decline almost linearly after 5 seconds, and the loop pressure decreases also; the reactor pressure decreases a little more rapidly (except for the 22-gallon case). With an initial gas volume of 55 gallons in the surge tank, 35 seconds after the start of the accident, the pressure differential would be only 535 psi.

The pressure differentials that are reached 35 seconds after the start of an AC power failure for several combinations of heat exchanger bypassing and initial gas volume in the surge tank are given in the following table.

Pressure Differentials Following an AC Power Failure

D_2O Bypassing Loop Heat Exchanger, % of Total Flow	Initial Gas Volume in Surge Tank, gal.	Reactor Power, MW	Loop Pressure after 35 sec, psia	Reactor Pressure after 35 sec, psia	Pressure Differential after 35 sec, psi
0	110	50	1492	1000	492
"	55	"	1497	"	497
"	22	"	1510	"	510
87.35	110	"	1526	"	526
"	55	"	1566	"	566
"	22	"	1696	"	696
"	110	scram	1410	816	594
"	55	"	1350	"	535
"	22	"	1199	"	384

Figure 6 shows the temperature of the test assembly effluent as a function of time for three possible conditions during an AC power failure accident: (1) constant reactor power of 50 MW, no D_2O bypassed around the loop heat exchanger; (2) constant reactor power of 50 MW, 87.35% of D_2O flow bypassed; and (3) reactor scrammed, 87.35% of D_2O flow bypassed. The test assembly fuel temperature and the D_2O temperature at the inlet and outlet to the test assembly are presented in the next table for the start of the accident and 35 seconds later.

Loop Temperatures Following an AC Power Failure

D_2O Bypassing Loop Heat Exchanger, % of Total Flow	Reactor Power, MW	Initial: Fuel Temp, °C	Initial: Inlet D_2O Temp, °C	Initial: Outlet D_2O Temp, °C	After 35 Seconds: Fuel Temp, °C	After 35 Seconds: Inlet D_2O Temp, °C	After 35 Seconds: Outlet D_2O Temp, °C
0	50	235	172	192	241	159	219
87.35	50	303	250	273	322	249	317
87.35	scram	303	250	273	237	235	237

c. Transients Following Changes in D_2O Bypassing the Loop Heat Exchanger

(1) Increased Cooling - Typical loop transients following a sudden decrease in the fraction of coolant D_2O flow bypassing the loop heat exchanger are shown in Figure 7. Continued operation of the reactor at 50 MW is assumed. As shown in the figure, the resultant pressure differential decreases slowly, but seal rupture (at a differential pressure of -200 psi) is not threatened. The data in the following tables indicate that this accident is not dangerous.

Pressure Differentials Following an Increased Cooling Accident

D_2O Bypassing Loop Heat Exchanger, % of Total Flow	Initial Gas Volume in Surge Tank, gal.	Reactor Power, MW	Loop Pressure after 35 sec, psia	Reactor Pressure after 35 sec, psia	Pressure Differential after 35 sec, psi
87.35 → 0	110	50	1419	1000	419
" "	55	"	1368	"	368
" "	22	"	1237	"	237

Loop Temperatures Following an Increased Cooling Accident

D_2O By-passing Loop Heat Exchanger, % of Total Flow	Reactor Power, MW	Initial Fuel Temp, °C	Initial Inlet D_2O Temp, °C	Initial Outlet D_2O Temp, °C	After 35 Seconds Fuel Temp, °C	After 35 Seconds Inlet D_2O Temp, °C	After 35 Seconds Outlet D_2O Temp, °C
87.35 → 0	50	303	250	273	280	224	247

(2) Decreased Cooling - Typical loop transients following a sudden increase in the fraction of coolant D_2O flow bypassing the loop heat exchanger are shown in Figure 8. Continued operation of the reactor at 50 MW is again assumed. The data in the following table show that this type of accident could lead to seal rupture.

Pressure Differentials Following a Decreased Cooling Accident

D_2O Bypassing Loop Heat Exchanger, % of Total Flow	Gas Volume in Surge Tank, gal.	Reactor Power, MW	Loop Pressure after 35 sec, psia	Reactor Pressure after 35 sec, psia	Pressure Differential after 35 sec, psi
0 → 50	110	50	1489	1000	489
" "	55	"	1492	"	492
" "	22	"	1501	"	501
0 → 95	110	"	1524	"	524
" "	55	"	1556	"	556
" "	22	"	1662	"	662
87.35 → 95	110	"	1545	"	545
" "	55	"	1604	"	604
" "	22	"	1810	"	810
87.35 → 99	110	"	1613	"	613
" "	55	"	1753	"	753
" "	22	"	2369	"	1387

If the initial gas volume in the surge tank were 55 gallons, a sudden increase from 87.35% to 99% bypassing of D_2O would cause the pressure differential to rise to about 700 psi in 30 seconds unless the reactor were scrammed. At the 700-psi pressure differential, the seal between the reactor gas system and the loop gas system would rupture and the D_2O in the test loop would subsequently boil rapidly.

Actually, however, the reactor would be scrammed within about 9 seconds by the "effluent-D_2O-temperature-too-high" signal (+15°C over steady-state value), since an effluent temperature of 288°C would be attained at that time. During this 9-second period the loop pressure would have risen to 1531 psi; the pressure differential of 531 psi would still be well below the rupture value of 700 psi. The "increased ΔT" warning signal (+3°C over the steady-state difference between test assembly inlet and outlet D_2O temperatures) would not give any protection in this accident, since the ΔT would have actually dropped a small amount within this period.

Figure 9 shows the temperature of the test assembly effluent D_2O as a function of time for four possible increased-bypass cooling accidents: 87.35% → 99%, 87.35% → 95%, 0% → 95%, and 0% → 50%. The D_2O coolant and fuel temperatures at the start of the accident and after 35 seconds are summarized in the following table.

Loop Temperatures Following a Decreased Cooling Accident

D_2O By-passing Loop Heat Exchanger, % of Total Flow	Reactor Power, MW	Initial Fuel Temp, °C	Initial Inlet D_2O Temp, °C	Initial Outlet D_2O Temp, °C	After 35 Seconds Fuel Temp, °C	After 35 Seconds Inlet D_2O Temp, °C	After 35 Seconds Outlet D_2O Temp, °C
0 → 50	50	235	172	192	237	175	195
0 → 95	"	"	"	"	254	195	215
87.35 → 95	"	303	250	273	320	269	292
87.35 → 99	"	"	"	"	335	286	309

2. PSE Measurements of HWCTR Temperature Effects

A more accurate analysis of the PSE measurements of the effects of moderator temperature in the HWCTR showed that a hot-and-dirty HWCTR lattice will probably have a slightly longer reactivity lifetime than had been previously calculated. The analysis described in this article differs from the previous analysis of the experimental data (DP-445) because (1) a different method is used to correct the measurements for the difference between the reflector thickness of the PSE mockup and that of the full-scale HWCTR, and (2) the effect of temperature change on rod worth is taken into account in the present analysis. The results of the experiments and calculations now show that the equivalent of 3.0 black control rods must be removed from the reactor in raising the moderator temperature from 20 to 240°C as compared to the 3.65 value reported in DP-445.

In the experiments, the change in the "material buckling" of the lattice was measured as a function of temperature for various control rod configurations. It was assumed that the "radial buckling" remained constant. As the temperature of the moderator in the mockup was increased from 20 to 240°C, the buckling of the mockup remained constant when the number of rods in the outer ring was decreased from 10 to 6 rods. By combining this measurement with the results of multiregion, two-group calculations (the LIL ABNER code), the change in reactivity that results from raising the HWCTR moderator temperature from 20 to 240°C can be obtained from the expression.

$$\Delta k_{eff} = \alpha \delta k_c^{PSE}(N_1 - \beta N_2),$$

where

> N_1 and N_2 are the required number of control rods in the HWCTR mockup at 20 and 240°C, respectively, that give the same material buckling in the PSE,
>
> δk_c^{PSE} is the reactivity worth of a single control rod in the mockup at a temperature of 20°C,
>
> β is the ratio of the worth of a control rod at 240°C to that at 20°C, and
>
> α is the ratio of the temperature coefficient of the full-scale HWCTR to the temperature coefficient of the PSE mockup with the smaller reflector.

Values of $N_1 = 10$ and $N_2 = 6.0$ were obtained from the PSE experiments. The experimental uncertainty in $N_1 - N_2$ is probably about ±0.5. The value of δk_c^{PSE} was obtained by use of the following expression:

$$\frac{\delta k_c^{PSE}}{\delta k_c^{HWCTR}} \approx \left[\frac{\delta B^2_{(PSE)}}{\delta B^2_{(HWCTR)}}\right] \left[\frac{M^2_{eff}(PSE)}{M^2_{eff}(HWCTR)}\right]$$

Comparison of the measured buckling changes due to control rod insertions in the full-scale PDP mockup of the HWCTR lattice and in the PSE mockup gave a value of 1.13 for the δB^2 ratio. The δk_c^{HWCTR} was measured to be 1.85% k_{eff} in the PDP experiments. An M^2_{eff} ratio of 0.806 was obtained from LIL ABNER calculations. Combination of these quantities resulted in a value of $\delta k_c^{PSE} = (1.13)(0.806)(1.85) = 1.69\%$ k_{eff}.

The value, $\alpha = 0.75$, was obtained by comparing the calculated temperature coefficient in the full-scale HWCTR mockup to that in the PSE mockup. A value of β was not measured, but was assumed to be 1.1 ±0.1. The above values for α, β, k_c^{PSE}, N_1, and N_2, lead to the following value for the temperature effect

$$\Delta k_{eff} = (-0.75)(1.69)[10 - (6.0)(1.1)] = -4.3\%\ k_{eff}$$

Uncertainties of ±0.1 in β and ±0.5 in $(N_1 - N_2)$ will each produce an error in Δk_{eff} of ±0.7% k_{eff}.

The number of black control rods that must be removed from the HWCTR to compensate for a moderator temperature increase from 20 to 240°C can be obtained as follows

$$(N_1' - 1.1N_2') = 2.32$$

where N_1' and N_2' are the number of black control rods that are in the reactor at 20 and 240°C, respectively. This shows that if $N_1' = 10$ in the cold clean condition, the temperature rise will be compensated for by the removal of 3.0 rods from the reactor.

The "experimental" value of 4.3% k for the effect of temperature is significantly lower than the theoretical value of 5.8% k that was reported in the HWCTR papers presented at the 1959 American Nuclear Society at Gatlinburg, Tennessee (DP-413). Therefore the margin of reactivity that will be available for fuel burnup is larger than had been anticipated.

C. COMPONENTS TESTING

1. Steam Quencher for the Boiling-D_2O-Cooled Loop

Exploratory tests were conducted at the Savannah River Laboratory to provide preliminary information for the design of a steam quencher for the boiling-D_2O-cooled loop of the HWCTR. Two types of pipeline quenchers were tested: (1) a 1-1/2-inch-diameter perforated sparger pipe in a 4-inch diameter steam line, and (2) an open end 1-1/2-inch-diameter pipe in a 4-inch-diameter steam line. The performance of both units was deemed to be satisfactory, but the perforated sparger is considered to be superior because of the lower noise level that accompanied its use. The details of the experiments are recorded below.

No significant increase in vibration and audible noise was observed when a stream of saturated steam at 180°C was quenched by sparging water at 16°C into it. As shown in Figure 10, water at 16°C was injected through a perforated sparger pipe, 1-1/2 inches in diameter, into a 4-inch-diameter pipe in which saturated steam at 135 psig was flowing. The difference in temperature between the steam and cold water approximated the maximum subcooling that has been proposed for the HWCTR quencher, 195°C. In one experiment, 17 gpm of water was sparged into a steam flow of 2000 lb/hr, at which rate the steam velocity approximated that intended in the approach piping to the HWCTR quencher. The water flow was about 20% greater than the stoichiometric quantity required to condense all of the steam. In a second experiment with this equipment, 37.5 gpm of water was sparged into steam that was flowing at a rate of about 4700 lb/hr, which roughly duplicates the mass flow rate expected in the HWCTR quencher. The flow of water in this case was about equal to the stoichiometric quantity required to condense all of the steam. In both experiments no significant noise or vibration was observed. In a third experiment, water was injected into the steam flow from the open end of a 1-1/2-inch-diameter pipe. At a steam flow of 4500 lb/hr and a water flow of 40 gpm, audible noise and vibration of the test apparatus were observed, but neither was of a serious nature. The effectiveness of the test equipment in quenching all of the steam downstream of the sparger was not determined in any of the tests.

2. Prototype Gripper Mechanism

Testing of the prototype gripper mechanism for the irradiated fuel transfer coffin has continued at the Savannah River Plant. The up-and-down drive for the gripper has been cycled at speeds as high as 40 ft/min. It is now expected that during HWCTR fuel discharge the gripper will be operated at a speed of 40 ft/min for part of the up-down cycle and with a reduced speed of about 20 ft/min in certain zones. This will approximately halve the time previously required for transport of irradiated fuel elements through air before cooling can be applied to them inside the transfer coffin. Testing is also in progress on the piston-ring hold-down devices for the fuel housing tubes. To determine the effect of sediment on the reproducibility of the hold-down force, water containing iron powder is being circulated through the test rig.

D. FUEL ELEMENT DESIGN

The designs of top fittings on all the subassemblies associated with both the driver and test fuel elements for HWCTR were modified to make them compatible with the latest gripper mechanisms on the charge-discharge coffin. Because these grippers now include distribution cones to direct emergency cooling water over the discharged fuel pieces, regardless of whether the fuel is discharged with or without its housing tube, it was necessary to lengthen the top fittings on the housings and to provide on both fuel pieces the necessary clearance for the long grippers.

The results of the design changes are shown in Figures 11 and 12. The outer housing subassembly was increased in length by one inch and the fuel subassemblies were also lengthened by the same amount. In all cases the increase was made in the length of the top fittings. The target subassembly was decreased in length by 3/4 inch to provide part of the needed clearance so that the top fitting of the driver fuel piece would not have to be lengthened excessively.

All of the parts that were redesigned have been fabricated. The dimensions are now being measured and the pieces are being fitted together prior to incorporation into flow test assemblies.

II. TECHNOLOGY OF FULL-SCALE REACTORS

A. PHYSICS

1. Modifications to the Process Development Pile

The versatility of the Process Development Pile (PDP) of the Savannah River Laboratory has been increased substantially by recent equipment modifications that will permit a greater variety of heavy-water-moderated lattices to be studied. The alterations were made during the last few months. The following discussion presents a brief description of the modified PDP.

The stainless steel PDP tank, which is approximately 16 feet in diameter and which has an effective height of 15 feet, is surmounted by a grid of lattice support beams onto which are clamped individual support plates for the separate lattice elements. By varying the positions and dimensions of these support plates almost any type of lattice element can be studied over a wide range of lattice spacings. The lattice support beams are in turn surmounted by a D_2O vapor seal consisting of large, gasketed, aluminum plates that are hinged and are swung out of the way during loading and unloading operations. The PDP is provided with three separate rod systems, the safety, shutdown, and control rods. All three sets of rods are motor-driven by a system of cables. The safety rod actuators have provision for scram; the control rods are driven individually; and the shutdown rods have jack-shaft drives. A total of nearly two hundred separate rods are provided in the three systems. These rods may be located at any place in the reactor tank. The auxiliary systems of the PDP include reactor instrumentation, and a water-handling system with storage, chemical purification, heating, and drying facilities.

2. Natural Uranium Metal Lattices

The initial lattice studies in the modified PDP facility are being made with full loadings of seven-rod clusters of 1-inch-diameter rods of natural uranium metal. The first experiments with these clusters were performed at triangular lattice spacings of 18.5 and 14 inches, center-to-center between the fuel clusters, in moderator having an isotopic purity of 99.3% D_2O. Both bare and reflected lattices were examined. The measurements included (a) critical-water-height and flux-distribution measurements to determine the lattice bucklings, (b) detailed thermal-flux and neutron-temperature measurements to determine the thermal utilizations and diffusion areas, (c) detailed resonance-flux measurements to determine resonance captures and the neutron slowing down distributions, and (d) detailed fast-flux measurements to determine fast fissions and captures. The same series of measurements is now being repeated in 99.76% D_2O to determine the effects of moderator purity on the various lattice parameters.

Analyses of these experiments are still incomplete, but the results generally appear to be in fair accord with those obtained earlier in studies of partial PDP loadings. An exception to this general agreement is the resonance flux distribution, which has more spatial variations than had been recognized previously.

B. D_2O LEAKAGE

The program to determine the leakage of D_2O from typical reactor components of conventional design continued at the Savannah River Laboratory. Unrecoverable losses of D_2O from components, such as the pump seals and valve stem closures reported below, could have considerable effect on the economics of heavy-water-moderated power reactors. However, results of SRL tests and Sargent & Lundy tests to date have shown that it is possible to limit the unrecoverable losses of D_2O from these sources to a tolerable level.

1. Pump Seals

The measurement of liquid leakage from the mechanical seals of a centrifugal pump in a flow loop at the Savannah River Laboratory is continuing. The average leakage of liquid water during 57 days of operation was 29 gal/yr from the inboard seal and 90 gal/yr from the outboard seal. During the last 15 days of test, the leakage rates decreased to average values of about 20 gal/yr for each seal. This type of leakage is usually recoverable. The pump suction pressure was maintained at 850 psig.

As reported previously, the vapor leakage from mechanical seals is of considerable interest because such leakage may not be readily recoverable. An attempt was made to measure the vapor leakage from one of the seals by analyzing the effluent stream of nitrogen from the seal collection chamber for moisture content. However, this method was found to be unsatisfactory because the effluent nitrogen stream was almost saturated with water vapor over a wide range of gas flows. At high purge flows of nitrogen, a substantial fraction of the liquid leakage was vaporized, and at low purge flows it was uncertain whether or not the vapor leakage was adequately swept out of the seal chamber into the moisture-detecting instrument. During the next scheduled shutdown of the test facility, one of the seal assemblies will be modified so that the vapor leakage can be measured without flow of nitrogen through the seal chamber.

2. Valve Stem Closures

Preliminary analysis of the data obtained in thermal- and pressure-cycling tests of three valve stem closures showed that the maximum leakage rate from any of the closures was less than 75 in^3/month (3.9 gal/yr) of liquid water at 25°C; the average leakage rate was considerably lower. In a typical three-hour cycle, each valve stem assembly was maintained at a maximum temperature of 260°C and a maximum pressure of 1000 psig for one hour; the balance of the time was used for heating and cooling at a pressure of 1000 psig and venting to atmospheric pressure at the end of the cycle. The tests included the bonnet-and-stem assemblies of a 3-inch globe valve and a 6-inch gate valve that were cycled 110 times, and the stem assembly of a 3/4-inch globe valve that was cycled 80 times. Testing of a second 3/4-inch valve stem assembly was terminated after 5 cycles because the leakage rate exceeded the maximum range of the moisture detection instrument. In this last test, a complete valve was tested, and it is believed that the excessive leakage may have resulted from distortion of the valve body when the flange of the leakage collection chamber was welded to it.

During the tests all of the valve stems were stationary and backseated. Each valve stem was polished to a No. 6 RMS finish, and preformed packing rings of white asbestos with copper wire reinforcement were used. It was reported last month that the leakage from the 3- and 6-inch valves was to be determined from the moisture increase in a nitrogen stream that was to pass through the lantern ring of the valve, between the upper and lower sets of packing rings. However, this method proved to be unsatisfactory because the leakage rates caused the range of the hygrometer to be exceeded. A leakage collection chamber was therefore constructed around each valve stem assembly. The leakage past the two sets of packing rings was then sufficiently low to be measured by the hygrometer. The leakage rates reported above were determined by this latter method.

Modifications were made to existing piping to permit testing of a complete 3-inch gate valve, through which deionized water will flow at 260°C and 1000 psig. During the test, the valve will be opened and closed at 2 to 5-minute intervals for 1000 cycles, and the water leakage from the valve stem closure will be continuously measured and recorded. The packing material and the finish on the valve stem are identical with those described above. Testing of this valve is scheduled to start early in the next report period.

C. REACTOR FUELS AND MATERIALS

The chief objective of the program on reactor fuels and materials is the development of a low cost natural uranium fuel, of oxide or metal, that can withstand the exposures and temperatures contemplated for D_2O-cooled-and-moderated power reactors. The work on uranium metal tubes is being conducted at Nuclear Metals, Inc., where the immediate emphasis is on the coextrusion of Zircaloy-clad tubes of various core compositions for irradiation tests. The major effort in the du Pont study on oxide fuel is at the Savannah River Laboratory, where experimental studies of a cold-swaging process for direct mechanical compaction of Zircaloy-clad elements of oxide are underway. The progress of these programs as well as the results of tests on joints between Zircaloy and stainless steel are reviewed in this section.

1. Fuel Elements of Uranium Metal

a. Unalloyed Tubes for VBWR Irradiation

Fabrication of two 3%-enriched unalloyed uranium tubes was completed at Nuclear Metals, Inc. These tubes, which are 2.060-inch OD, 1.465-inch ID, and 50 inches long are candidate test specimens for irradiation in the Vallecitos Boiling Water Reactor. VBWR irradiation of such tubes will permit comparison of the behavior of unalloyed uranium with the U - 2 w/o Zr alloy at temperatures and pressures similar to those of a full-scale power reactor. The progress of a VBWR irradiation test of the alloy material has been recorded in earlier reports. The fabrication of the unalloyed tubes was similar to that used for four natural uranium prototype tubes. The fabrication of the prototypes was discussed in previous reports. The results of visual inspection and evaluation tests indicate that the two enriched tubes are satisfactory irradiation candidates.

Radiography measurements showed that the total core length of each tube is very close to 40 inches; the front taper is 6 inches long, the uniform core is 29 inches long, and the rear taper is 5 inches long. Front and rear shifts are less than 1 inch. After beta heat treatment, pickling, and etching, the Zircaloy surfaces were smooth and contained no discernible defects. All diameters and wall thicknesses were within specifications. Ultrasonic tests showed that good-quality clad-core bonds had been obtained over the core section of the tube. Notch-fracture tests on rings cut from the tube ends showed that good bonds were also obtained between the clad and the end seals. An autoclave test for 4 hours in water at 345°C followed by 4 hours in steam at 400°C and 1500 psi produced the usual black oxide film that is desired for corrosion-resistant Zircaloy.

An autoradiographic test to locate areas of thin cladding is the only evaluation test that is not yet complete. A preliminary autoradiograph survey that was made before etching and autoclaving indicated that these tubes probably do not contain areas of thin cladding. Final measurements, after etching and autoclaving, are now in progress.

b. Beta-Phase Uranium Alloys

Study is continuing at NMI to find an alloy that will retain a beta phase under reactor conditions. Uranium in the beta phase may be more creep resistant and dimensionally stable in irradiation than alpha uranium. Earlier studies of fabrication, heat treatment, and structure have revealed four promising alloys:

(a) U - 0.3 w/o Cr

(b) U - 0.3 w/o Cr - 0.3 w/o Mo

(c) U - 0.4 w/o Al - 0.4 w/o Si

(d) U - 0.3 w/o Cr - 0.4 w/o Si

Further tests of these alloys are reported below.

Tensile properties of the four promising beta-phase alloys were measured at room temperature. The strength and ductility of the alloys were lower than usually encountered with beta-phase uranium and both properties decreased with increasing amounts of the beta phase. It should be noted, however, that the amount of beta phase present in the alloy can be controlled by varying the fabrication and heat treatment procedures. The significance of the lower strength and ductility on the irradiation behavior of the alloys cannot be surmised, but will be determined by irradiation tests.

c. Improvements in the Coextrusion Process

(1) Copper Extrusion Sleeves - The source of the surface depressions that marred the quality of many of the coextruded tubes that had been produced at Nuclear Metals has been found - iron particles in the copper extrusion cans.

Thin-walled copper tubing is used to encase the extrusion billets of the power reactor metal fuel tubes. This copper jacket prevents oxidation of the billet components during pre-extrusion heating and then acts as a lubricant during extrusion. Nuclear Metals has recently found that tiny iron particles in the copper cause severe depressions in the cladding of the extruded tube. For example, a recently extruded tube at NMI was rejected as an irradiation candidate because of depressions that were 0.0065 inch deep.

In order to improve the quality of the copper jackets used for the coextrusion, the usual visual inspection of the copper sleeves before billet assembly is now being supplemented by a ferrocyanide test for iron contamination of the surfaces. Additional tests are being considered, including the possibility of using an eddy current test to locate iron particles that are not exposed on the sleeve surfaces.

(2) Zircaloy Etching - In the initial development of fabrication procedures for Zircaloy-clad tubes, a comparatively complex, time-consuming, and expensive process was used for bright etching the tube surfaces before heat treatment and before the final autoclave test.

The objective was to obtain Zircaloy cladding surfaces that would be free of contamination and would be corrosion resistant to high temperature water and steam. It now appears that a significant cost reduction can be achieved without any sacrifice in tube quality, through changes in the etching procedures. Studies are now in progress to investigate the possibilities of a simplified etching process.

Among the items under investigation are the following which have a strong influence on the cost and complexity of etching: (1) prior surface preparation, (2) reagent quality, (3) etchant contamination, (4) etchant life, (5) rinsing procedure, and (6) etchant additives such as wetting agents. A series of experiments has been designed to evaluate these items. Results reported this month pertain to Item (4), etchant life.

Repeated use of a batch of etchant has been tested through a total of ten cycles. For this test, a Zircaloy tube of the same dimensions as the prototype power reactor fuel tubes was repeatedly etched, with HF addition to the bath after each cycle. The performance of the etchant in each cycle was evaluated by also etching a test ring in a sample of the bath, and subjecting this ring to a corrosion test in 750°F, 1500-psi steam for 16 days. The appearance of the ten test rings after the corrosion test was essentially identical; all had a black oxide film with no evidence of white oxide; and the weight gains were uniformly low. These results indicate that satisfactory etching can be obtained with replenished etchant.

In a second test, in which fresh etchant was used, the full-length Zircaloy tube was etched for a total of five cycles, and the tube was then subjected to a corrosion test for 18 hours in water at 345°C, followed by 24 hours in steam at 400°C and 1500 psi. The appearance of the tube after this test was satisfactory except for the presence of thin white stringers on the black oxide surface. These stringers do not resemble the white oxide normally associated with poor corrosion resistance, and are believed to be the result of some difference between the solid Zircaloy tube, extruded from forged material, and the Zircaloy cladding of standard composite fuel tubes.

2. Fuel Elements of Natural Uranium Oxide

a. Swaged Tubes

(1) Stainless-Steel-Clad Irradiation Specimens - A second group of tubular specimens of stainless-steel-clad uranium oxide was prepared at the Savannah River Laboratory for irradiation testing. The specimens were fabricated by the same process that was used for an earlier set of tubes; crushed, fused oxide was loaded into the annulus between concentric sheaths of 0.022-inch-thick tubing of Type 304 stainless steel and swage-compacted to final density. During swaging, the tube was supported by a hardened steel mandrel. Fifteen tubular

specimens, 2-feet long, were cut from six swaged tubes. Stainless steel end fittings were then welded into each end of the tubes.

Evaluation of the fabricated specimens has been partially completed. Dimensional data show that the tubes are of adequate quality for irradiation testing. Outer and inner diameters averaged 2.138 and 1.458 inches, respectively. Eccentricity of the outer sheath with respect to the inner sheath averaged 0.010 with a maximum of 0.013 inch. Cladding thickness for inner and outer sheaths averaged 0.017 and 0.020 inch, respectively, with minimum values of 0.015 and 0.018 inch. If the maximum penetration of the oxide particles into the cladding were no greater than that previously encountered, 0.005 inch for the inner sheath and 0.004 inch for the outer sheath, then the minimum cladding at any point would be 0.010 and 0.014 inch for the inner and outer sheaths, respectively.

All of the tubes contained fused uranium oxide from the Norton Company. Although determination of the oxygen-to-uranium ratio for the oxide is still incomplete, none of the core material tested to date has had an O/U ratio greater than 2.023. The tubes ranged in density from 88 to 90% of the theoretical density.

Some difficulty was encountered in the fabrication of two 10-foot-long tubes. Feeding of the tubes into the swager was accompanied by whipping of the tube. The combined weight of the tube and mandrel and the lack of support guides interfered with manual feeding and caused the whipping. A hydraulic feed mechanism is being designed to overcome this difficulty. The density of the long tubes averaged 89.7 and 88.9% of the theoretical value with a range of 88 to 90.6%.

(2) Zircaloy-Clad Oxide - Two tubes, approximately 5-feet long and clad with Zircaloy-2, were swaged successfully through a 37% area reduction; the average oxide density was 91.3% of theoretical with a minimum of 90.7%, and a maximum of 91.8%. The fabrication behavior of the Zircaloy-clad tubes was similar to that of stainless-steel-clad tubes except that the springback of the outer sheaths after the last swaging pass was nearly twice that of Type 304 stainless steel. During the fabrication of these tubes, the inner sheaths became bound to the mandrels, because the mandrels had been inadequately lubricated. One of the tubes was cut from the mandrel and the oxide was removed to perform a "Zyglo" test for cracks on the inside surfaces of the swaged sheaths. No cracks were observed. Metallographic examination of the sheaths is in progress. Special tools that were fabricated for attachment to the drawbench were used to remove the mandrel from the other oxide tube.

b. Vibratory Compaction of Uranium Oxide

Vibratory compaction is being investigated as a means of increasing the packed density of the uranium oxide contained in the tubular sheaths prior to swaging. The increased density is desired in order to reduce the possibility of sheath wrinkling and cracking that frequently accompanies the swaging of loosely packed oxide tubes. Also, this precompaction would reduce the number of swaging passes currently required to fabricate a tube by a factor of more than two.

A second series of tests was conducted by the Savannah River Laboratory at Dayton T. Brown, Incorporated, on 7-foot-long stainless steel tubes that were filled with crushed, fused uranium oxide. The diameters of the tubes were nominally 2.510-inch OD and 1.466-inch ID; the cladding thickness was 0.022 inch; and stainless steel plugs were welded in each end of the tubes. The results of these tests were similar to the results that were reported last month in DP-455 on tubes of 3 to 4-foot length. Maximum densities of 69 to 79% of theoretical were attained at the resonant frequency for a particular assembly. For the longer tubes this frequency was in the range 145 to 212 cps as compared to 255 to 310 cps for the shorter tubes of the first test. A vibratory compaction machine is being obtained for the Savannah River Laboratory in order to continue this study and to add a vibratory compaction step in the cold-swaging process for the manufacture of Zircaloy-clad oxide tubes.

c. Stability of the Oxygen-to-Uranium Ratio of Fused UO_2

A test to determine the stability of the oxygen-to-uranium ratio of fused uranium oxide has been in progress at the Savannah River Laboratory since July 1959. No significant increases in the oxygen-to-uranium ratios (O/U) were observed in the first four months of test. Uranium oxide of several different initial O/U ratios and particle sizes was exposed to either conditioned or nonconditioned air. Eight specimens increased in average O/U, three decreased, and three showed no change. Several analyses were run on each specimen, and the changes in the O/U averages after four months were within the scatter of the original data. Companion samples will be analyzed after periods of eight months and one year to see if any significant amounts of oxidation develop.

3. Cladding Studies

a. Zircaloy Sheathing for Swaged Tubes

Four 8-foot lengths of thin-walled Zircaloy-2 tubing were received from Harvey Aluminum Company for preliminary evaluation. This material was the initial delivery of an order for 200 feet of Zircaloy tubing that will be used for the inner and outer sheaths of the swage-compacted oxide fuel tubes.

The sheaths were examined for surface quality, dimensional control, and straightness. Although the dimensional variations and the bow were greater than desired, the over-all quality of the tubing was good. "Zyglo" examination of all the outer surfaces and representative samples of the inner surfaces revealed no surface defects that would interfere with the swaging of the sheaths. The dimensional data are summarized in the following table:

Zircaloy Sheath Dimensions, inches

	Outer Sheath			*Inner Sheath*		
	Specified	Average	Range	Specified	Average	Range
Outer diameter	2.500 ±0.015 -0.000	2.494	2.506 - 2.475	-	-	-
Inner diameter	-	-	-	1.465 +0.015 -0.000	1.460	1.468 - 1.445
Wall thickness	0.030 ±0.003	0.034	0.032 - 0.036	0.030 ±0.003	0.031	0.029 - 0.033

The bow in the tubes may give rise to alignment problems during oxide loading and to eccentricity after swaging. The two inner sheaths and one of the outer sheaths exhibited 3/32 to 1/8-inch bow. The maximum bow on the remaining outer sheath was only 1/32 inch; however, this tube exhibited a reverse curvature, which indicated that some form of gag-press straightening had been performed. Effort will be made by the vendor to supply straighter tubes for the remainder of the order.

b. Cladding Thickness Tester for Zircaloy-Clad Fuel Tubes

It is necessary to verify that sufficient cladding exists over an entire reactor fuel element of uranium metal or uranium oxide before it can be safely accepted for irradiation. Eddy current testers had been used successfully to measure the thickness of aluminum cladding over uranium cores. However, these early models of the eddy current devices were not sufficiently sensitive to measure the thickness of zirconium cladding on a uranium metal element because the resistivity of zirconium is much closer to that of uranium than is that of aluminum.

A more sensitive eddy current tester was developed for use with zirconium or Zircaloy cladding. This instrument was described in DP-408. Extensive use of this device showed that it was overly sensitive to zirconium surface conditions, such as scale and roughness, and nonlinear in the range of cladding thickness of current interest.

A new circuit for the zirconium cladding thickness tester has been developed. The new instrument has a linear response over a cladding thickness range of 0.006 to 0.015 inch of zirconium and is not sensitive to rough or scaly surface conditions. In addition, the cladding thickness measurement is not altered even if the measuring probe is lifted off the surface by as much as 0.004 inch. This characteristic permits the measurement of cladding thickness on surfaces of somewhat irregular shape. The new instrument may be used on both the interior and exterior surfaces of tubular shapes. It has also been applied to the measurement of total wall thickness of zirconium tubes with 0.030-inch wall thickness.

c. Electron Beam Welder

It is difficult to make a welded closure on zirconium- or steel-jacketed uranium oxide fuel elements because the gases contained in the oxide are released by the welding heat and cause bubbles in the weld. It has been demonstrated at Hanford that electron beam welding, carried out in a vacuum, provides a satisfactory solution to this problem. However, the Hanford experimental welder is too small for economical welding of full-sized fuel elements.

A large-capacity electron beam welder has been assembled at the Savannah River Laboratory. A surplus ion beam power supply from the electromagnetic uranium separators built in 1945 was obtained from Oak Ridge. Up to one ampere of direct current at voltages of up to 50 kilovolts is available to the welder. One-half ampere may be obtained at up to 100 kilovolts.

Initial experiments have shown that zirconium and steel can be readily welded with the new facility. The voltage has been maintained below 15 kilovolts to avoid the X-ray shielding required at higher levels.

The welding of experimental fuel elements will be attempted in the near future. It is anticipated that the uranium oxide may be difficult to outgas because of the slow diffusion from the interior of the tube.

4. Zircaloy-to-Stainless-Steel Joints

a. Fabrication and Testing at Nuclear Metals, Inc.

In the program to develop a process for obtaining metallurgically bonded tubular joints between stainless steel and Zircaloy, effort is being concentrated on the joints that exhibit corrosion behavior that is essentially equivalent to that of high quality Zircaloy. About fifty rod specimens and several tubes were made to explore the effects of several design and process variables on joint quality. The conclusions derived from destructive tests of these joints are summarized below.

The effect on joint quality of using different types of stainless steel was investigated. The steels tested were Types 304, 304L, 321, and 347, all of which contain 18% Cr and 8% Ni. The latter three were modified for improved corrosion resistance after welding, by limiting the carbon content in the 304L, by adding titanium to the 321, or by adding niobium to the 347. There was no marked difference in the bond strengths obtained with these four types of stainless steel, although one of two rods with Type 321 and one of five rods with Type 347 had weak bonds. Five rods with Type 304L had consistently good bonds, and this steel is therefore favored for future work, although Types 321 and 347 probably are also satisfactory.

The fabrication conditions that provided good bond strengths on rods were used to manufacture a set of six tubular joints, each about 2 inches in diameter. Longitudinal strips, about 1/2-inch wide, were cut from three of these joints and subjected to various mechanical tests. When subjected to tensile stresses, the strips necked and fractured in the Zircaloy section, at values that are typical of the ultimate strength of Zircaloy. None failed preferentially at the joint interface. Bend tests of 360° on a 2-inch-diameter mandrel did not produce any separation at the interface. Rolling to produce a 13% elongation had no apparent effect on the joints. Subsequent tensile testing of these rolled strips resulted in fracture in the Zircaloy section, as in the other tensile tests. Stud-weld tests of bond strength (in a direction normal to the tube axis and approximately normal to the interface) gave values for one joint that, although considered satisfactory, are somewhat below those expected on the basis of the rod tests. It is possible that these lower values are associated with the different method of sample preparation necessitated by the different geometry of the tubular joints compared with the rods. Corrosion testing of specimens from these joints has been started.

b. Hydrostatic and Cycling Tests

Two NMI-fabricated tubular joints (Nos. 2 and 5) between Zircaloy-2 and 347 stainless steel successfully passed a 116-cycle test in which they were internally pressurized with deionized water at a maximum temperature of 260°C and a pressure of 1000 psig. No visual change in the appearance of the joints was observed after completion of the cyclic tests. In a typical three-hour cycle at 1000 psig, each joint was heated for one hour, maintained at the maximum operating temperature for one hour, and then cooled for one hour. At the completion of each cycle the joints were vented to atmospheric pressure.

Joint No. 2, which was about 1.8-inch OD and 1.5-inch ID, was hydrostatically tested at 3100 psig prior to the cyclic test; Joint No. 5, which was 3.85-inch OD and 3.59-inch ID, was hydrostatically tested at 2400 psig. The calculated hoop stress at the maximum pressure of 1000 psig during the cyclic test was 6,700 psi for Joint No. 2 and 14,100 psi for Joint No. 5. Both joints will be hydrostatically tested to rupture in February. Metallurgical examination of the joints after completion of the cyclic tests revealed no distortion and no corrosion.

During a destructive hydrostatic test of a third NMI-fabricated joint, the Zircaloy end closure of the test assembly ruptured; the internal pressure at the time of rupture was 16,500 psig. The rupture was a brittle circumferential fracture of the Zircaloy tube wall immediately adjacent to the weld joining the Zircaloy end plug to the tube. Except for a permanent hoop strain of 0.16%, no damage was observed in the joint region. This joint was not cyclically tested. The calculated hoop stress in the joint was about 70,000 psi when the hydrostatic pressure was 16,500 psig. The joint was about 1.9-inch OD and 1.5-inch ID.

D. IRRADIATION OF REACTOR MATERIALS

The status of the irradiation tests remained essentially unchanged since the last report period except that a modest gain in exposure was made by the materials and fuels undergoing irradiation and that an SRP irradiation test of a stainless-steel-clad tube of uranium oxide has begun. The oxide tube was fabricated at the Savannah River Laboratory by the cold-swaging process that was discussed earlier in this report. The details of these tests will be recorded in a forthcoming classified document.

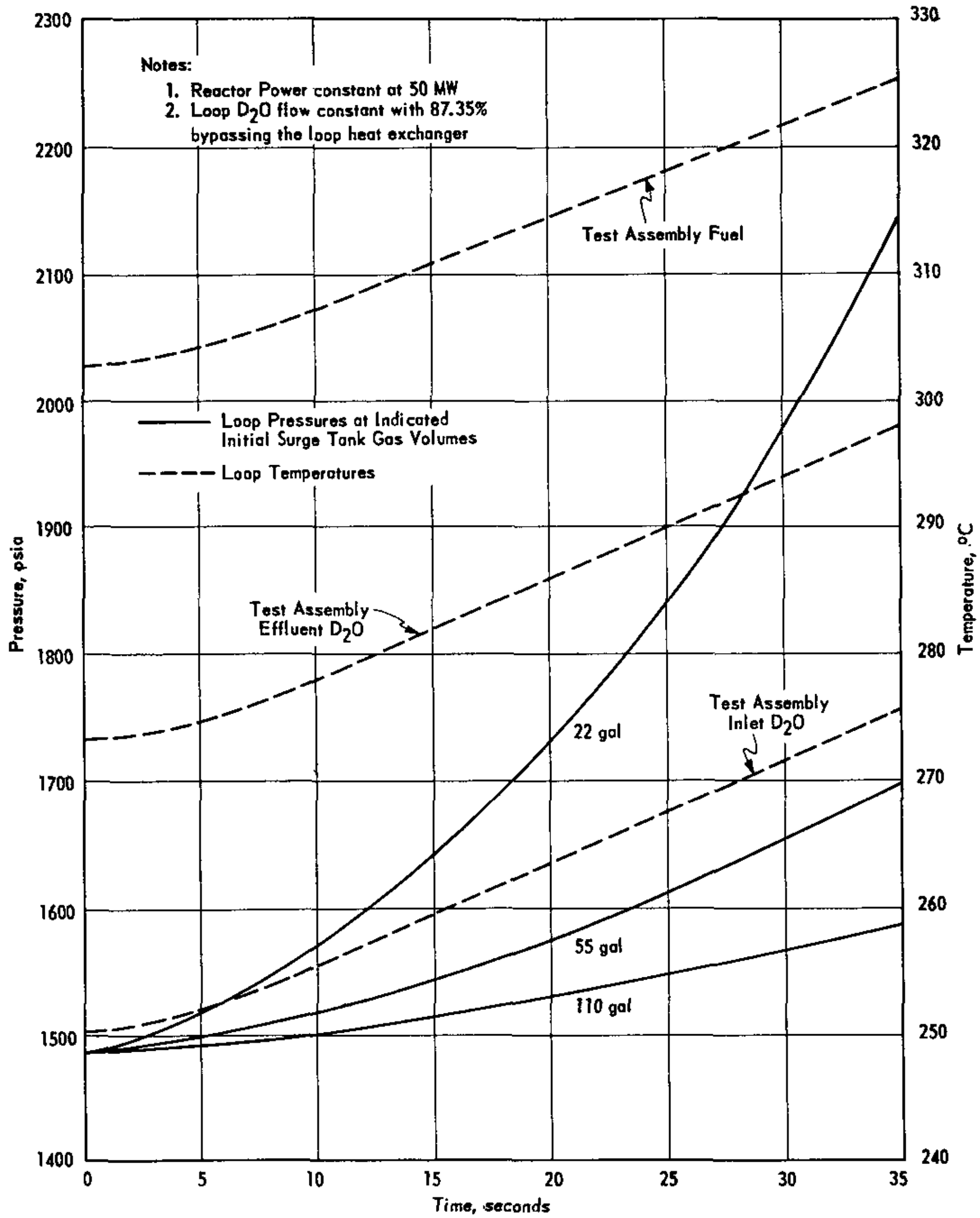

FIG. 1 HWCTR LIQUID-D_2O-COOLED LOOP
Effect of Sudden Loss of H_2O Coolant to Loop Heat Exchanger

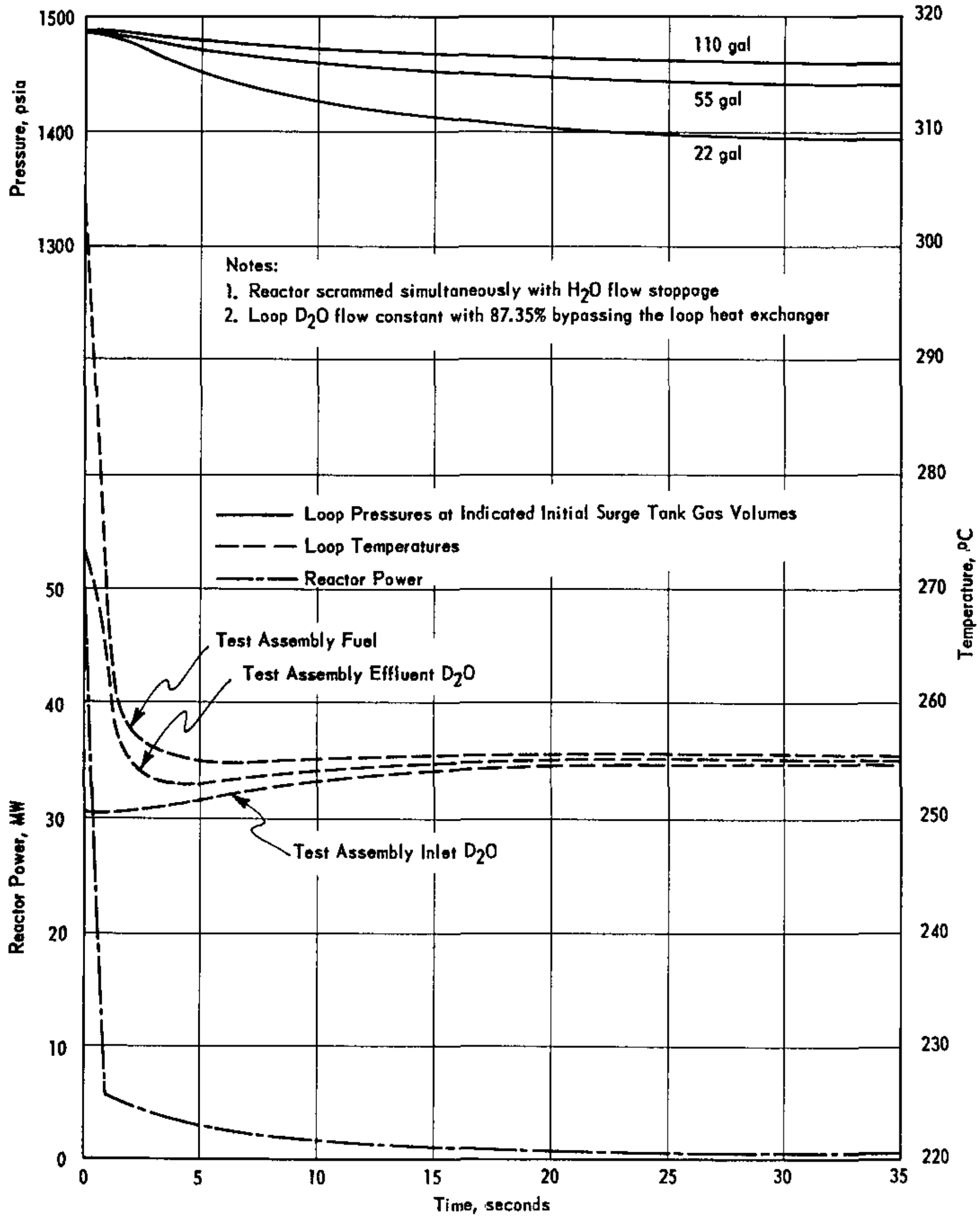

FIG. 2 HWCTR LIQUID-D_2O-COOLED LOOP
Effect of Sudden Loss of H_2O Coolant
to Loop Heat Exchanger

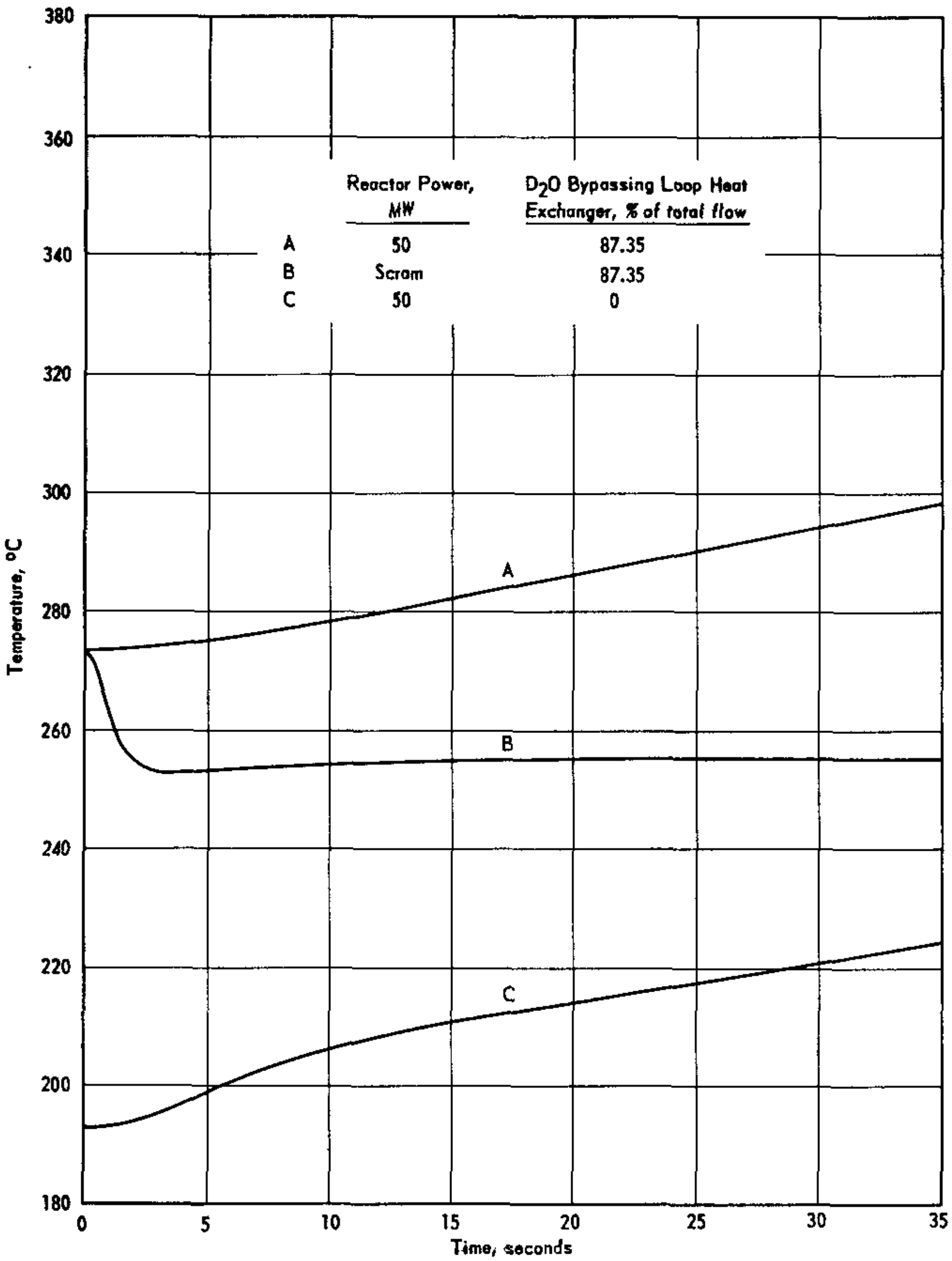

FIG. 3 HWCTR LIQUID-D_2O-COOLED LOOP
Effect of Sudden Loss of H_2O Coolant
to Loop Heat Exchanger on Temperature
of Fuel Assembly Effluent D_2O

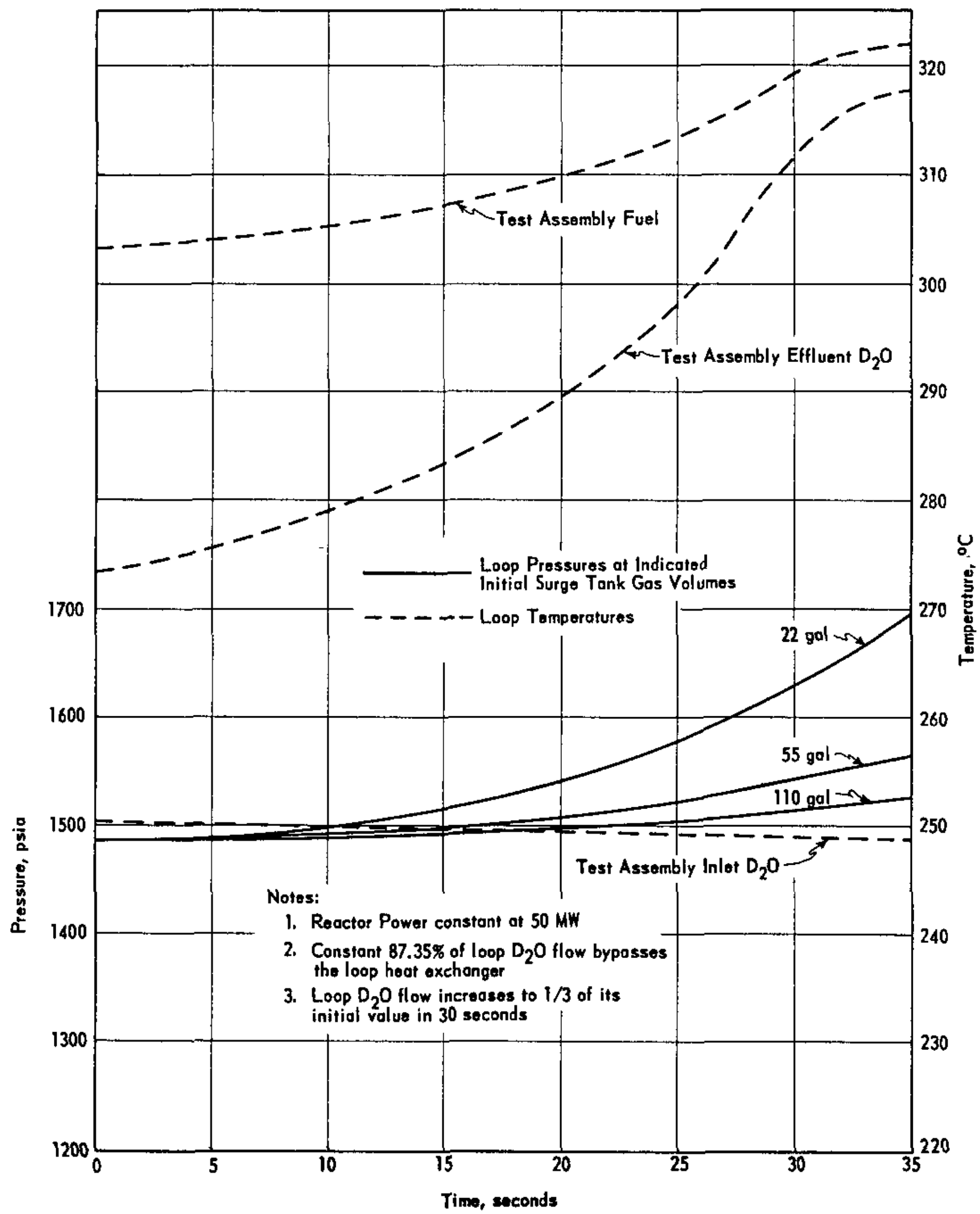

FIG. 4 HWCTR LIQUID - D_2O - COOLED LOOP
Effect of Failure of AC Power Supply to D_2O Pump

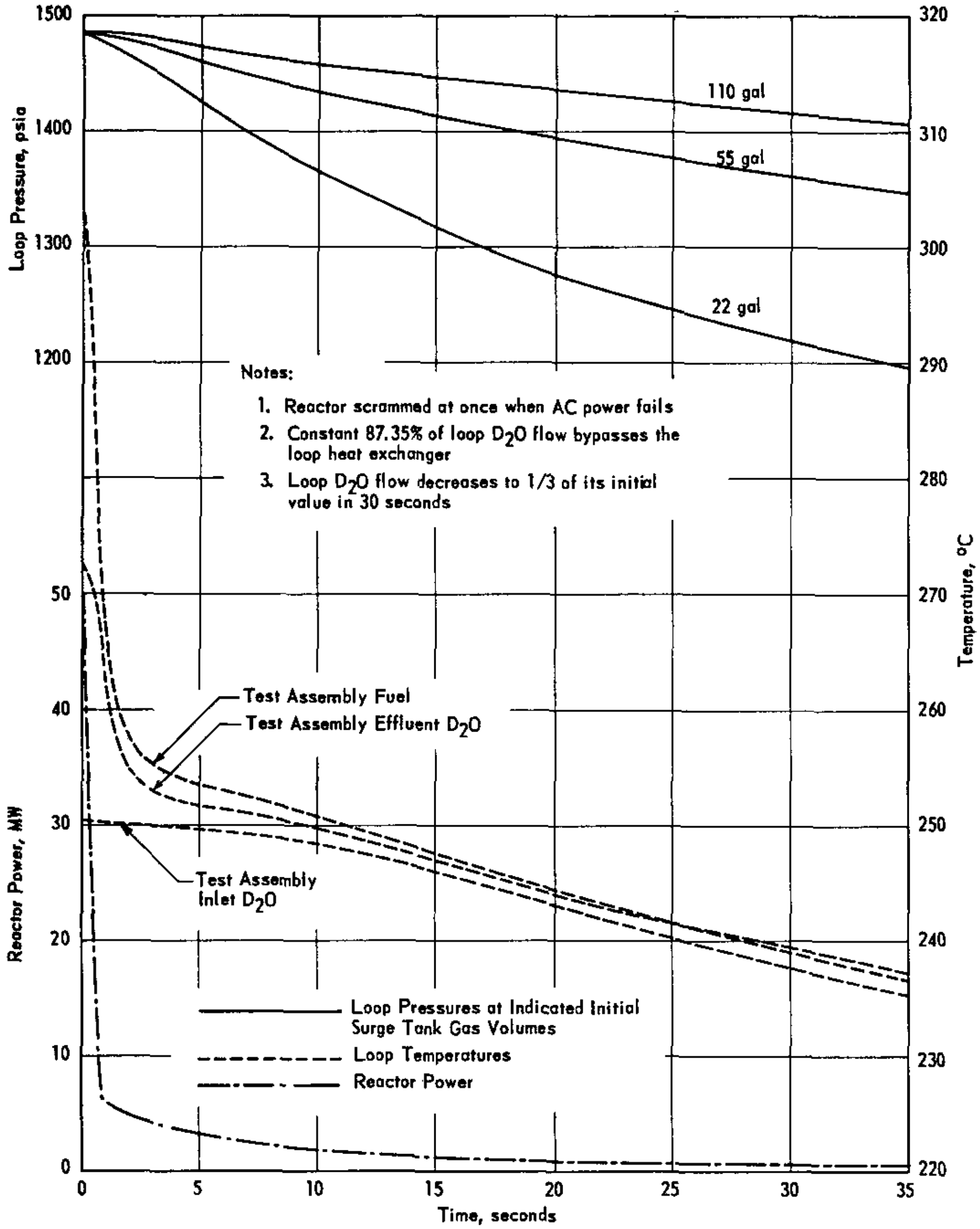

FIG. 5 HWCTR LIQUID-D_2O-COOLED LOOP
Effect of Failure of AC Power Supply to D_2O Pump

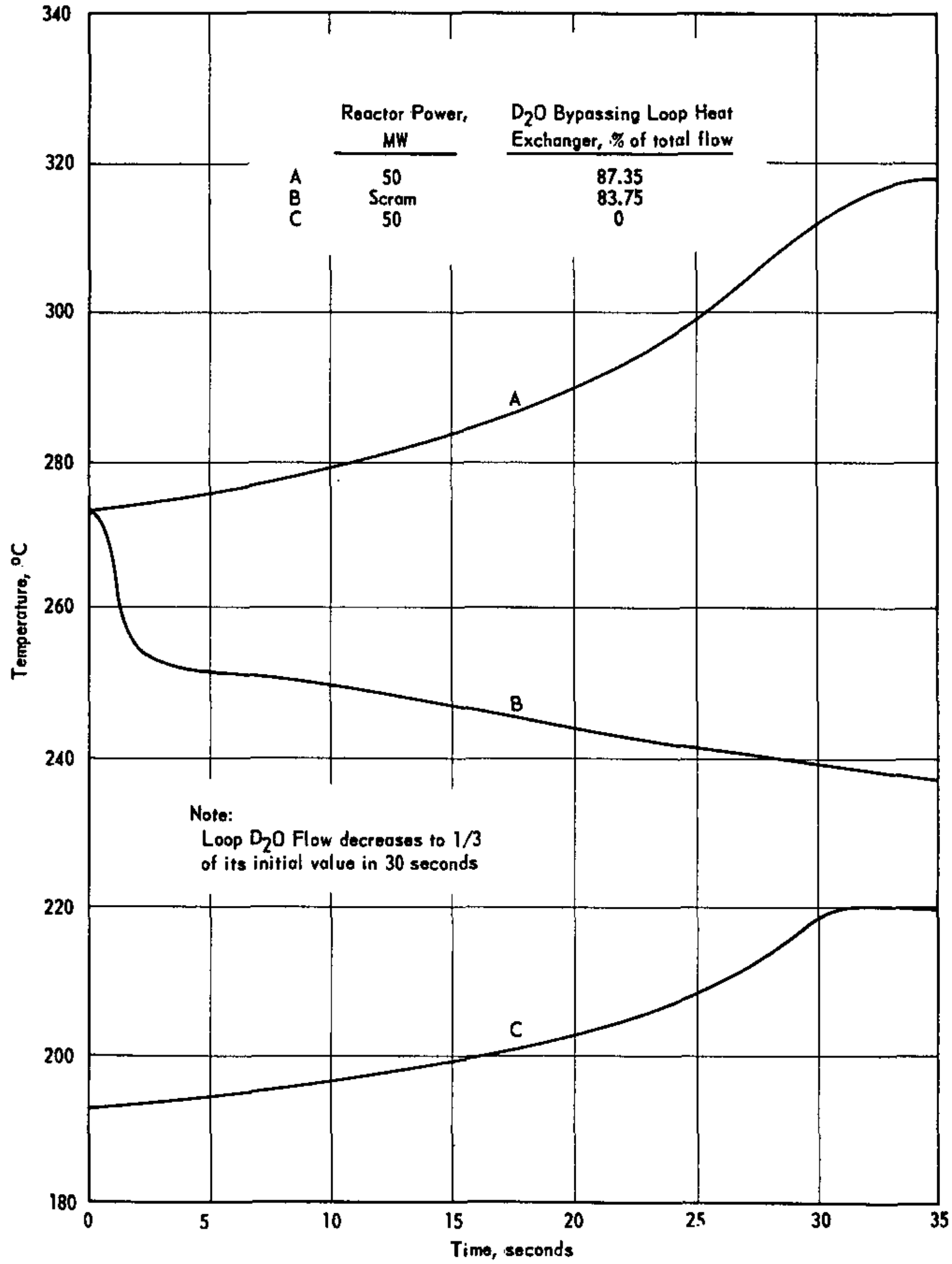

FIG. 6 HWCTR LIQUID-D_2O-COOLED LOOP
Effect of Failure of AC Power Supply to D_2O Pump on Fuel Assembly Effluent Temperature

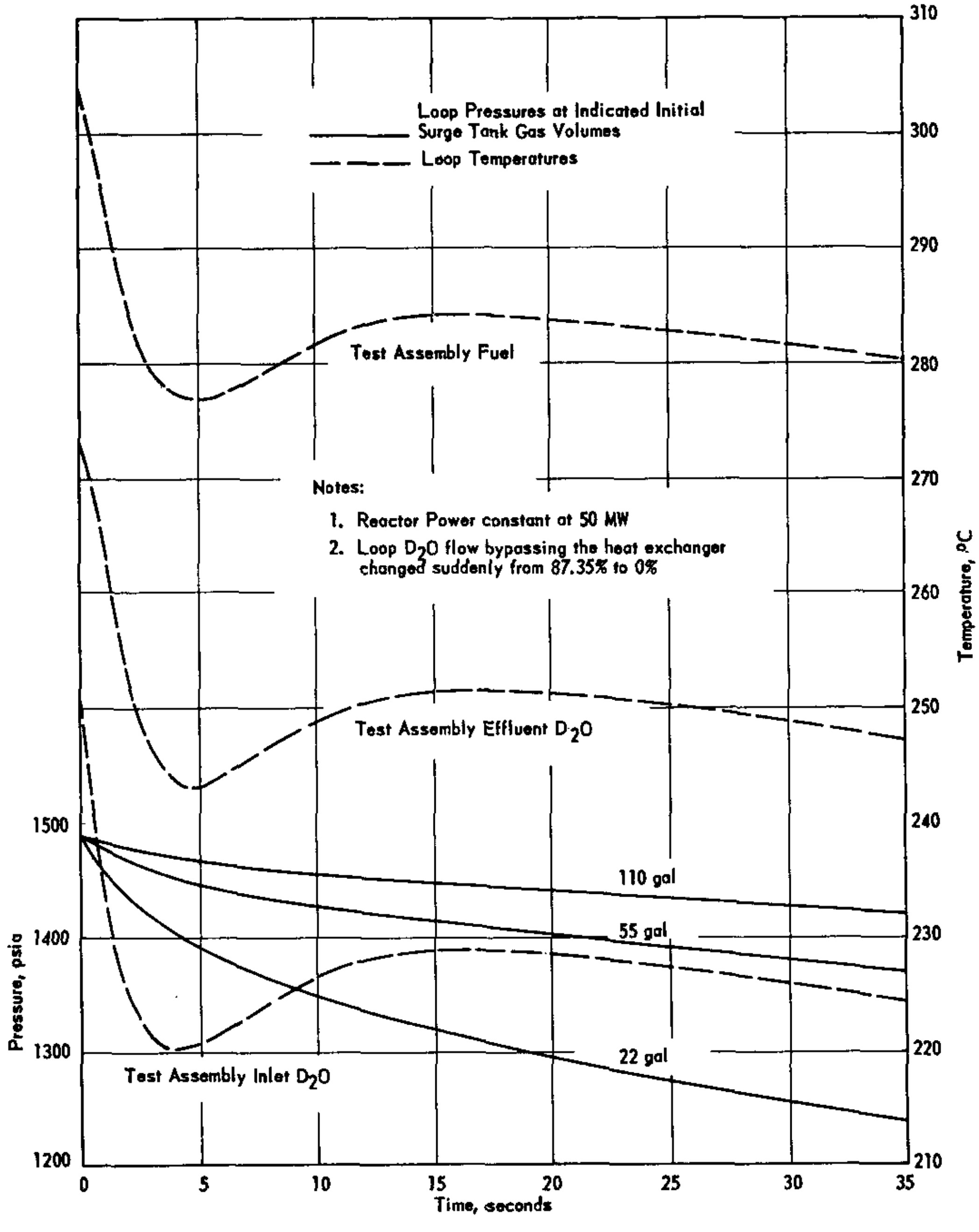

FIG. 7 HWCTR LIQUID-D_2O-COOLED LOOP
Effect of Accidental Step Increase in Cooling

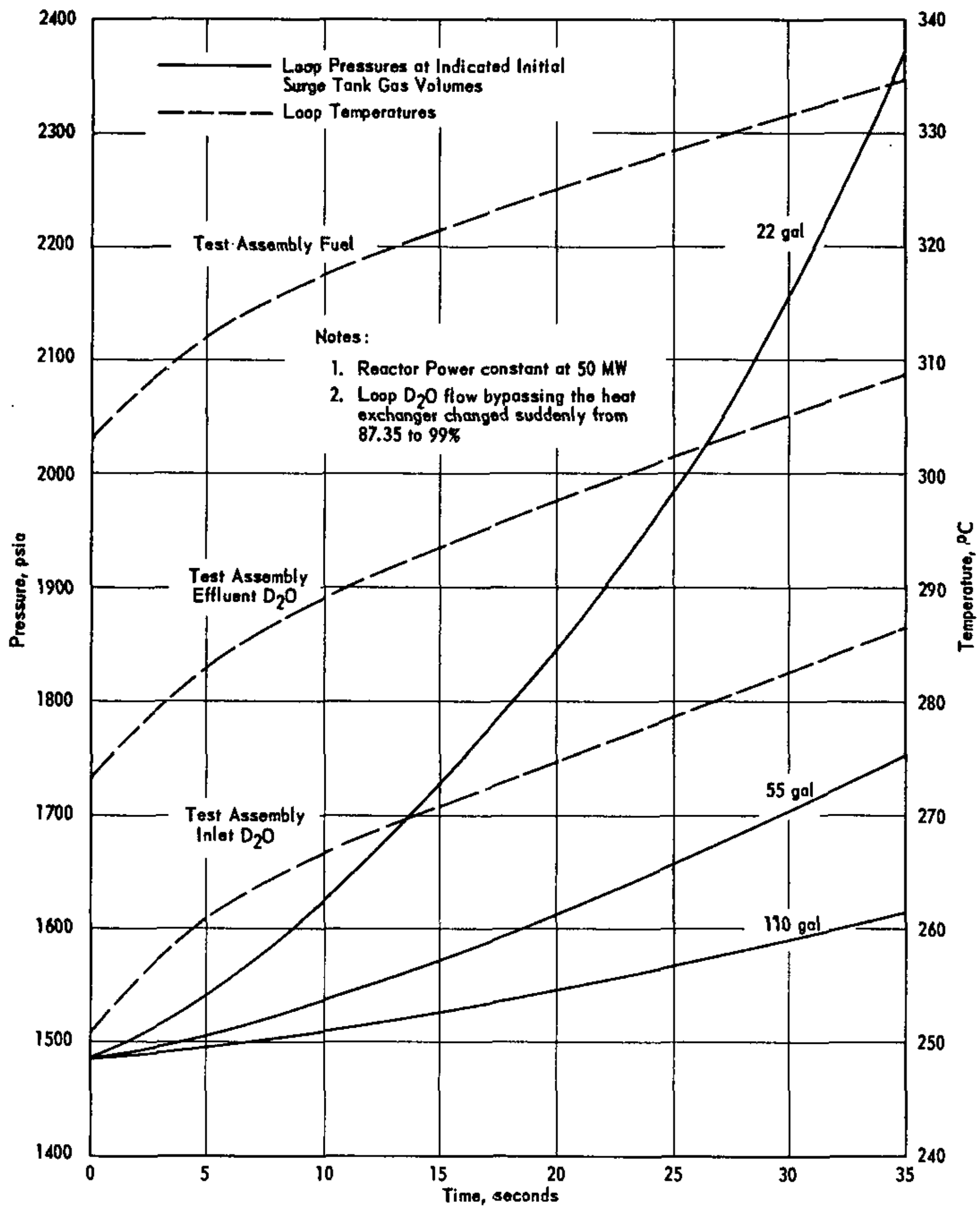

FIG. 8 HWCTR LIQUID-D_2O-COOLED LOOP
Effect of Accidental Step Decrease in Cooling

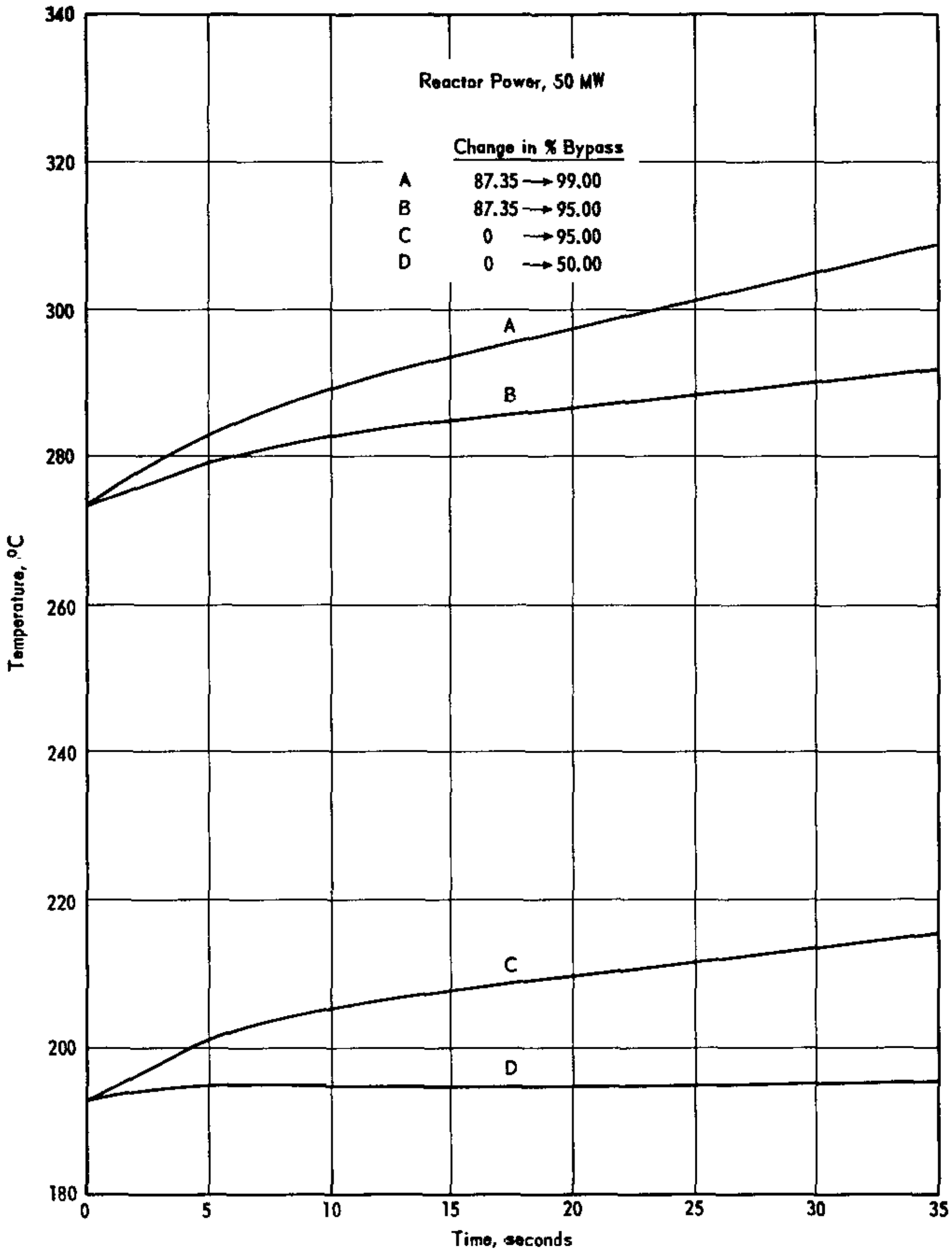

FIG. 9 HWCTR LIQUID-D_2O-COOLED LOOP
Effect of Step Increase in D_2O Flow
Bypassing The Loop Heat Exchanger
(Decreased Cooling) on Temperature
of Fuel Assembly Effluent D_2O

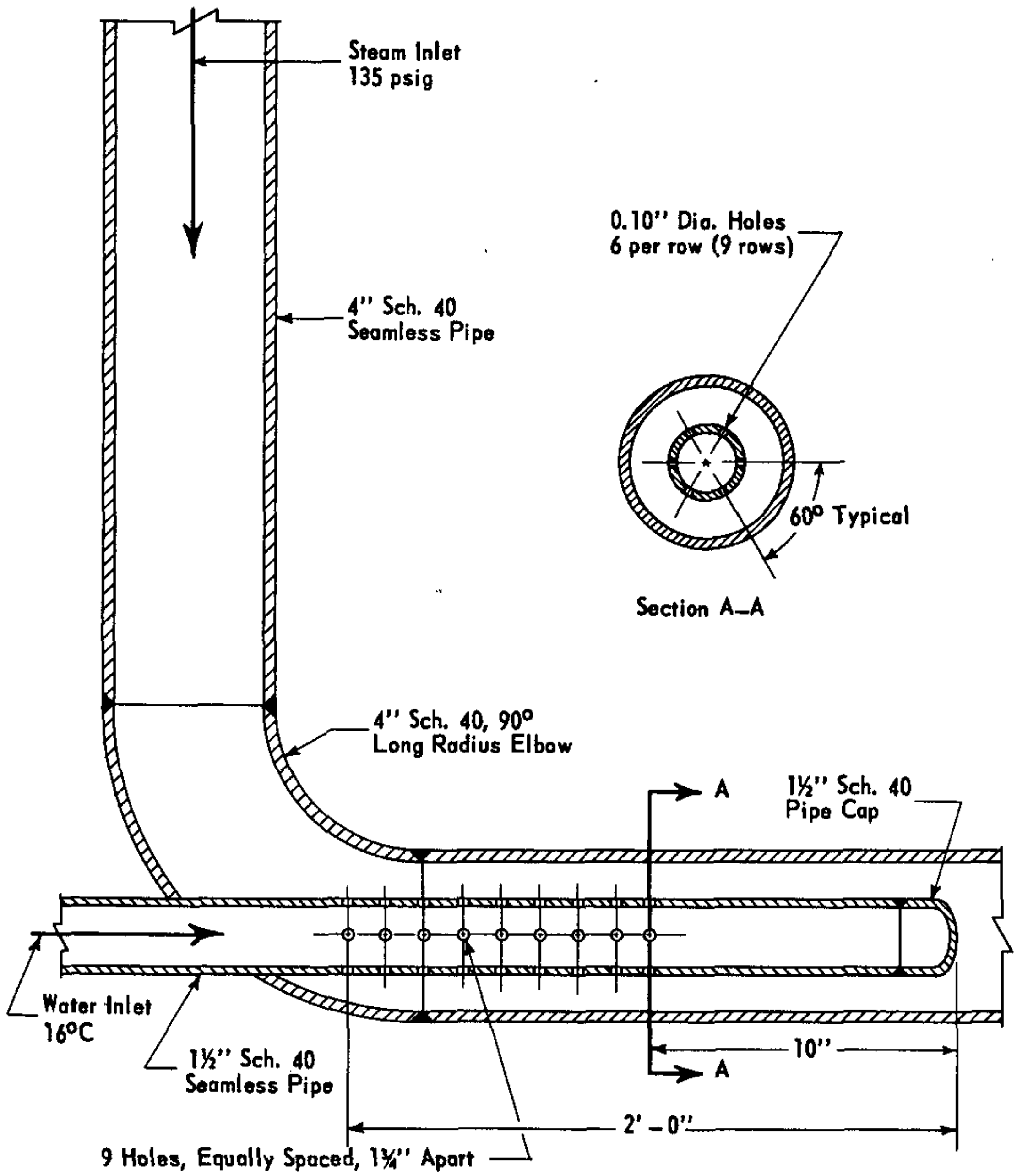

FIG. 10 SCHEMATIC DIAGRAM OF TEST STEAM QUENCHER

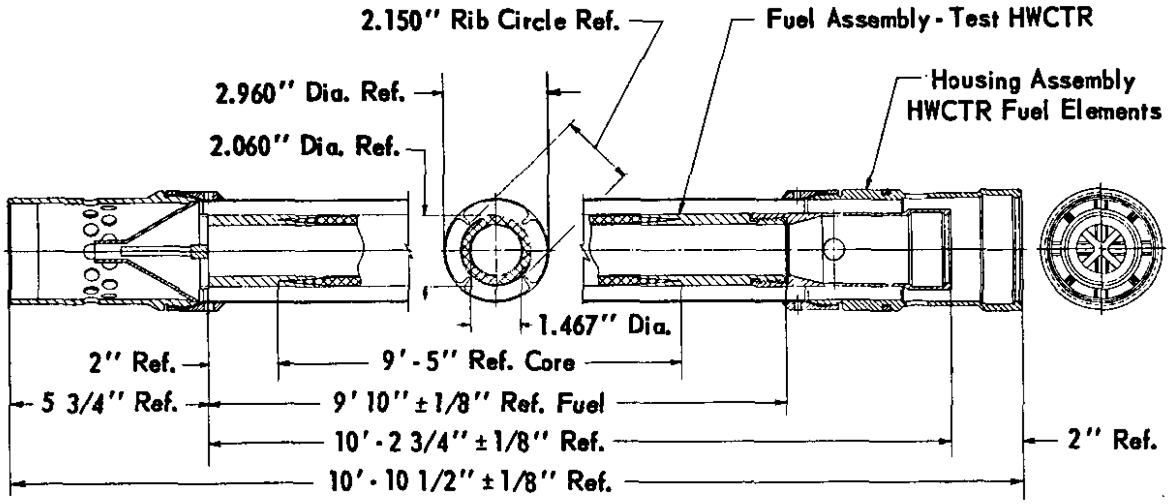

FIG. 11 HWCTR TEST FUEL ASSEMBLY

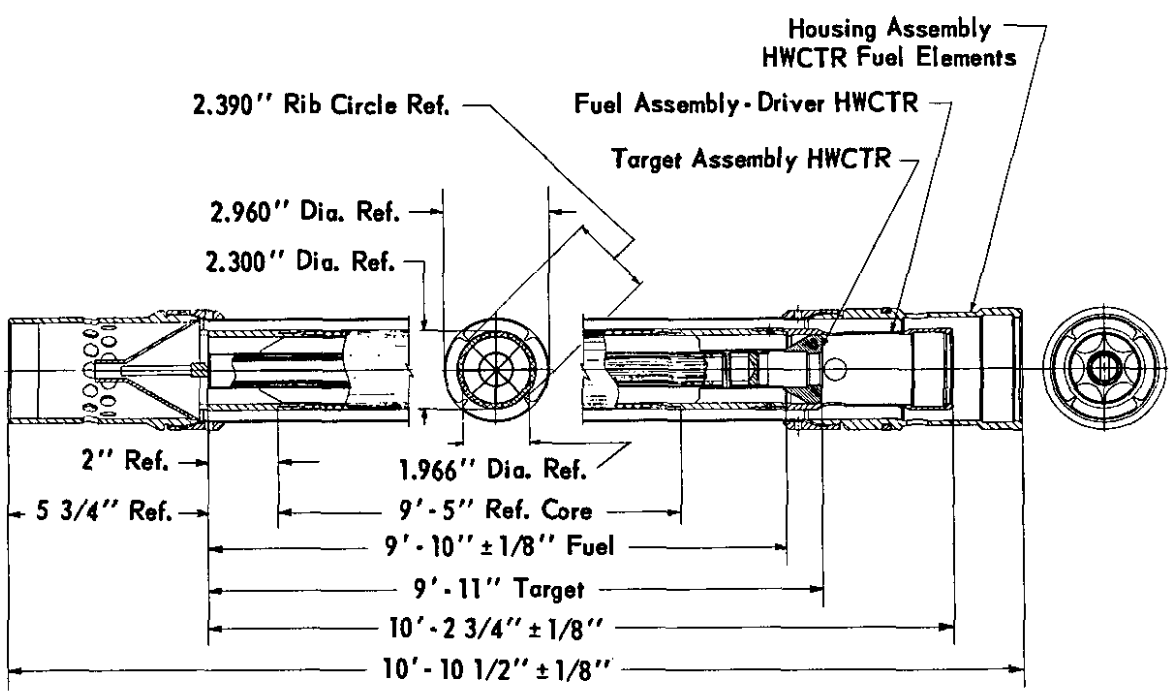

FIG. 12 HWCTR DRIVER FUEL ASSEMBLY