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Reactors - Power

AEC Research and Development Report

PRELIMINARY HAZARDS EVALUATION OF THE ISOLATED COOLANT LOOPS IN THE HWCTR

by

L. M. Arnett, T. F. Parkinson, D. Randall, and C. P. Ross

Savannah River Laboratory

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ABSTRACT

This Preliminary Hazards Evaluation describes the design features of two isolated coolant loops in the Heavy Water Components Test Reactor (HWCTR) and the effect of these features on reactor safety. It is shown that limitations on the operating conditions backed up by properly chosen safety systems will permit operation of the loops without adding appreciably to the reactor hazards described in Reference 1.

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PRELIMINARY HAZARDS EVALUATION OF THE ISOLATED COOLANT LOOPS IN THE HWCTR

INTRODUCTION

The Heavy Water Components Test Reactor (HWCTR) is designed to demonstrate the performance of fuel elements that are candidates for use in power reactors moderated with D_2O and fueled with natural uranium. The HWCTR will have the capability of testing as many as 12 fuel elements, up to 10 feet in length, at temperatures and power ratings equal to those expected in power reactors. The HWCTR is designed to permit the utilization of as many as six positions for isolated loops in which operating conditions may be maintained at levels different from those in the main reactor. It is planned to install two of these isolated loops, with one fuel position in each, at the startup of the HWCTR. The fuel elements in the two loops will be cooled with liquid D_2O and with boiling D_2O, respectively. Detailed descriptions of the HWCTR facility can be found in References 1 through 5.

A "Preliminary Hazards Evaluation of the Heavy Water Components Test Reactor"[1], issued earlier, dealt with the hazards associated with the reactor site and the reactor operation. These details are not repeated in this report, which presents only the additional hazards connected with the operation of the isolated loops. Consequently, this report should be considered an addendum to the earlier Hazards Evaluation[1] and reference to the earlier report will be necessary for a complete description of the entire facility.

The secondary coolant system, which consisted of heat exchangers and a spray pond, described in the previous report[1], has been redesigned to consist of two pressurized steam generators. The redesigned system has the following advantages over the former system: (1) greater economy, (2) elimination of bypass and automatic valves in the heavy water loops, and (3) greater protection against rapid temperature variations during scrams. The D_2O coolant will be pumped at a constant rate through the tubes of the steam generators. The rate of H_2O steam generation, and thus the power of the reactor, will be controlled by regulating valves in the steam discharge lines. The temperature of the D_2O moderator will be controlled by the position of the control rods. This arrangement permits the independent adjustment of the temperature of the D_2O moderator and the reactor power. This design modification does not significantly modify the hazards evaluation presented in the earlier report[1].

The description of the isolated loops corresponds to the design as it existed on April 1, 1960, at which time the design was about 50% complete. Subsequent changes are likely to be made only in details rather than in major features. A Final Hazards Evaluation, which will cover the complete facility, will be presented to the Atomic Energy Commission prior to reactor startup. That evaluation will detail any final design changes and their effects on the safety of the facility.

SUMMARY

Description of the Loops

Two isolated coolant loops, each with a capacity of one fuel assembly, are to be installed in the HWCTR to permit operation at conditions that differ from those that exist in the reactor. The coolant in one loop will be liquid D_2O circulated at pressures up to 500 psi above that in the reactor. The coolant in the other loop will be boiling D_2O at a pressure equal to that in the reactor.

Each loop consists of a Zircaloy bayonet supported in the reactor, and a coolant circulating system. All of the loop equipment and piping outside of the reactor is stainless steel with the exception of the helium piping which is carbon steel. The loops share a low pressure purification system.

Loop Physics

The reactivity changes introduced into the reactor by variation in the operation of the loops are small because the loops occupy only two of the thirty-six positions for fuel elements. Calculations made for the effects of (1) fuel element composition, (2) empty loop positions, and (3) voids indicate that the variation in k_{eff} ranges from plus 0.003 to minus 0.005.

Calculations of the excursions of the pressures and temperatures in the loops that occur during a reactor scram indicate that these variables will follow closely those of the reactor so that no serious differentials will exist.

Loop Operation

During reactor startups and scheduled shutdowns the loops will be operated with their gas pressurization systems connected to that of the main reactor, and with excess heat removal capacity. In this way, the reactor power, temperature, and pressure can be brought up to the nominal operating conditions without having to make concurrent adjustment of conditions in the loops. Thereafter, the desired conditions may be set up in each loop with only negligible effects on conditions in either the reactor or the other loop.

During level operation, conditions in the loops can be controlled either automatically or manually.

Safeguards Analysis

Scrams that are initiated from any of several sources will occur soon enough to prevent serious conditions from developing in the loops should a reactivity transient be initiated in the reactor. The

reactivity additions considered were the same ones that were analyzed in Reference 1 for the main reactor system.

In general, hydraulic accidents in the loops that impair the heat removal capacity produce transients that are arrested by the initiation of a scram before dangerous conditions develop. The exception is a major break in the high pressure system that permits the rapid loss of heavy water. This is the maximum credible accident.

Maximum Credible Accident

The maximum credible accident is a major break in the high pressure D_2O or gas systems and is the same accident discussed in detail for the reactor itself in Reference 1. The operation of the loops increases the probability of this accident but does not increase its severity. Since the loops are designed with the same margin of safety in the pressure specifications as was the reactor, the probability of the maximum credible accident is increased by the addition of the loops in direct proportion to the increase in the amount of high pressure piping.

DISCUSSION

1. DESCRIPTION OF THE LOOPS

1.1 GENERAL

Two isolated coolant loops will be installed and ready for operation at reactor startup. These loops will enable the testing of fuel elements under a wide range of operating conditions. One loop is designed for liquid flow at a pressure 500 psi higher than reactor pressure, and will initially be operated with D_2O. The other loop is to operate with boiling D_2O coolant and can be converted to cooling with D_2O steam or CO_2 with the addition of new auxiliary equipment.

Each loop comprises a sealed housing tube, or bayonet, supported in the reactor, a cooling system for removing the heat from the fuel, and a seal supply system for flushing and cooling the pump seals. The loops share a low pressure purification system. All loop equipment and piping is stainless steel with the exception of the helium piping, which is carbon steel. The gas spaces in the low pressure equipment are connected to the low pressure helium of the main system. The high pressure gas space in the boiling loop surge tank is connected directly to the gas space of the reactor, and the gas space of the liquid loop surge tank is connected to the reactor through a differential pressure control valve. The He and D_2 that escape from the purge streams of the loops and the main purge stream are recompressed and returned to the high pressure side of the control valve. D_2O can be added to the loops from the reactor purge line in an emergency.

The bayonets can be installed in any of the six outer test positions shown in the lattice diagram, Figure 1-2. The two loops that will be installed initially will be placed symmetrically with respect to the center to avoid a flux peak that would occur when the reactor operates with empty bayonets.

Design and operating data for the loops are presented in Table 1-I, and the flow diagram for the loops is shown in Figure 1-1.

1.2 BAYONETS

The proposed bayonet design, which is the same for both loops, is shown in Figure 1-4, and the position of a bayonet in the reactor is shown in Figure 1-3. The bayonet consists primarily of two concentric tubes that form an annulus for the downward flow of coolant and provide a central space for upward flow of coolant through a test fuel assembly. The outer bayonet tube and the top portion of the inner tube are pressure tubes.

The upper section of the bayonet is made of stainless steel and is connected to the inlet and outlet piping and to the Zircaloy lower section by gasketed joints. The connection between the upper and lower sections, shown in Figure 1-5, will be a permanent joint and will effect a stainless-steel-to-Zircaloy transition. Nickel-iron transition sections will be used to provide a gradual change in thermal expansion coefficient through the joint. The lower section of one of the bayonets will be Zircaloy-2 and the other will be Zircaloy-4.

The upper section of the bayonet is capped by a top closure. This closure, shown in Figure 1-6, is removed for charging and discharging fuel assemblies. The fuel assemblies are supported at the bottom by the bottom cap of the outer bayonet tube and are held down by the muff tube, which is attached to the top closure. The inner bayonet tube acts as a guide for the fuel housing tube and the muff tube.

The weight of the bayonet is supported by inlet and outlet connections welded to nozzles in the reactor shell as shown in Figure 1-4. The bayonet is free to expand vertically but is guided at its midplane by the top shield and at the bottom by the bottom thermal shield. With the bayonets in place in the reactor, adequate clearance exists for the removal of all other reactor components.

Loop coolant enters the lower side connection and flows down the annulus formed by the insulating liner and the inner bayonet tube. At the lower end of the inner bayonet tube the flow continues down the annulus formed by the insulating liner and the fuel housing tube. Turning at the bottom of the bayonet, the coolant flows upward through the fuel assembly and out the upper side connection. The insulating liner is a thin, dimpled (or ribbed) tube that makes a tight fit with the outer bayonet tube. The small annulus formed by the dimples contains essentially stagnant D_2O. Small holes are drilled at each end of the liner so the annulus will fill with water.

The bayonet side joints will be connected and disconnected by tools handled from the main floor. Preparations for the replacement of a bayonet will require the removal of the reactor top head, all fuel elements, indexing top shield plug, all control and safety rod guide tubes, top guide plate, and shield plug support ring. The D_2O level in the reactor will be lowered to expose the two connections. The upper section of the bayonet will be sawed off and removed by the crane. The lower section will be radioactive enough to require shielding and will be removed to the spent fuel basin by a special gripper in the transfer coffin.

The stainless steel portions of the bayonets will meet the requirements of the ASME Unfired Pressure Vessel code. The Zircaloy portions will be designed on the basis of Non-Mandatory Appendix Q of the code. Design temperatures and pressures are listed in Table 1-I. The design exposure life of the bayonets is 5 years in a thermal neutron flux of 3.9×10^{14} n/(cm^2)(sec) and a fast flux of 5.2×10^{13} n/(cm^2)(sec). They are also designed to withstand a minimum of 100 scram cycles and the corresponding temperature and pressure transients as given in Table 1-II.

The bayonets will be chemically resistant to D_2O with a pD of 10 to 11 on the outside surfaces and 7 to 11 on the inside surfaces. The oxygen content should be less than 0.1 ppm to prevent corrosion of the nickel-iron transition sections. A carbon steel mock-up of a bayonet, its nozzle supports, and adjacent piping will be subjected to severe conditions of steam-water flow to determine whether potentially damaging vibration will occur.

1.3 PRIMARY FLOW SYSTEM

1.3.1 Liquid Loop

Liquid D_2O is circulated in this loop at a pressure that is as much as 500 psi higher than the pressure in the reactor vessel. The effluent from the bayonet passes through a heat exchanger, surge tank, and two pumps before returning to the bayonet. The pumps can be operated in series or in parallel. Loop pressure is controlled by a differential pressure valve in the line connecting the gas space of the surge tank to the main system. Protection of the bayonet from excessive pressure differential is provided by positive and negative pressure rupture discs, designed to rupture at +700 and -200 psi, respectively.

1.3.2 Boiling Loop

In the boiling loop, D_2O is pumped as a liquid to the bayonet and is removed as a liquid-steam mixture at reactor pressure. The mixture is quenched by cold D_2O in a quencher just downstream of the bayonet, and the subcooled liquid passes through a surge tank and two pumps. Part of this flow returns directly to the bayonet, and the rest passes through a heat exchanger and becomes the quench water for the two-phase bayonet effluent. A separate bypass is provided around the heat exchanger so that the pressure drops across the throttling valves to the bayonet and the heat exchanger can be regulated.

A high pressure test loop has been constructed by the Savannah River Laboratory to obtain hydraulic and heat transfer data pertinent to the operation of this boiling loop.

1.4 PURIFICATION

Separate purge streams from each loop pass through purge coolers and pressure-reducing valves and are combined to provide the feed to the loop purification system. The purge flow is controlled by the liquid level in the surge tank. The purification system consists of a pump, a deionizer in the Li-OD form, a deionizer in the D-OD form, and an afterfilter. Provision has been made for future installation of a prefilter. Normal flow to purification is about 2 gpm from each loop.

1.5 SEAL SUPPLY AND MAKEUP

The purification effluent feeds the seal water supply pumps. Each loop has its own high head seal pump and its own head tank, which is located at the +52 foot level. The seal pump supplies the head tank through a pressure control valve so that when the loop circulating pumps are operated in series the downstream pump seal is supplied from the high pressure side of the control valve and the upstream pump seal is supplied from the head tank side of the control valve. The valve setting is controlled by the difference between the seal water pressure and the suction pressure of the downstream pump. When the pumps are operated in parallel, the head tank provides a constant supply to both seals. The gas spaces of the head tank and surge tank are connected, and overflow from the head tank flows down to the surge tank.

An alternate supply of water for the purification system and seal supply system is the loop storage tank. Provision is also made so that either or both loops can use the purification system of the reactor.

1.6 SECONDARY COOLANT

The circulating D_2O in each loop passes through the shell side of a heat exchanger. H_2O flows by gravity through the tubes of the exchanger to remove all the heat generated in the loop fuel assembly except the small fraction that appears as moderator heating. The H_2O is used on the tube side so that a high velocity can be maintained to prevent boiling. The source of the H_2O, which is the same that services the main reactor system, is a new 70-foot-high, 36-inch-diameter standpipe that is supplied by an existing 150,000-gallon elevated storage tank. After passing through these heat exchangers and other miscellaneous exchangers in the building, the water is recovered and used as feedwater for the main steam generators. The secondary cooling system is shown in Figure 1-7.

1.7 SAFETY CIRCUITS

The loop safety circuits in general duplicate the safety circuits of the main system. Table 1-III is a summary of the loop conditions that cause the reactor to scram, and Table 1-IV lists the annunciator circuits. Each of the conditions, including the scram items, sounds an audible alarm and lights an identifying plate on the annunciator panel in the control room.

1.8 POWER SUPPLY

The normal and emergency power supplies for the loops are the same as for the reactor. Each loop has two main circulating pumps, one of which is on the emergency AC power supply. Each pump has, in addition to its AC motor, a DC motor that is continuously energized and continues to turn the pump at one-third speed should the AC motor fail. A flywheel assures a flow decay curve that overrides the heat decay of the fuel after a scram. One pump driven by its DC motor provides enough flow for shutdown cooling.

1.9 LOCATION OF EQUIPMENT

The placement of the loop equipment in the building is shown in Figure 1-8. The liquid loop circulating pumps, heat exchanger, and surge tank are in the left (east) pump room and its purge cooler is in the left generator room. The boiling loop surge tank is in the right pump room and its pumps, heat exchanger, and purge cooler are in the right generator room. The seal water pumps are in the left purification room along with the high head pumps that supply seal water and makeup to the main system. The loop storage tank and purification pump are also in the left purification room, and the loop deionizers and filter are in the left generator room. The seal supply tanks are supported at the +52 foot level along with the seal supply tank for the main pumps.

TABLE 1-I

Design and Operating Data
for Isolated Coolant Loops of the HWCTR

Loop No.	1	2
Coolant	D_2O (Liquid)	D_2O (Boiling)
Reactor conditions		
Power, MW	61	30 - 50
Nominal operating pressure, psig	1000	780
Design pressure, psig	1500	1500
Inlet temperature, °C	214	200
Outlet temperature, °C	239	225 - 242
Maximum outlet temperature, °C	290	290
Loop conditions		
Power, MW	1.80	1.43
Operating pressure, psig	1500	780
Maximum pressure, psig(c)	1800	1000
Minimum pressure, psig(c)	845	660
Design pressure, psig	2200	1500
Inlet operating temperature, °C	250	258
Inlet minimum temperature, °C	-	219
Outlet operating temperature, °C	274	269
Outlet maximum temperature, °C	310	310
Coolant flow, lb/hr	114,000	45,500
ΔP fuel passage, psi	15.3	20
Quality of coolant, %	0	14(e)
Bayonet pressures		
External maximum pressure, psi(a)(c)	75	(b)
External design pressure, psi(a)(d)	200	200
Internal operating pressure, psi(a)	500	none
Internal maximum pressure, psi(a)(c)	650	(b)
Internal design pressure, psi(a)(d)	700	700

(a) Pressure given is ΔP - bayonet to reactor
(b) Loop and reactor have common pressurization systems
(c) Reactor scram conditions
(d) Pressure relief to reactor
(e) Quality variable over a 0% to 30% range

TABLE 1-II

Calculated Scram Conditions

Time, minutes	Coolant Outlet Temp, °C		
	Reactor	Liquid Loop	Boiling Loop
0	225	274	269
0.5	220	230	260
1	214	190	253
2	202	127	239
3	192	97	225
4	184	72	210
5	176	---	198
10	145	---	150
15	128	---	112
20	118	---	87
25	109	---	68
30	100	---	---

TABLE 1-III

Scram Circuits for Isolated Coolant Loops

Item	Tentative Setting	Actuating Signal
Liquid Loop		
High differential pressure	+100 to +200 psi on loop-reactor ΔP	ΔP across differential pressure control valve
Low differential pressure	-100 to -200 psi on loop-reactor ΔP	ΔP across differential pressure control valve
Pump power failure	Under voltage	Loss of power to AC motor of either circulating pump
Low process water flow	-8%	ΔP across orifice meter
High bayonet effluent temperature	+5°C	Thermocouple in bayonet effluent line
Low surge tank level	-10%	ΔP across purge lines
Low cooling water flow	-20%	ΔP across orifice meter
Boiling Loop		
Pump power failure	Under voltage	Loss of power to AC motor of either circulating pump
Low process water flow	-8%	ΔP across orifice meter
Low quencher ΔT	-5°C	ΔP across thermocouples in quencher influent and effluent lines
Low surge tank level	-10%	ΔP across purge lines
Low cooling water flow	-20%	ΔP across orifice meter
Low bayonet ΔP	-50%	ΔP across taps in bayonet influent and effluent lines

TABLE 1-IV

Annunciator Circuits for Isolated Coolant Loops

Item	Tentative Setting	Actuating Signal
Liquid Loop		
Process water activity	0 to 100%, adjustable	Process water gamma monitor
Gas activity	0 to 100%, adjustable	Gaseous fission product monitor
Differential pressure	-100 to +100 psi on loop-reactor ΔP	ΔP across differential pressure control valve
Pressure	-100 to +100 psi	Pressure in gas space
Seal pressure abnormal	-5 to +5 psi	ΔP across seal water supply and suction of downstream pump
Pump power	Power loss	Loss of power to DC motor of either circulating pump
Seal pump power	Under voltage	Loss of power to seal water pump
Process water flow	-5%	ΔP across orifice meter
High coolant ΔT (rod reversal)	+3°C	ΔT across thermocouples in bayonet influent and effluent lines
Coolant ΔT	+2°C	ΔT across thermocouples in bayonet influent and effluent lines
Bayonet effluent temperature	+2°C	Thermocouple in bayonet effluent line
Surge tank influent temperature	-5°C	Thermocouple in surge tank influent line
Surge tank level	-5 to +5%	ΔP across purge lines
Seal tank level	6 inches below overflow	Liquid head in tank
Pump lube abnormal		Lube pressure, temperature, or level switch on either circulating pump
Cooling water flow	-10%	ΔP across orifice meter
Seal pump lube abnormal		Lube pressure, temperature, or level switch on seal water pump

TABLE 1-IV (Continued)

Item	Tentative Setting	Actuating Signal
Boiling Loop		
Process water activity	0 to 100%, adjustable	Process water gamma monitor
Gas activity	0 to 100%, adjustable	Gaseous fission product monitor
Seal pressure abnormal	-5 to +5 psi	ΔP across seal water supply and suction of downstream pump
Pump power	Power loss	Loss of power to DC motor of either circulating pump
Seal pump power	Under voltage	Loss of power to seal water pump
Process water flow	-5%	ΔP across orifice meter
Quencher ΔT	-2°C	ΔT across thermocouples in quencher influent and effluent lines
Surge tank level	-5 to +5%	ΔP across purge lines
Seal tank level	6 inches below overflow	Liquid head in tank
Pump lube abnormal		Lube pressure, temperature, or level switch on either circulating pump
Cooling water flow	-10%	ΔP across orifice meter
Seal pump lube abnormal		Lube pressure, temperature, or level switch on seal water pump
Bayonet ΔP	-25%	ΔP across taps in bayonet influent and effluent lines
Both Loops		
Storage tank level	-6 to +6 inches	ΔP across purge lines

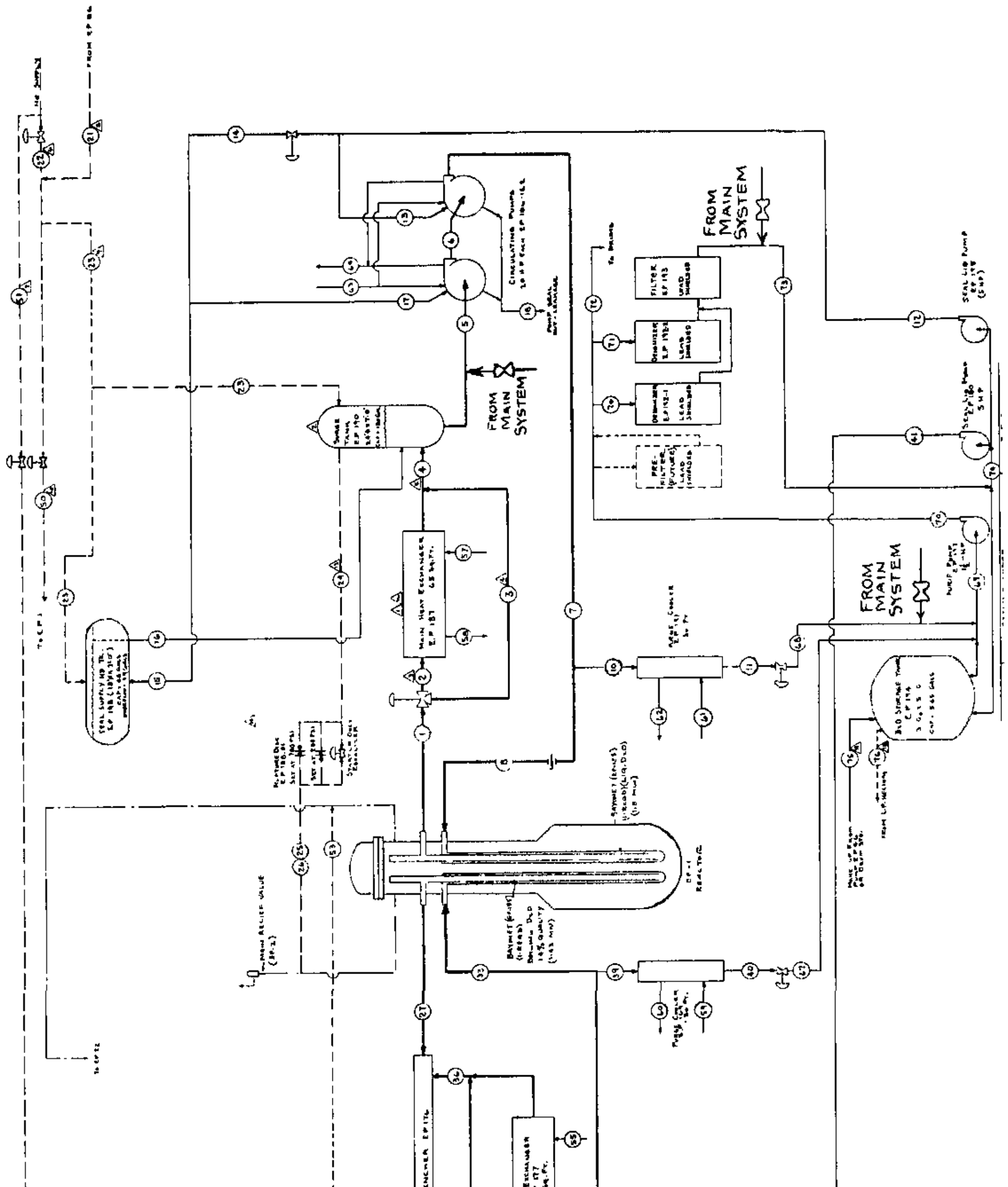


FIG. 1-1 ISOLATED COOLANT LOOPS FLOW DIAGRAM

Liquid D_2O Loop

	Line No.	1	2	3	4	5	6	7	8	9	10	11	12	13	14
1	Mat'l.	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	-	D_2O	D_2O	D_2O	D_2O	D_2O
2	Oper. press. psig or as noted	1463	1460	1449	1449	1446	1487	1523	1500	-	1523	1523	1530	1530	1488
3	Max. press. psig or as noted	2200	2200	2200	2200	2200	2200	2200	2200	-	2200	2200	2200	2200	2200
4	Operating temp., °C	274	274	274	250	250	250	250	250	-	250	30	30	30	30
5	Maximum temp., °C	294	294	294	270	270	270	270	270	-	270	45	45	45	45
6	Oper. flow, gpm, or as noted	260	As Req'd	As Req'd	260	261	261.5	262	260	-	2.0	2.0	2.0	0.5	1.5
7	Max. flow, gpm, or as noted	300	300	300	300	300	300	300	300	-	4.0	4.0	2.0	2.0	2.0
8	Scram flow, gpm, or as noted	-	27	234	-	-	-	-	-	-	-	-	-	-	-

	Line No.	15	16	17	18	19	20	21	22	23	24	25
1	Mat'l.	D_2O	D_2O	D_2O	D_2O	-	-	He + D_2	He	He + D_2	He	He
2	Oper. press. psig or as noted	1463	1463	1488	10" H_2O	-	-	1460	1460	1460	1460	1000
3	Max. press. psig or as noted	2200	2200	2200	10" H_2O	-	-	2200	2200	2200	2200	1500
4	Operating temp., °C	30	30	30	30	-	-	Amb.	Amb.	Amb.	Amb.	Amb.
5	Maximum temp., °C	45	45	45	80	-	-	Amb.	Amb.	Amb.	294	294
6	Oper. flow, gpm, or as noted	1.0	1.0	0.5	Neg.	-	-	As Req'd	-	Neg.	-	-
7	Max. flow, gpm, or as noted	2.0	2.0	2.0	1.0	-	-	8.0 SCFM	4.0 SCFM	8.0 SCFM	-	-
8	Scram flow, gpm, or as noted	-	-	-	-	-	-	-	-	-	-	-

Boiling D_2O Loop

	Line No.	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40
1	Mat'l.	He	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	-	-	D_2O	D_2O
2	Oper. press. psig or as noted	780	780	780	777	814	856	815	812	790	780	780	-	-	815	815
3	Max. press. psig or as noted	1500	1500	1500	1500	1500	1500	1500	1500	1500	1500	1500	-	-	1500	1500
4	Operating temp., °C	Amb.	269	258	258	258	258	258	258	258	258	134	-	-	258	30
5	Maximum temp., °C	300	300	300	300	300	300	300	300	300	300	300	-	-	300	45
6	Oper. flow, gpm, or as noted	-	45500 #/HR	152	153	153.5	154	107	105	47	0	47	-	-	2.0	2.0
7	Max. flow, gpm, or as noted	-	71000 #/HR	300	300	300	300	164	162	220	200	250	-	-	4.0	4.0
8	Scram flow, gpm, or as noted		Quality 14%	-	-	-	-	-	-	-	-	-	-	-	-	-

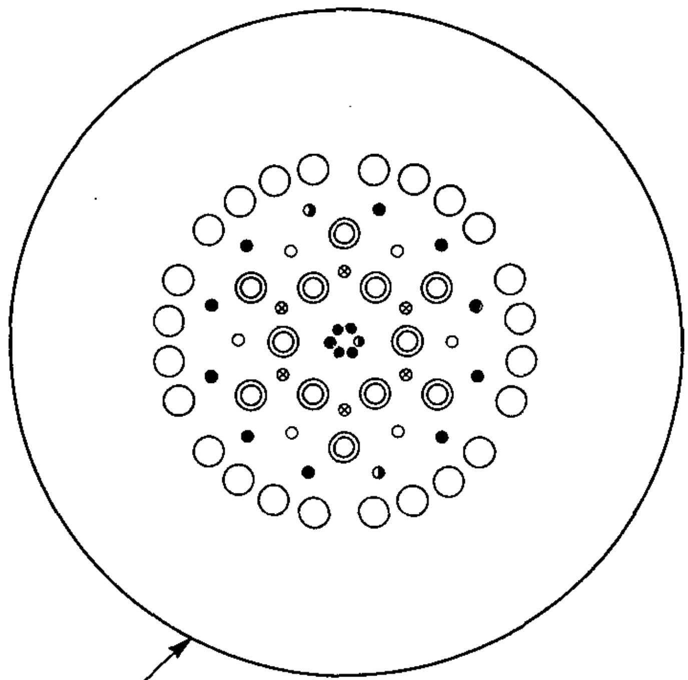
	Line No.	41	42	43	44	45	46	47	48	49	50	51	52	53	54
1	Mat'l.	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	-	-	He + D_2	He	He	He + D_2	He + D_2
2	Oper. press. psig or as noted	847	847	805	780	780	805	10" H_2O	-	-	780	1700	780	780	780
3	Max. press. psig or as noted	1500	1500	1500	1500	1500	1500	10" H_2O	-	-	1500	2000	1500	1500	1500
4	Operating temp., °C	30	30	30	30	30	30	30	-	-	Amb.	Amb.	Amb.	Amb.	Amb.
5	Maximum temp., °C	45	45	45	45	45	45	80	-	-	Amb.	Amb.	Amb.	300	Amb.
6	Oper. flow, gpm, or as noted	2.0	0.5	1.5	1.0	1.0	0.5	Neg.	-	-	Neg.	Neg.	Neg.	Neg.	Neg.
7	Max. flow, gpm, or as noted	2.0	2.0	2.0	2.0	2.0	2.0	1.0	-	-	8.0 SCFM	4.0 SCFM	4.0 SCFM	70 CFM	4.0 SCFM
8	Scram flow, gpm, or as noted	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Cooling Water

	Line No.	55	56	57	58	59	60	61	62	63	64	65	66
1	Mat'l.	C.W.	C.W.	C.W.	C.W.	C.W.	C.W.	C.W.	C.W.	C.W.	C.W.	C.W.	C.W.
2	Oper. press. psig or as noted	50	40	50	40	50	40	50	40	50	40	50	40
3	Max. press. psig or as noted	50	50	50	50	50	50	50	50	50	50	50	50
4	Operating temp., °C	22	52	22	52	22	52	22	52	22	28	22	28
5	Maximum temp., °C	22	52	22	52	22	52	22	52	22	28	22	28
6	Oper. flow, gpm, or as noted	180	180	225	225	50	50	50	50	20	20	20	20
7	Max. flow, gpm, or as noted	200	200	250	250	50	50	50	50	20	20	20	20
8	Scram flow, gpm, or as noted	-	-	-	-	-	-	-	-	-	-	-	-

Purification

	Line No.	67	68	69	70	71	72	73	74	75	76	77
1	Mat'l.	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	He + D_2	-
2	Oper. press. psig or as noted	10" H_2O	10" H_2O	10" H_2O	50	50	50	10" H_2O	10" H_2O	50	10" H_2O	-
3	Max. press. psig or as noted	10" H_2O	10" H_2O	10" H_2O	50	50	50	10" H_2O	10" H_2O	50	10" H_2O	-
4	Operating temp., °C	30	30	30	30	30	30	30	30	30	Amb.	-
5	Maximum temp., °C	45	45	45	45	45	45	45	45	30	Amb.	-
6	Oper. flow, gpm, or as noted	2.0	2.0	4.0	4.0	0	0	4.0	4.0	As Req'd	Neg.	-
7	Max. flow, gpm, or as noted	4.0	4.0	6.0	6.0	6.0	6.0	6.0	4.0	15	4.0 SCFM	-
8	Scram flow, gpm, or as noted	-	-	-	-	-	-	-	-	-	-	-



76" diameter inside shield

◯ Driver fuel assembly (enriched uranium)

◎ Test fuel assembly (natural uranium)
(Outer six positions can hold bayonets)

Control Rods

- ● Black full rod
- ◑ Black half rod
- ● Gray full rod

o Safety rod

⊗ Instrument position

FIG. 1-2 LATTICE DIAGRAM

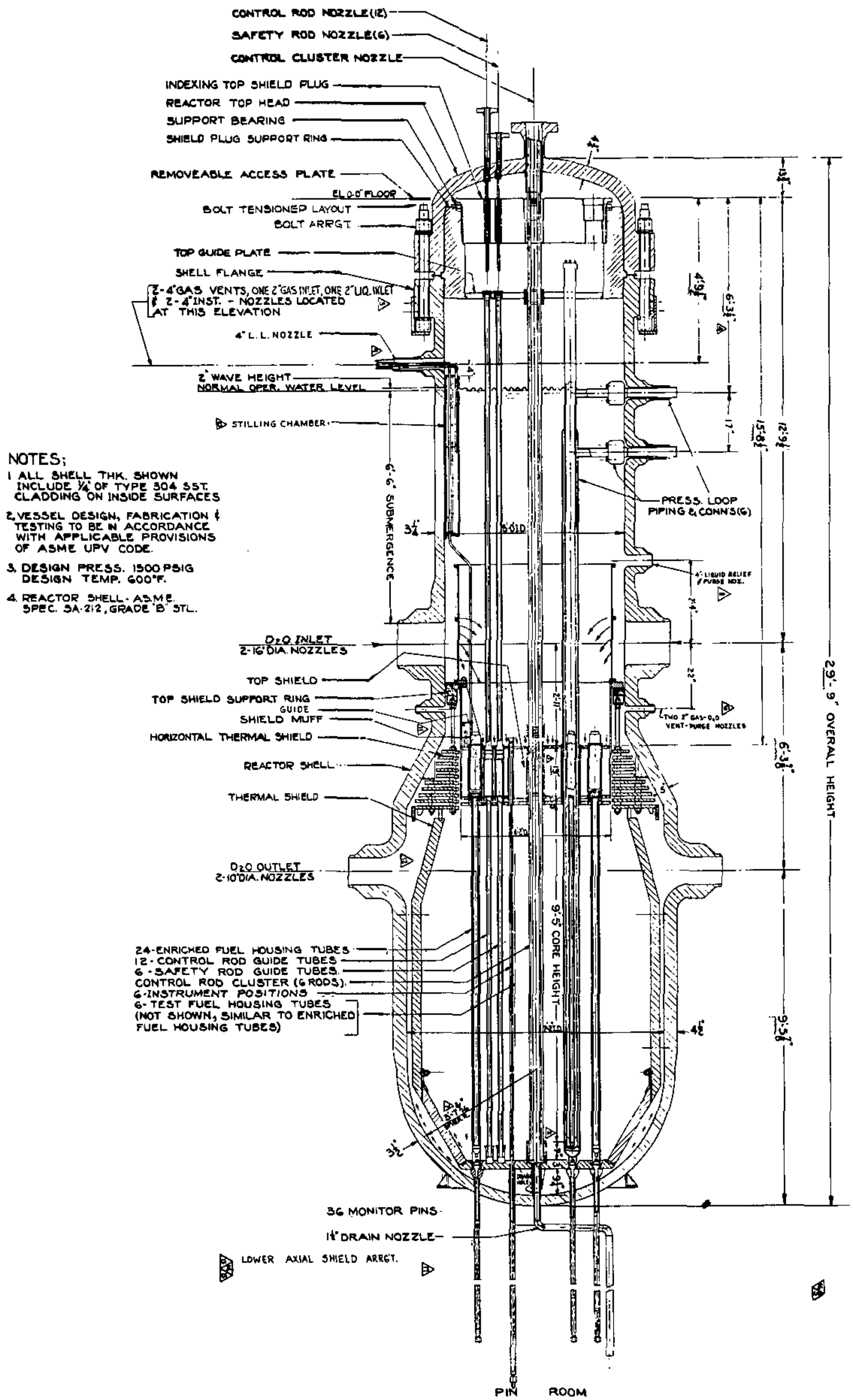


FIG. 1-3 REACTOR ARRANGEMENT

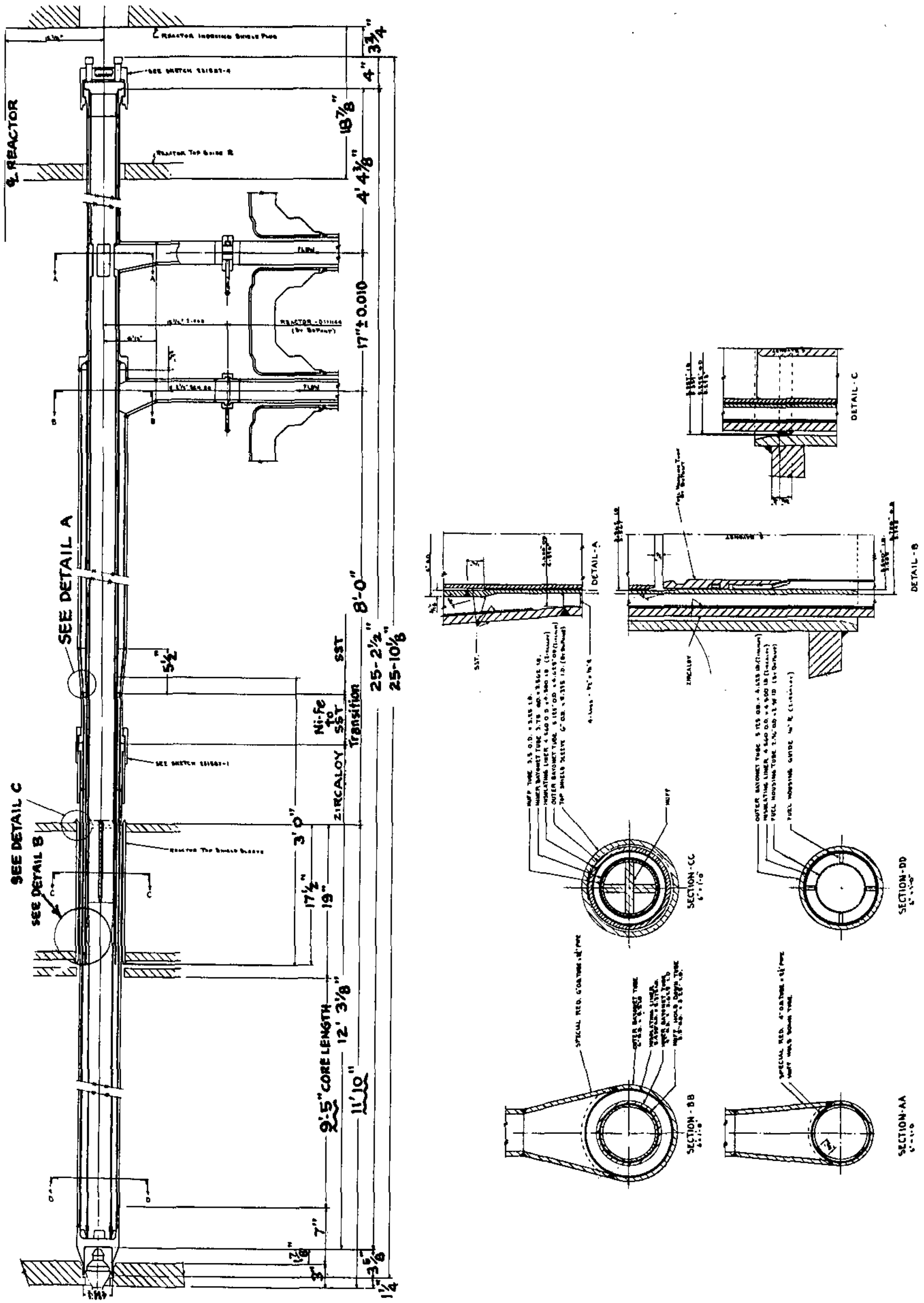


FIG. 1-4 PROPOSED BAYONET ARRANGEMENT

Notes:

(4) Mat'l. - Stainless steel construction except where noted to be Zircaloy. Zircaloy parts to be Zircaloy-2 on EP 185-1 and Zircaloy-4 on EP 185-2.

Design Press. 700 psi internal, 200 psi external } at 600°F

Approx. Finished wt. of Zircaloy parts 155#

Tubing Tolerances:

	SST 3.5" & 4" OD Tube	Zircaloy 5-1/8" OD Tube	SST 6" OD Tube	Zircaloy & SST Insulating Liner
OD	±.015	±.030	±.030	*
ID	±.015	±.015	±.030	*
Thickness	± 10%	± 10%	± 10%	±15%
Ovality		Within Diameter Tolerances		
Straightness	.010" Bow/ft	.030" Bow/3'-0"	.030" Bow/3'-0"	*
Over-all tube straightness	1/16" Bow/12'-0"	1/8" Bow/12'-0"	1/16" Bow/7'-0	*
Over-all bayonet straightness		3/16" Bow/26'-0		*

* To be dimpled or ribbed to conform to bayonet tube contour providing a gap of .030" ±.015

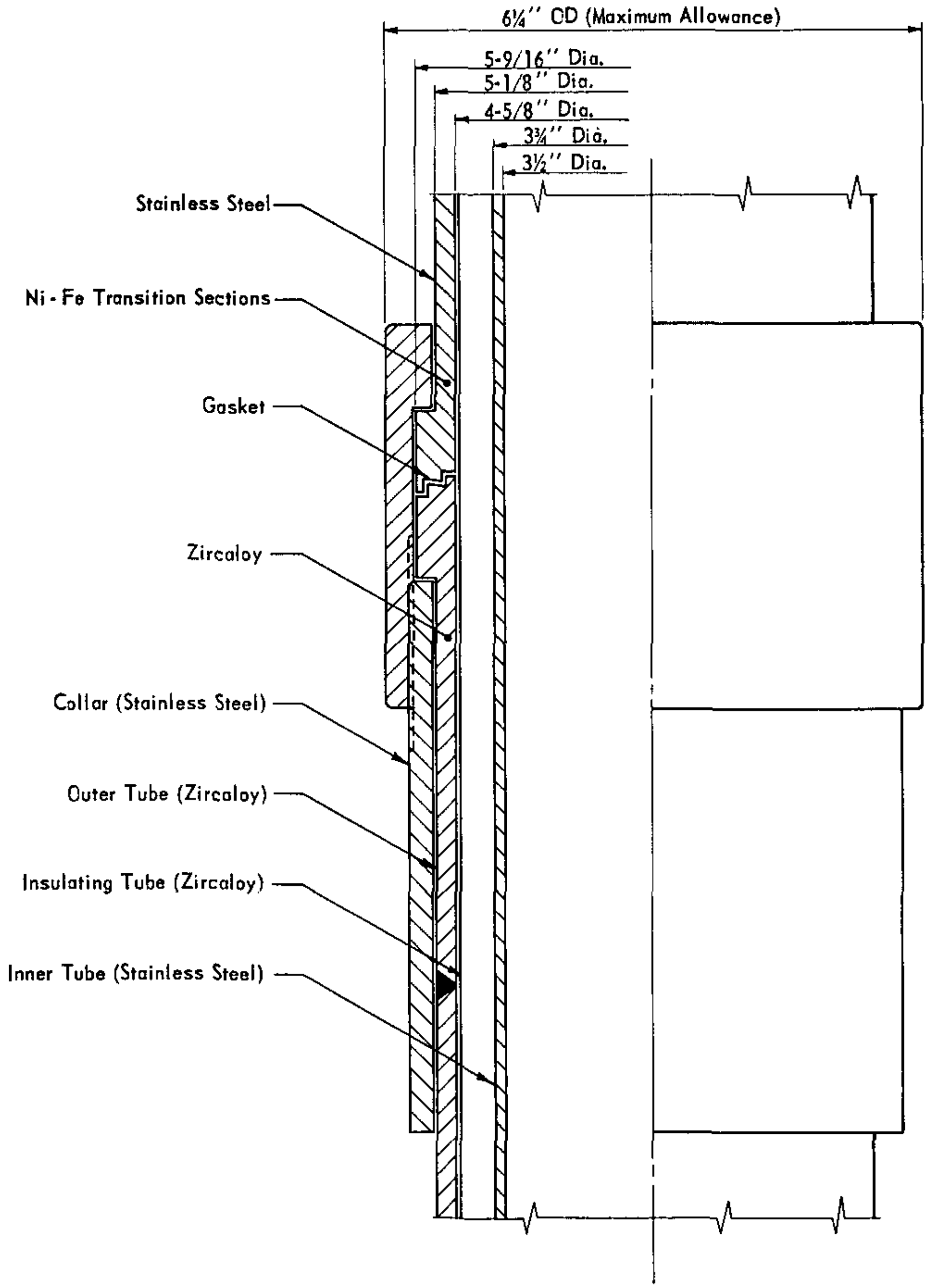


FIG. 1-5 PROPOSED IN-LINE ZIRCALOY-TO-STAINLESS-STEEL CONNECTION

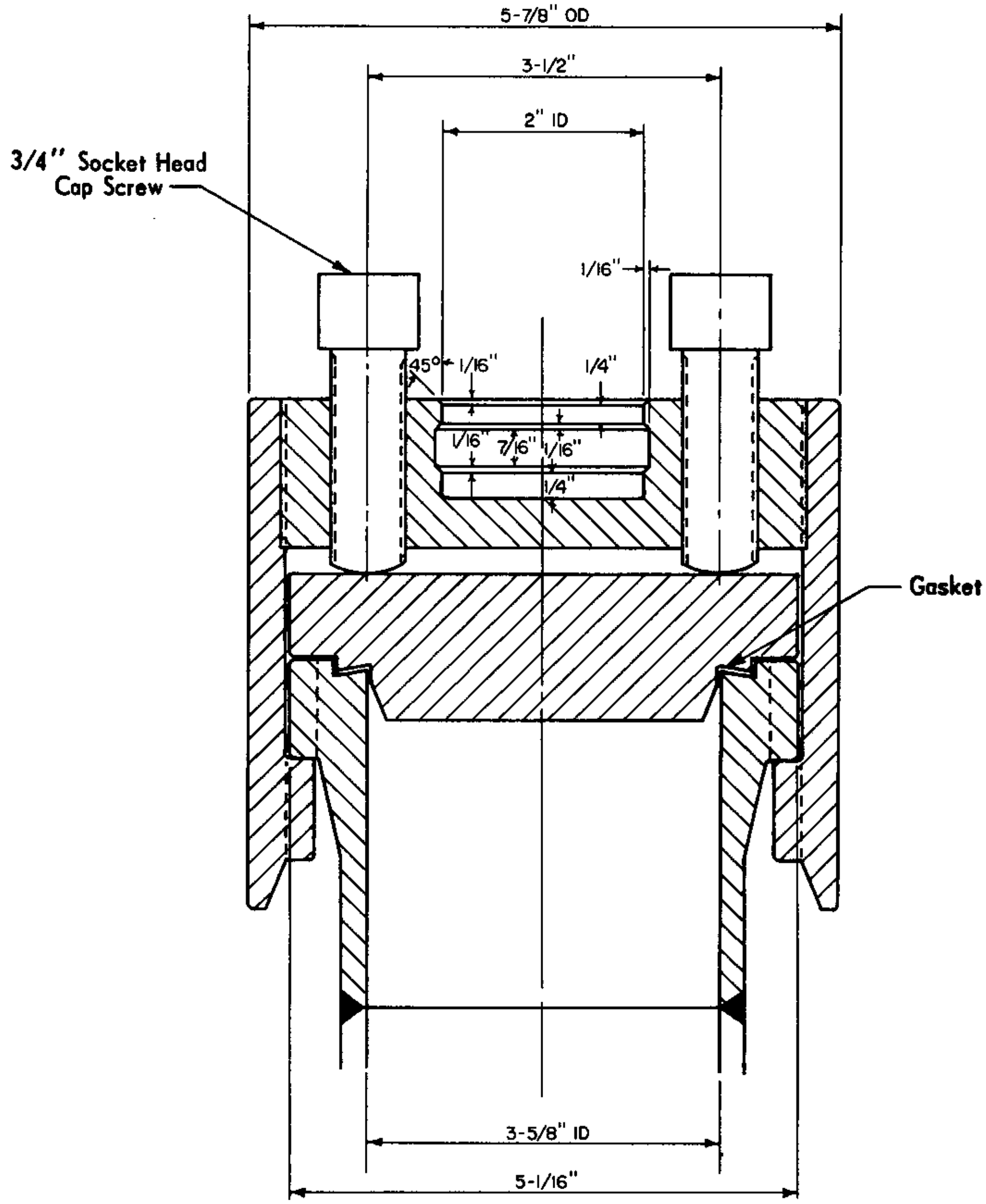
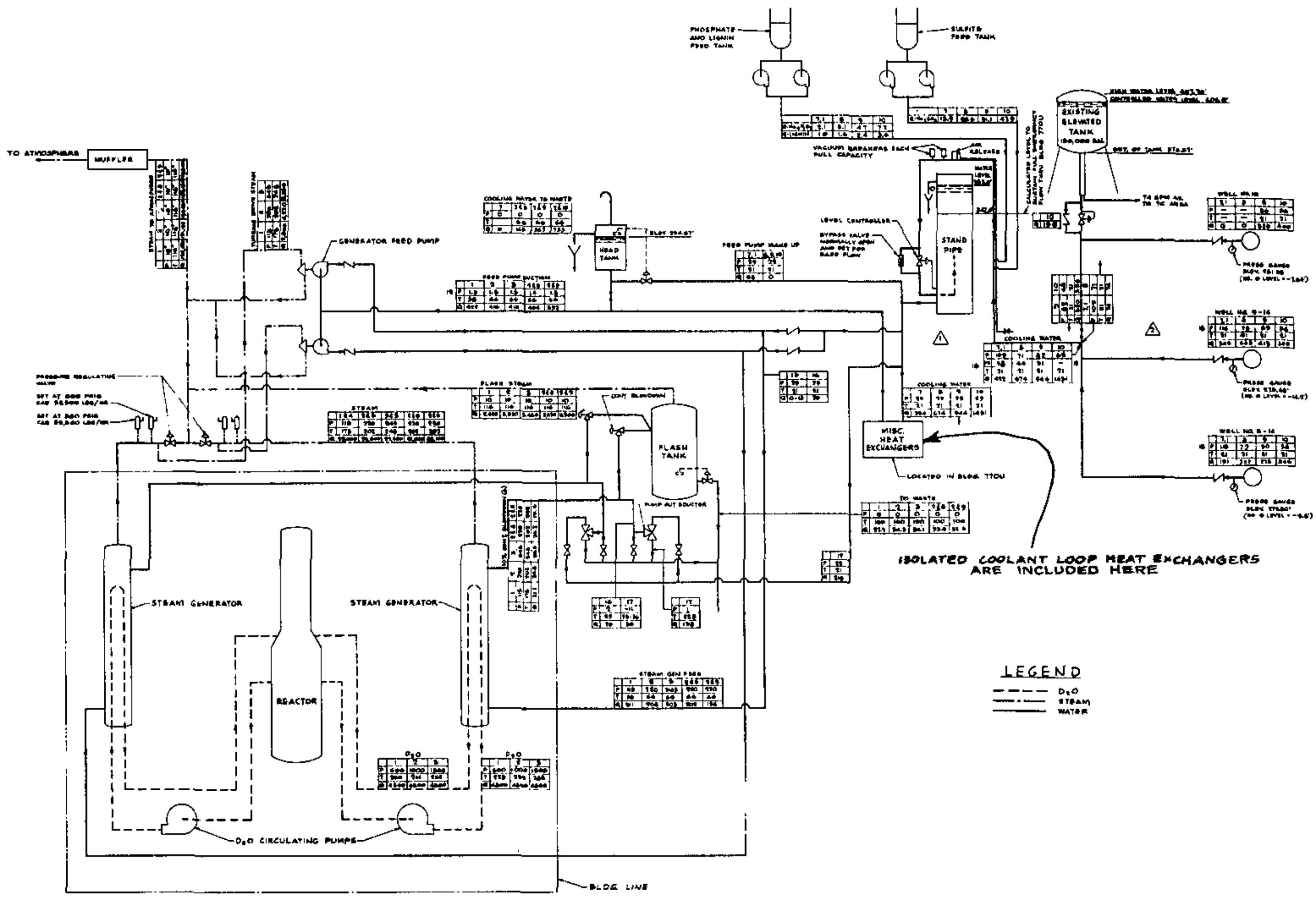


FIG. 1-6 PROPOSED TOP CLOSURE



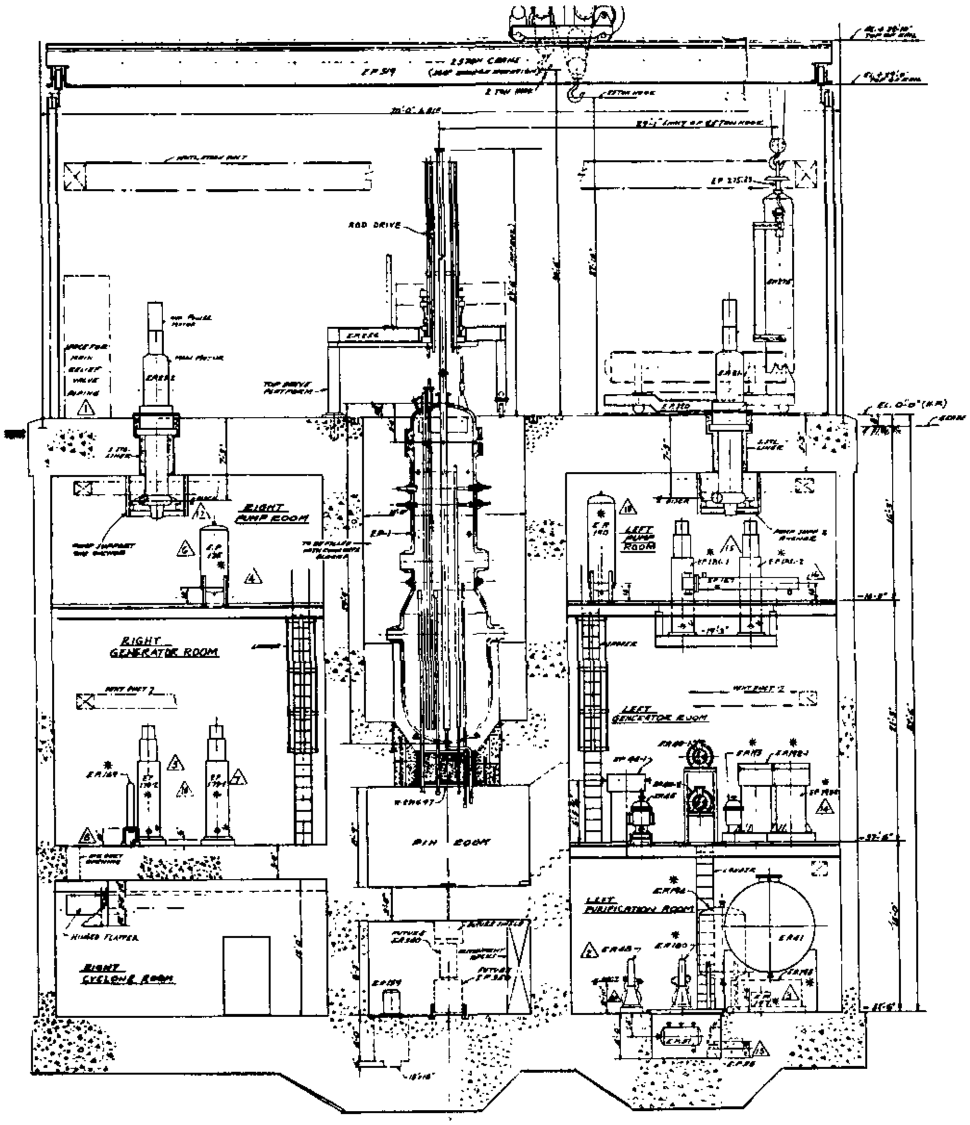
General Notes

P - Pressure, psig, with respect to 0 floor in reactor Bldg., unless noted.
0 floor = 289.67' grade elevation.
T - Temperature, ^{o}C
Q - Flow in lb/hr for steam, $Na_4P_2O_7$, lignin and Na_2SO_3; gpm for water

1 Conditions for minimum design D_2O temp of 225°C leaving reactor
2 Conditions for normal design D_2O temp of 239°C leaving reactor
3 Conditions for maximum design D_2O temp of 285°C leaving reactor
4 Based on 0.0003 fouling factor on light water side of steam generator tubes
5 Based on clean tubes on light water side of steam generator
6 System is designed for maximum of 30% continuous blowdown
7 Operating main reactor only
8 *Operating main reactor and liquid D_2O loop*
9 Operating main reactor, liquid D_2O loop, and boiling D_2O loop
10 Operating main reactor, liquid D_2O loop, boiling D_2O loop, and purge cooler emergency flow
12 Pressure at ℄ generator feed pump suction
13 Pressure at well pump gauge
14 Well pump flows and pressure are based on 1959 test data. Pipe friction is based on C=115 although above tests indicate C=138 which is 0.71 x friction at C=115
15 Flow from standpipe during startup and shutdown when generator pressure is below 25 psig
16 Flow when reactor is being recharged and steam generators are used as water to water heat exchanger
17 Steam generator pumpdown. Used after 16 to prepare for startup
18 PR = Required pressure when control valve bypass is closed and butterfly valve is open 60°
19 Butterfly valve 73° open with bypass valve closed

General - where duplicate circuits exist flow data are given for one circuit only

FIG. 1-7 SECONDARY COOLING SYSTEM



EQUIPMENT IDENTIFICATION

EP 1	Reactor	186-1 & 186-2	Liquid Loop Circulating Pumps
21-1 & 21-2	Main Circulating Pumps	187	Liquid Loop Heat Exchanger
40-1 & 40-2	Main Purge Coolers	190	Liquid Loop Surge Tank
41	Main Storage Tank	192-1 & 192-2	Loop Deionizers
44-1	Main System Deionizer	193	Loop Afterfilter
45	Main System Afterfilter	194	Loop Storage Tank
48	Main System Seal Water Pump	195	Liquid Loop Seal Water Pump
51	Drain Tank	197	Loop Purification Pump
55	Drain Tank Pump	Items not shown	
169	Boiling Loop Purge Cooler	176	Boiling Loop Quencher in Right Generator Room
175	Boiling Loop Surge Tank	177	Boiling Loop Heat Exchanger in Right Generator Room
178-1 & 178-2	Boiling Loop Circulating Pumps	181	Boiling Loop Seal Supply Tank in Dome
180	Boiling Loop Seal Water Pump	191	Liquid Loop Purge Cooler in Left Generator Room
185-1 & 185-2	Bayonets	198	Liquid Loop Seal Supply Tank in Dome

FIG. 1-8 REACTOR BUILDING ELEVATION
(Looking North)

2. LOOP PHYSICS

2.1 REACTIVITY COEFFICIENTS

2.1.1 Test Assembly Design

The effect on pile reactivity of changing the design of all 12 test assemblies was determined experimentally. The correlation between reactivity and the material buckling of an infinite triangular lattice of test assemblies on a 7-inch lattice spacing is shown in Figure 2-1. If the buckling is known for the two loop test assemblies (assuming they are identical), then the reactivity effect is approximately one-sixth the value obtained from Figure 2-1. For example, two test assemblies having a buckling 400 μB lower than the buckling of the standard test assemblies would decrease the reactivity by 0.02/6, or 0.003.

For test assemblies with bucklings outside the range of the experimental data, the reactivity effect may be estimated with the two-group diffusion theory method described in References 1 and 4. Replacement of all 12 "standard" test assemblies with 19-rod clusters of UO_2 rods results in an estimated reduction in k_{eff} of 0.03. Thus, if only the two fuel assemblies in the loops were changed from the standard design to the UO_2 rod clusters, k_{eff} would be reduced by 0.005.

2.1.2 Operation without Test Assemblies

If operation of the pile with no test assemblies in the two loops is desired, the reactivity effect may be estimated from HWCTR critical experiments[4]. Removal of a test lattice having zero buckling from the central region so that only D_2O was present caused an increase in k_{eff} of 0.016. Thus, removal of the test assemblies from the two isolated loops would result in an estimated increase of 0.0027 in k_{eff}.

2.1.3 Void Coefficients

Severe boiling might result in the complete removal of the D_2O from the coolant channels of a test assembly, with a concomitant increase in pile reactivity. Expulsion of the coolant from a single test assembly (tubular U metal) gave a measured increase of only 0.00013 in k_{eff}.[4]

2.2 POWER DISTRIBUTION

With the standard type of test fuel assembly in all 12 positions, the reactor power is limited at the start of a cycle by the burnout safety factor (BOSF) of the driver elements. When a different type of fuel assembly is loaded in an isolated coolant loop, the power generated in that assembly is limited so that the BOSF in both the driver and the test assembly is at least 1.8.

To facilitate the calculation of power generation in test assemblies of different types, the ratio of the average flux in a test assembly to the average flux in the driver assemblies has been calculated for several test assembly designs. The flux ratios were obtained from the P_3 approximation to transport theory; the extreme values in this ratio ranged from 1.6 to 2.1.

2.3 TRANSIENTS FOLLOWING A REACTOR SCRAM

2.3.1 Method of Calculation

Models of the liquid-D_2O-cooled loop and the boiling-D_2O-cooled loop that were assumed for the calculations are shown in Figures 2-2 and 2-3. A heat balance equation is written for each component of the loop, and the system of equations is solved using numerical integration for a perturbation in flow from some initial, steady-state condition. For the liquid loop the method is similar to that described in Appendix B, Reference 1; for the boiling loop, the derivation of the equations is given in detail in DP-485.[5]

2.3.2 Liquid Loop

It was assumed that prior to the scram the reactor was operating at a power of 50 MW, that 87.35% of the D_2O coolant flow through the loop bypassed the heat exchanger, that the loop pressure was 485 psi above the reactor pressure and that the steam valves on the reactor boilers were set at 22.2 $in.^2$. Transient pressures and temperatures in the reactor and in the loop were calculated for 35 seconds following a scram. An example of temperature and power variations is shown in Figure 2-4. The following quantities were varied in the calculation: the initial He-D_2O gas volume in the loop surge tank, the fraction of the D_2O coolant that bypassed the loop heat exchanger, and the steam valve setting on the HWCTR steam generators.

In order to avoid rupturing the protective seals between the loop and the reactor gas systems, it is desirable that the pressure differential between the two systems remain as nearly constant as possible after a reactor scram. The transient calculations indicate that the simplest method of achieving this nearly constant pressure differential is to specify a gas volume of 55 gallons in the loop surge tank and to leave the steam valves in the operating positions.

The loop pressure, the reactor pressure, and the pressure differential are shown in Figure 2-4 as functions of time following a scram for the optimum conditions. Also shown are the two cases that produced the extreme changes in pressure differential compared to the optimum case.

An alternative mode of operation that would control the differential pressure transient would be to close the steam valve from its initial setting of 22.2 to 2.0 $in.^2$. The reactor then cools more slowly and the differential pressure between the reactor and the loop again changes only slightly.

In conclusion, the calculations indicate that if the gas volume in the loop surge tank is 55 gallons, then no damage to the liquid-D_2O-cooled loop will result from a reactor scram.

2.3.3 Boiling Loop

The changes in various temperatures in the boiling loop following a reactor scram are shown in Figure 2-5. In this calculation, it was assumed that prior to the scram the reactor was operating at a power level of 30 MW (steam valve setting of 13.2 in.^{2}) and that the total D_2O flow through the loop was 154 gpm. To achieve the nominal operating conditions for the test assembly (258°C inlet temperature and 16% exit steam quality), 60% of the D_2O passed through the test assembly and 40% passed through the heat exchanger. Following the scram, no changes were made in the steam valve setting or in the valves that control the flow of D_2O coolant through the test assembly and through the loop heat exchanger. Thirty-five seconds after the scram, the temperature of the D_2O at the bayonet outlet was 240°C; no boiling would occur, since the vapor pressure at this temperature is well below the pressure in the reactor.

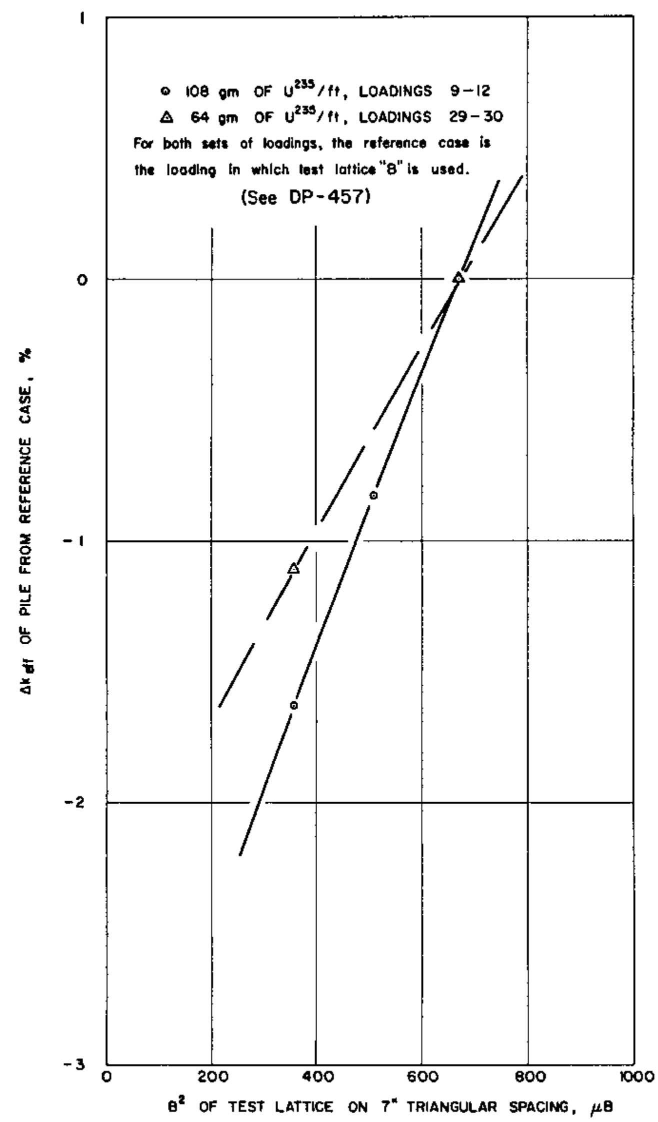


FIG. 2-1 CHANGE IN k_{eff} OF PILE VS. BUCKLING OF TEST LATTICE

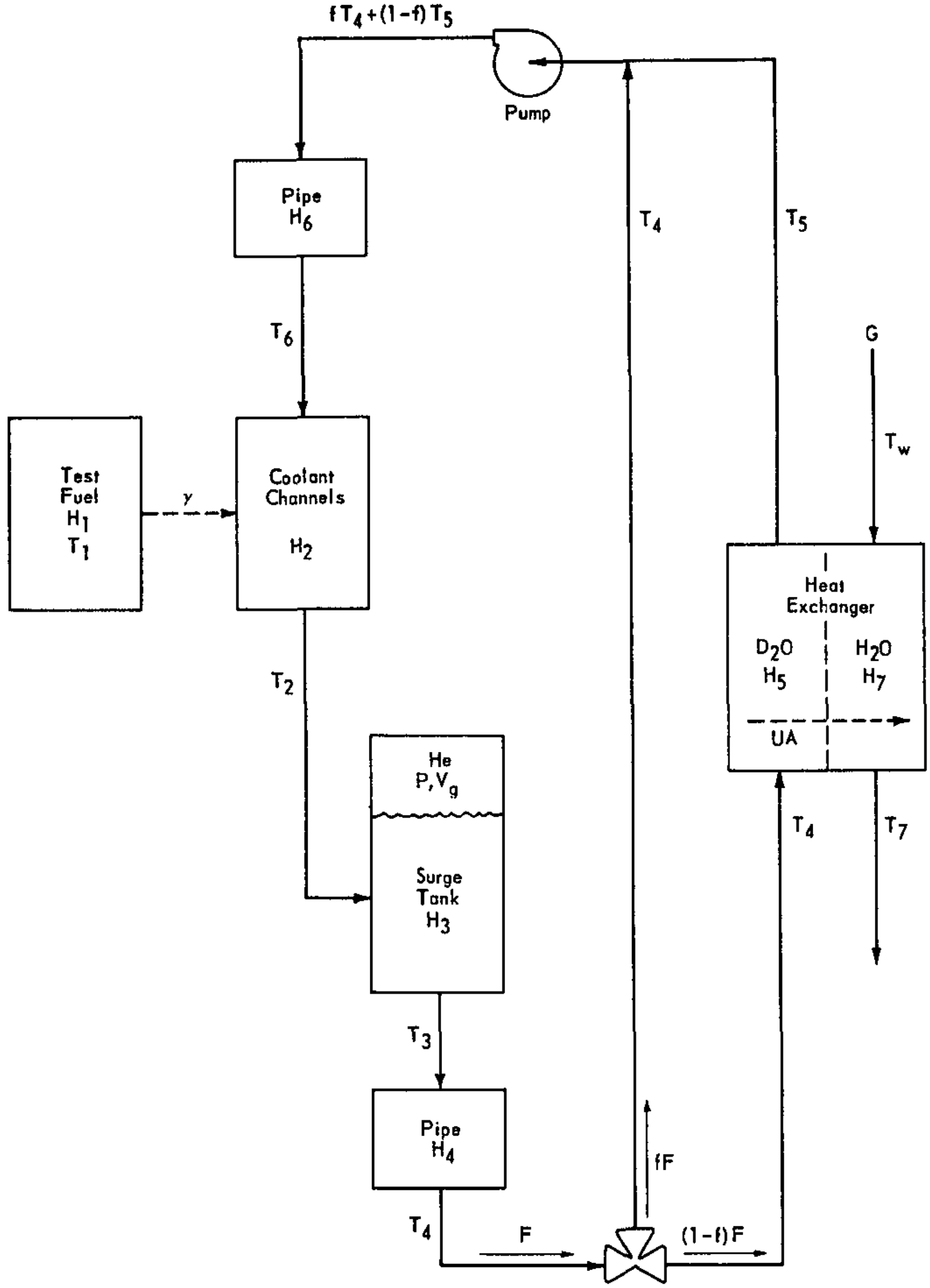


FIG. 2-2 BLOCK DIAGRAM OF LIQUID LOOP

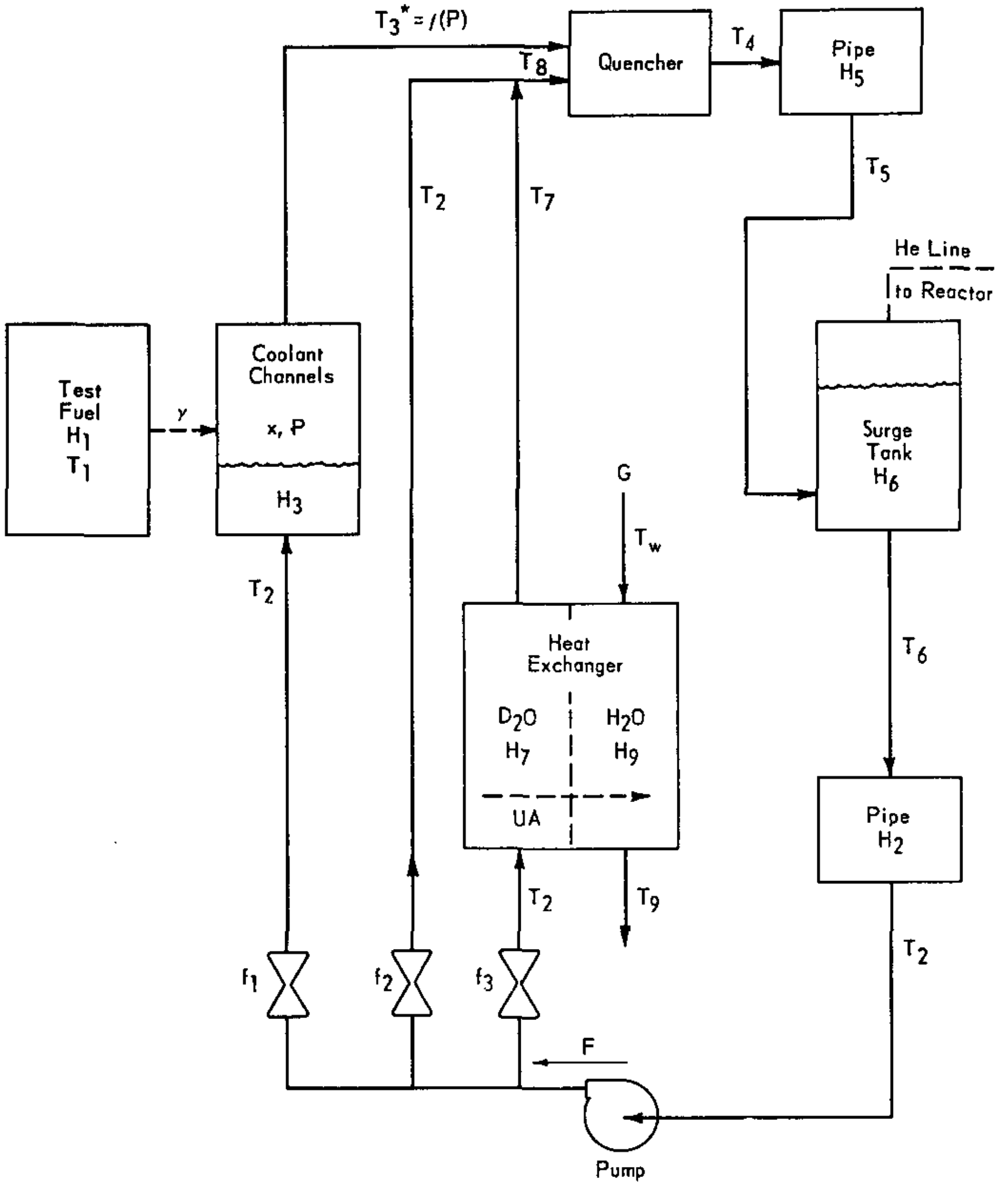


FIG. 2-3 BLOCK DIAGRAM OF BOILING LOOP

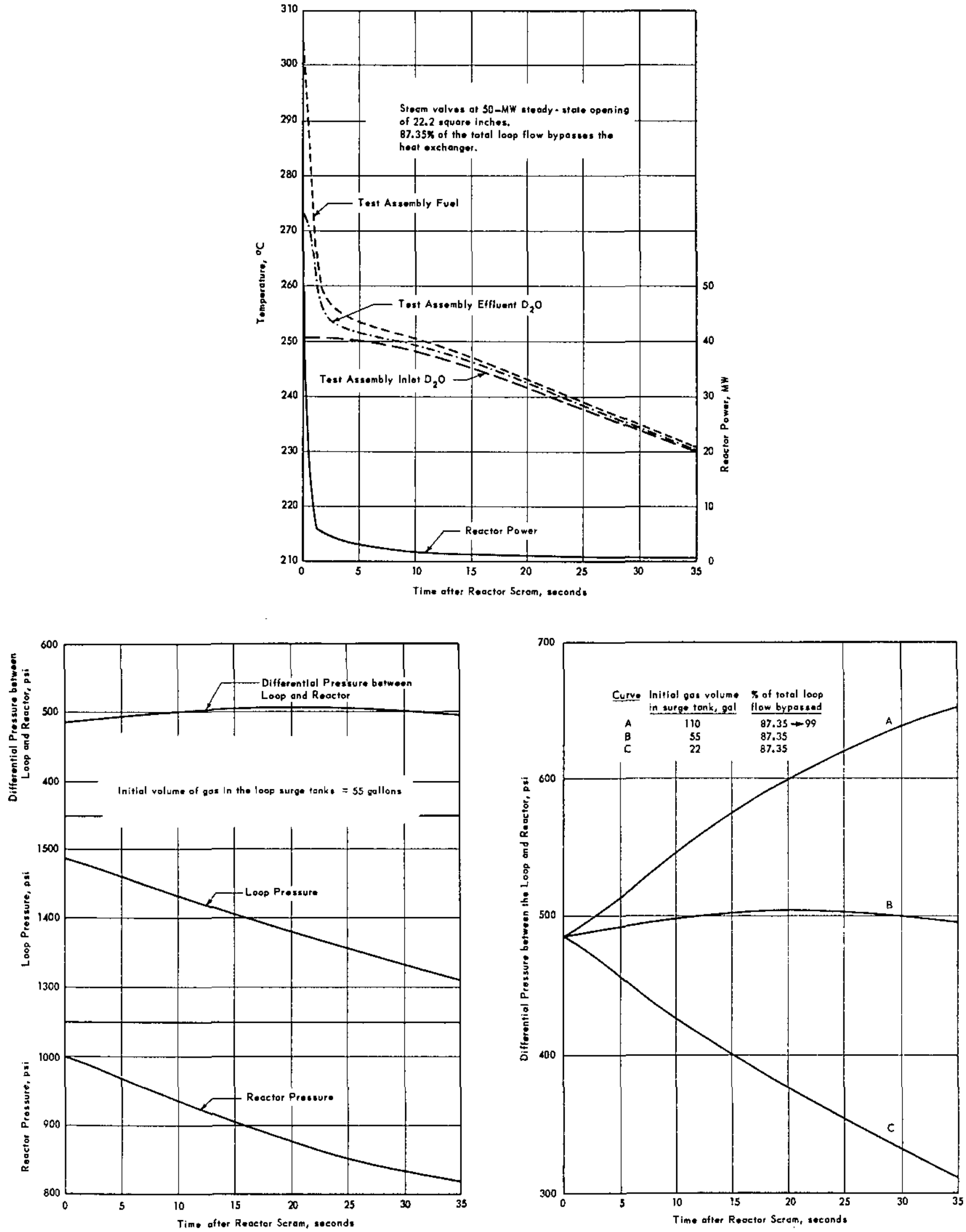


FIG. 2-4 TRANSIENTS IN THE LIQUID LOOP FOLLOWING A REACTOR SCRAM

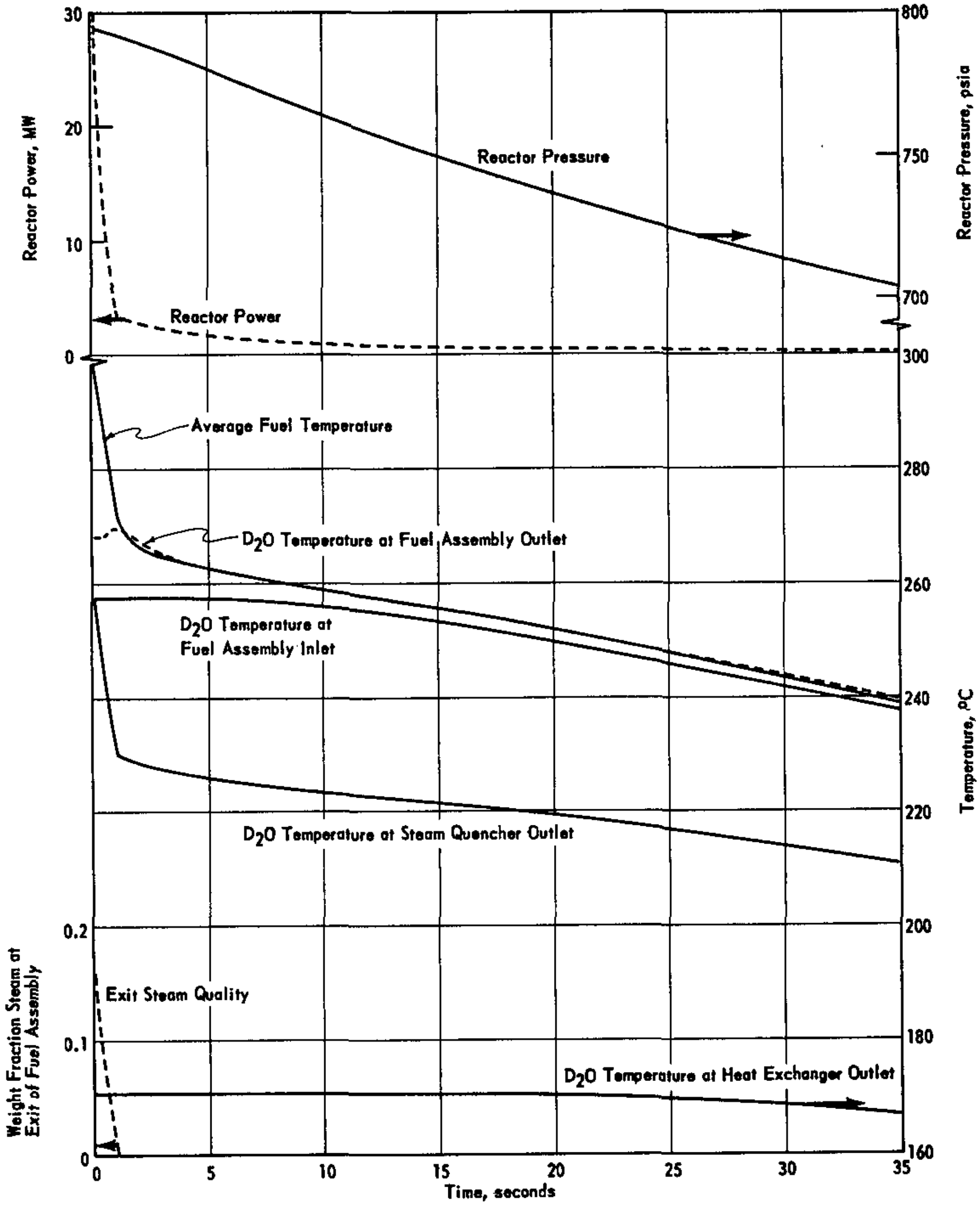


FIG. 2-5 TRANSIENTS IN THE BOILING LOOP FOLLOWING A REACTOR SCRAM

3. LOOP OPERATION

3.1 OPERATING LIMITS AND CONTROLS

3.1.1 Fuel Assembly Power

The reactor power level will be limited by the maximum power that may be generated in a single fuel assembly, either a driver assembly or one of the test assemblies. The maximum power per assembly may be limited by coolant temperature, heat flux, cladding temperature, or fuel temperature. The earlier hazards report[1] discussed these limits in detail. The exception to the limits described previously is that boiling will be permitted in the assembly being irradiated in the isolated boiling loop. Here, the heat flux will be limited by a minimum burnout safety factor (BOSF) of 1.8.

3.1.2 Pressure

The pressure in the boiling loop will equal the pressure in the reactor since the gas spaces in the two systems are connected by lines that contain no valves. The difference in the pressure between the isolated liquid loop and the reactor is limited to the range of -200 to +700 psi by rupture discs in the lines connecting the gas spaces of the two systems. The nominal operating pressure difference will be +500 psi and will be controlled by a signal from a differential pressure controller that measures pressures in the lines connecting the two systems. Gases that escape from the systems through evolution from purge streams and through other losses will be made up by additions to the liquid loop and thence to the reactor through the differential pressure control valve.

3.1.3 Coolant Temperature

The maximum temperature of the coolant in the boiling loop will be the saturation temperature of the D_2O at the reactor operating pressure. The temperature of the coolant in the liquid loop will be controlled at a value less than the saturation temperature at the operating pressure of the reactor. Thus, if the rupture discs between the isolated liquid loop and the reactor should break, the D_2O in the liquid loop will not boil.

3.2 STEADY-STATE OPERATING CONDITIONS

3.2.1 Liquid Loop

Figure 3-1 shows the effect of changing the fraction of D_2O flow that bypasses the loop heat exchanger on the test assembly inlet and outlet temperatures and on the heat exchanger outlet temperature. When more than 80% of the D_2O flow bypasses the heat exchanger, these temperatures are very sensitive to changes in the amount bypassed. The calculations indicate that if 87.35% of the flow bypasses the heat exchangers the nominal values of the test assembly inlet and outlet temperatures, 250 and 274°C, respectively, would be attained.

3.2.2 Boiling Loop

Analysis of the boiling-D_2O-cooled loop for several choices of operating conditions is illustrated in Figure 3-2. For the results in this figure, the following conditions were assumed: (1) the surge tank pressure was 780 psig; (2) no D_2O was bypassed around the heat exchanger; and (3) H_2O flow and temperature at the heat exchanger inlet were constant. The D_2O flow through the test assembly was taken to be 25, 50, 100, and 150 gpm. The steam quality in the test assembly effluent D_2O is shown as a function of the D_2O flow through the loop heat exchanger for several values of the power generation in the test assembly. For example, with 100 gpm through the test assembly, the maximum steam quality is about 23% and the minimum power generation in the test assembly to produce any steam in the coolant channels is 0.80 MW. The maximum power generation is limited to 1.83 MW in all cases to ensure that all the steam is condensed in the steam quencher.

3.3 OPERATING PROCEDURES

3.3.1 Startup Procedures

The startup procedures described in the previous report[1] will be followed whenever either or both isolated loops are to be operated in conjunction with the reactor. At the time that all conditions are established to permit nuclear startup, flow of coolant in excess of nominal will exist in the isolated loops and the block valve in the gas line between the reactor and the liquid loop will be open so that all pressures in the entire system will be equal.

When control rods and steam valves have been adjusted to bring the reactor up to operating temperature and power, both loops will be operating at the pressure of the reactor and at coolant temperatures below the final operating levels. The flow of coolant in the boiling loop at this time will be sufficient to maintain all the coolant in this loop below the boiling point.

The boiling loop will be brought to the boiling condition by adjustment of the flows of D_2O to the fuel assembly and to the heat exchanger that supplies coolant to the steam quencher. These adjustments will have negligible effects on fuel assembly power and the other operating parameters of the reactors. The alternate adjustment of the two flows of coolant will result in the desired quality of the effluent coolant from the fuel assembly and the subcooling across the steam quencher. When these parameters are established at their desired values, they are thereafter controlled by either manual or automatic control. The flow to the fuel assembly is maintained constant by the signal from a flowmeter controlling the throttling valve, and the subcooling is maintained constant by the signal from the ΔT measurement across the quencher controlling the throttling valve to the heat exchanger. The ranges of interest of these two parameters are steam qualities from 0 to 30% and subcoolings greater than about $5^{\circ}C$. However, the steam quality is limited only by the allowable heat flux of the fuel, which is in turn limited by the BOSF.

The desired operating conditions in the liquid-cooled loop will be established by isolating the pressurizing system of the loop from the pressurizing system of the reactor, raising the pressure in the liquid loop, and adjusting the flow in the bypass around the loop heat exchanger. After the two pressurizing systems are isolated, the differential pressure controller will be set at the desired value, nominally +500 psi. Helium will then be admitted to the loop system to establish the desired differential, which will thereafter be maintained as described above. The temperature of the coolant in the liquid loop will then be raised by increasing the flow of coolant in the bypass around the heat exchanger. Thereafter, the flow through the bypass will be automatically controlled to maintain the coolant to the fuel assembly at a constant temperature.

3.3.2 Shutdown Procedures

During a normal shutdown of the reactor and the isolated loops, the reverse of the above operations will result in smooth reductions in coolant temperatures and pressures. The transients that occur in the loops during a scram are discussed in Section 2.

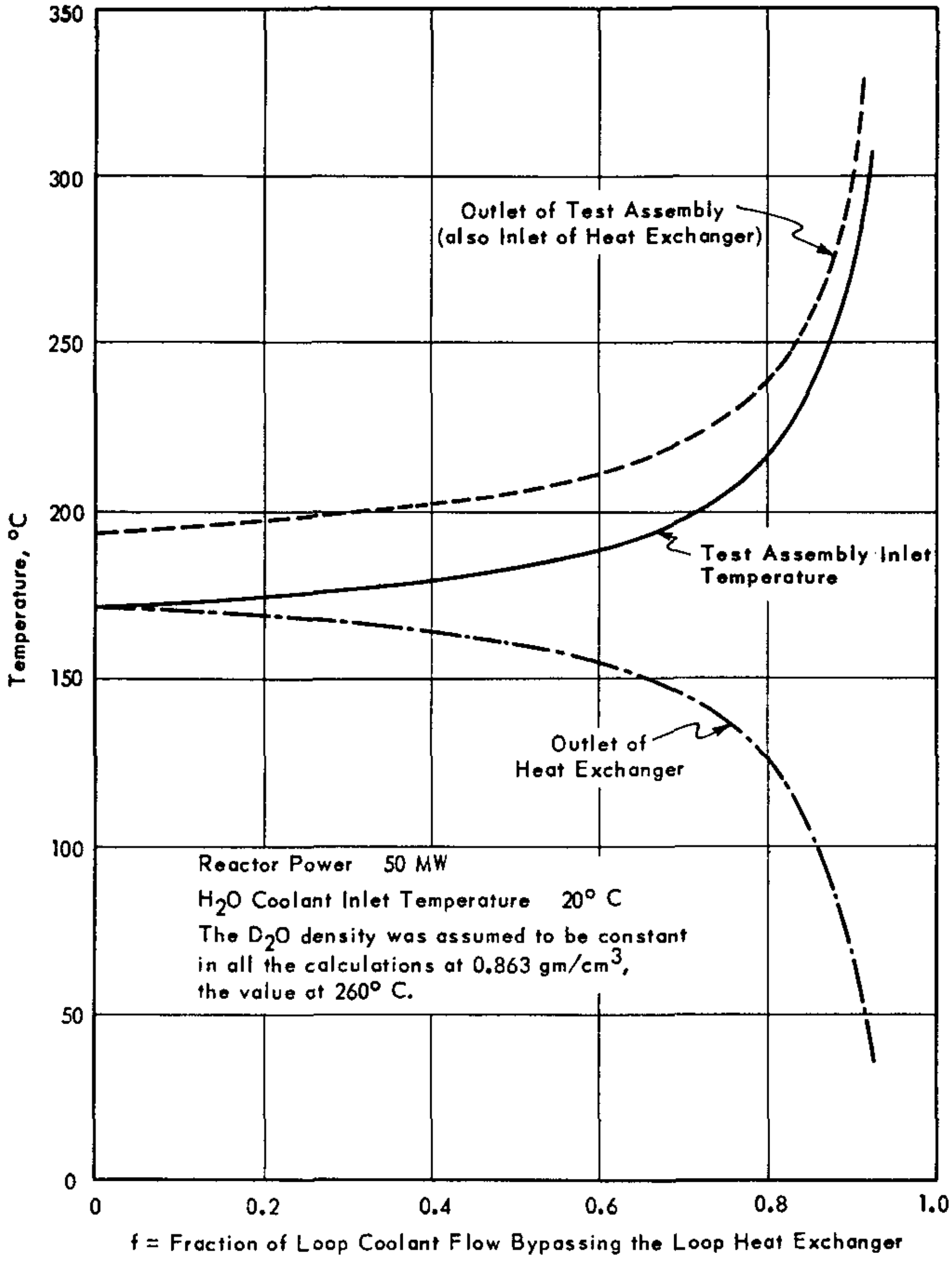


FIG. 3-1 STEADY-STATE TEMPERATURES IN THE LIQUID LOOP

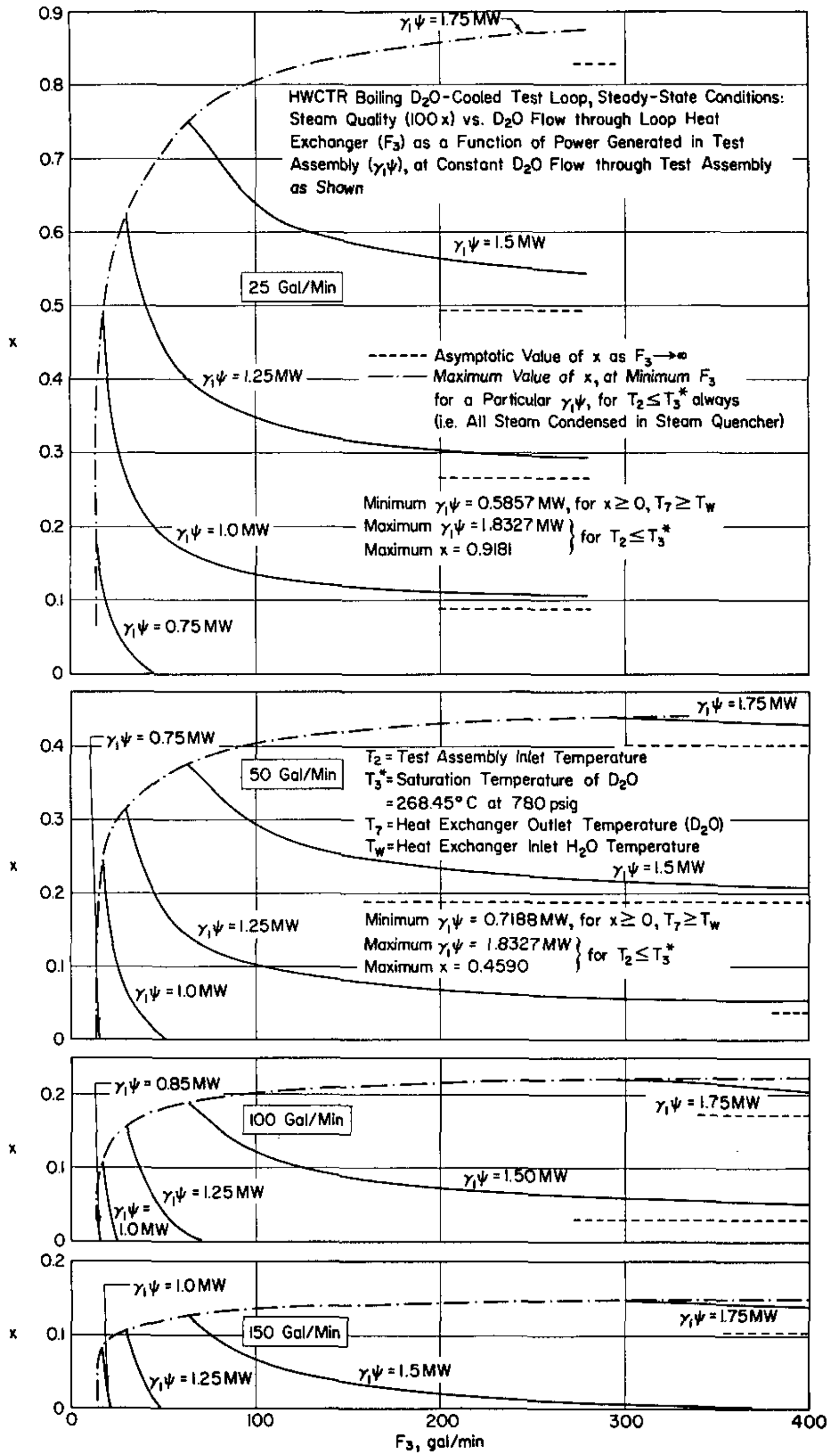


FIG. 3-2 STEADY-STATE STEAM QUALITY IN THE BOILING LOOP

4. SAFEGUARDS ANALYSIS

4.1 GENERAL

Possible equipment failures and operating errors were postulated, and their consequences were determined. The accidents described in sections 4.2 and 4.3 were investigated by the use of methods similar to those described in Appendix B, Reference 1. Figures 2-2 and 2-3 show the models of the isolated coolant loops that were employed.

4.2 ACCIDENTS TO THE LIQUID LOOP

4.2.1 Loss of H_2O Coolant

The initial conditions postulated for calculating the transients following the loss of H_2O coolant were: (1) reactor power level of 50 MW; (2) 87.35% of the D_2O coolant flow bypassing the loop heat exchanger; and (3) loop pressure 485 psi above the reactor pressure. A typical case is shown in Figure 4-1a; it was assumed for this problem that the reactor did not scram after the accident. Under this condition and with an initial surge tank gas volume of 55 gallons, the loop pressure increases to 700 psi above the reactor pressure in 35 seconds, rupturing the seal between the loop and the reactor gas systems. The resultant decrease in pressure in the loop, together with the rise in D_2O temperature in the loop, would cause boiling, and eventually would melt down the test fuel assembly.

The accident just described requires two independent system failures: interruption of the H_2O coolant flow and failure of the reactor scram system. Normally a scram would occur automatically due to the reduction in H_2O coolant flow or to the temperature rise in the loop D_2O coolant. Thus, a more realistic condition is that the reactor scrams following the loss of H_2O coolant. Figure 4-1b shows the result under these conditions: the loop temperatures rapidly approach values about 5°C above the initial inlet D_2O temperature, and the pressure differential between the loop and the reactor increases from the initial value of 485 to 625 psi in 35 seconds. This pressure differential is well below the value required to rupture the seal, and sufficient time is available to adjust the pressures in the loop and reactor as they cool.

Figure 4-1c shows the temperature of the test assembly effluent D_2O as a function of time for three conditions: (1) no D_2O is bypassing the loop heat exchanger and the reactor does not scram after loss of H_2O coolant; (2) 87.35% of the D_2O is bypassed and no scram occurs; and (3) 87.35% of the D_2O is bypassed and the reactor is scrammed.

The results of these calculations indicate that if H_2O coolant is stopped and the reactor is scrammed, either automatically or manually, no damage to the loop or to the reactor will result. If no corrective action is taken and if the scram system fails, then boiling of the D_2O would ensue.

4.2.2 AC Power Failure

For the calculation of these transients, it was assumed that after an AC power failure, the D_2O coolant to the loop decreases linearly to one-third its initial value within 30 seconds and then remains constant. This assumption is based on a slow decrease in pump speed due to the inertia of the pump flywheel, followed by resumption of pump operation on the DC emergency power.

Figure 4-2a illustrates typical loop temperature and pressure transients under the conditions that the reactor continues operation at 50 MW and that 87.35% of the loop D_2O flow continues to bypass the heat exchanger. The consequences of this accident are less serious than those considered in Section 4.2.1; with a 55-gallon gas volume in the surge tank, the pressure differential between the loop and the reactor increases from 485 to 566 psi after 35 seconds so that the protective seal is not ruptured, even if the reactor continues to operate.

Normally, the reactor would be scrammed after AC power failure; Figure 4-2b shows the transient behavior in this case. The pressure differential increases to 535 psi after 35 seconds, compared to 566 psi that was obtained when the scram system failed.

In Figure 4-2c the change in the test assembly effluent temperature with time is plotted for three conditions: (1) no D_2O is bypassing the loop heat exchanger and the reactor does not scram after an AC power failure; (2) 87.35% of the D_2O is bypassed and no scram occurs; and (3) 87.35% of the flow is bypassed and the reactor is scrammed.

4.2.3 Changes in D_2O Bypassing the Loop Heat Exchanger

4.2.3.1 Decreased D_2O Bypassing

The increased cooling caused by increasing the amount of D_2O through the heat exchanger results in the transients that are illustrated in Figure 4-3a. Continued operation of the reactor at 50 MW was assumed. The calculation indicates that this accident results in a slow decrease in the pressure differential between the loop and the reactor, but seal rupture (at a differential of -200 psi) is not threatened.

4.2.3.2 Increased D_2O Bypassing

Typical transients following a sudden increase in D_2O bypassing are shown in Figure 4-3b. It is apparent that if the reactor were not scrammed, the pressure differential would rise to about 700 psi in 30 seconds, if the initial volume of gas in the surge tank is 55 gallons and the bypass flow increases from 87.35% to 99%. However, the temperature rise in the effluent D_2O would initiate a scram signal after about 3 seconds, in which interval the pressure differential would not exceed the rupture value.

Figure 4-3c shows the variation of the temperature of the test assembly effluent D_2O for four possible conditions.

Only if the scram system fails does this type of accident result in breaking of the rupture disc and possible damage to the system. This would require about 30 seconds.

4.2.4 Effect of Reactor Transients

The response of the liquid loop temperatures and pressures to reactor transients was calculated for both step and ramp reactivity additions. Results for step additions of 0.4% and 0.6% k and a ramp addition of 0.03% k per second are shown in Figure 4-4. The following assumptions were made: (1) the initial reactor power was 50 MW; (2) the H_2O coolant temperature at the inlet of the loop heat exchanger was $22^{\circ}C$; and (3) the fraction of D_2O flow that bypassed the loop heat exchanger was 0.90. For these cases, the pressure differential between the loop and the reactor does not approach the value that causes the rupture disc to fail. Therefore, this type of accident does not result in damage to the loop. The effect of reactivity transients on the reactor itself are discussed in Reference 1.

4.3 ACCIDENTS TO THE BOILING LOOP

4.3.1 Loss of H_2O Coolant

A 20% reduction in the flow of H_2O coolant to the loop heat exchanger normally causes a reactor scram. Complete cessation of coolant flow might result from an accidental break in the H_2O line. The system transients following such a break are shown in Figure 4-5a. A reactor power of 30 MW was assumed prior to the accident. After 4 seconds all the H_2O boils out of the heat exchanger, and after 35 seconds the temperature of the D_2O from the heat exchanger rises from an initial value of 171 to $212^{\circ}C$. However, the temperature of the D_2O from the steam quencher changes very little after 10 seconds. Thus, if the scram system functions, no damage to the reactor or loop results.

In the event of a simultaneous loss of H_2O coolant and failure of the scram system, the steam quencher outlet temperature rises in approximately 19 seconds to the saturation temperature of D_2O at 780 psig, and the steam is not completely condensed. The quality of the steam issuing from the test assembly increases, until, eventually, meltdown of the test fuel occurs. The transient behavior under these conditions is illustrated in Figure 4-5b.

4.3.2 AC Power Failure

The assumptions that were made for the calculation of the transients after AC power failure are the same as those given in Section 4.2.2. The results are shown in Figure 4-6a; no damage would be caused by AC power failure if the scram system functioned normally and pump operation continued on the emergency DC power supply.

For transients shown in Figure 4-6b, it was assumed that the scram system failed to operate after AC power failure. The steam quencher outlet temperature rises in approximately 11 seconds to the saturation temperature of D_2O at 780 psig, and the steam is not completely condensed. Eventually meltdown of the test fuel would occur under these conditions.

4.3.3 Change in D_2O Bypassing the Loop Heat Exchanger

The result of bypassing part of the flow through the heat exchanger while maintaining a constant flow through the test assembly is shown in Figure 4-7. It was assumed that 60% of the D_2O flow passed through the test assembly and that the reactor power was 30 MW. For the case considered, a 10% decrease in flow through the heat exchanger in 5 seconds, the outlet temperature from the steam quencher comes within 2°C of the saturation temperature at the loop operating pressure. Thus very little additional bypassing would result in incomplete quenching.

4.3.4 Effect of Reactor Transients

The response of the boiling loop to inadvertent changes in pile reactivity is illustrated in Figure 4-8. The cases that are shown include step increases of 0.1 and 0.2% and a ramp increase of 0.03% k/sec. An initial pile power of 30 MW (corresponding to a steam valve opening of 13.2 in.^{2}) was assumed. The D_2O flow split in the loop was assumed to be 0.6 through the fuel assembly and 0.4 through the heat exchanger.

The effects of reactor transients are more serious in the boiling loop than in the liquid loop. For example, the ramp reactivity addition causes the temperature of the D_2O at the outlet of the steam quencher to increase to the saturation temperature in 5 seconds. After this time, the D_2O steam would not be completely condensed. If the reactor power continued to increase, pump cavitation would likely occur and meltdown of the fuel in the loop would follow.

An accidental step increase of 0.4% k (not shown in Figure 4-8) also results in incomplete quenching of the D_2O steam after only 0.5 second. Thus it appears from the calculations that the boiling loop is more sensitive to reactor transients than is the liquid loop (see Section 4.2.4).

4.4 ACCIDENTS TO EITHER LOOP

4.4.1 D_2O Leaks from High Pressure System

Small leaks that develop during operation of the loops are not hazardous and will be detected promptly. Leaks of less than 2 gallons per minute will be made up by the seal pump and will be detected by a decrease in the purge flow and a decrease in the loop storage tank level. A low level alarm will sound when the volume of D_2O in the storage tank decreases by about 50 gallons.

Leaks of more than 2 gpm will result in decreases in the surge tank level and in the system pressure. A scram will be initiated by a loss of about 12 gallons from the surge tank and a loss of about 100 psi in pressure.

The system can cope with sudden leaks of up to about 40 gpm if the D_2O addition valve is opened before the loop is pumped dry (about 3 minutes). With the main system makeup pump operating, D_2O is supplied to the loop from the reactor purge line until the system has cooled to 100°C and the reactor is depressurized manually (about 30 minutes). The leak will have diminished to less than 5 gpm as a result of the decrease in pressure, and emergency H_2O can be added to the system if necessary. The limits on the size of leak that can be handled are the capacity of the makeup pump (30 gpm) and the volume of D_2O in the main storage tank (normally about 1000 gallons).

For leaks of more than 40 gpm, the reactor level will fall below the purge nozzle and the loop will drain before the moderator has cooled to 100°C. Since emergency cooling cannot be provided immediately, it is probable that a metallic fuel element in the loop will melt and release fission products to the containment building. When the loop drains, the ensuing gas release will depressurize the reactor and some flashing of the moderator will occur.

For sudden leaks of more than about 100 gpm, the loop will be pumped dry while the moderator is still at a high temperature. The resulting rapid depressurization will lead to the maximum credible accident described in Reference 1.

4.4.2 Helium Leaks from High Pressure System

Leaks of less than 8 standard cubic feet per minute from the high pressure gas system will be made up by the He-D_2 recompressor. Leaks will be detected by an increase in flow through the gas meter in the low pressure helium supply line and by abnormal operation of the main helium compressor. Leaks of more than 8 SCFM will be indicated by a loss in system pressure, and the reactor will scram if the pressure drops about 100 psi.

Gas leaks are tolerable if the resulting pressure decrease is slow enough to prevent flashing of the moderator. It is estimated that for an initial leakage rate of about 400 SCFM, pressure will be maintained in the system until the moderator has cooled to 100°C. Leakage of much more than 400 SCFM will cause rapid depressurization and will lead to the maximum credible accident.

4.4.3 Break in Bayonet

Leaks of less than 2 gpm into the reactor from the bayonet side connections or a break in the bayonet will be detected by a decrease in the purge flow and a decrease in the storage tank level. A leak of more than 2 gpm will result in a scram from the decrease in surge tank level, and leaks in the liquid loop bayonet will also cause a low pressure scram. In addition, if the leak is upstream of the fuel assembly it will be detected by a high effluent temperature in the liquid loop or a low bayonet ΔP in the boiling loop. A scram will be initiated by an increase of $5^{o}C$ in the liquid loop effluent temperature or a decrease of 50% in the boiling loop bayonet ΔP.

Since there is no loss of D_2O from the high pressure system, leaks of any size can be handled if the D_2O addition valve can be opened before the loop is pumped dry. For a leak of 100 gpm this would take about a minute. If a large leak occurs and the valve is not opened, it is expected that melting will result if a metallic fuel element is in the loop.

In the event of a sudden large leak upstream of the fuel, burnout of the fuel could occur before the reactor scrams. After the power is down flow will be re-established to the fuel assembly unless the coolant channels are blocked by deformed fuel or there is a clean break in the system. Any radioactivity released in this accident will be contained in the high pressure system.

4.4.4 Pump Shaft Break

In the event that the shaft of one of the circulating pumps breaks between the pump and the flywheel, flow will continue to be provided by the other pump. When the pumps are operating in series, the loop flow will drop rapidly to about 75% of its normal value, and a reactor scram will be initiated by the reduction in flow. No burnout will occur. Scrams will also be caused by a high effluent temperature in the liquid loop and a low bayonet ΔP in the boiling loop.

When the pumps are operating in parallel, backflow through the idle pump is prevented by a check valve, and the flow remains high enough to prevent burnout of the fuel. Should the check valve fail, most of the flow will bypass through the disabled pump and burnout could occur before the reactor scrams. Again, any radioactivity released will be contained in the high pressure system.

4.4.5 Break in Rupture Disc

The accidental rupture of one of the discs separating the gas spaces of the liquid loop and the reactor will not lead to a serious accident. The liquid loop will be operated so that the vapor pressure of the D_2O in the loop is less than the reactor pressure, so that no flashing of the loop coolant will occur when the disc breaks.

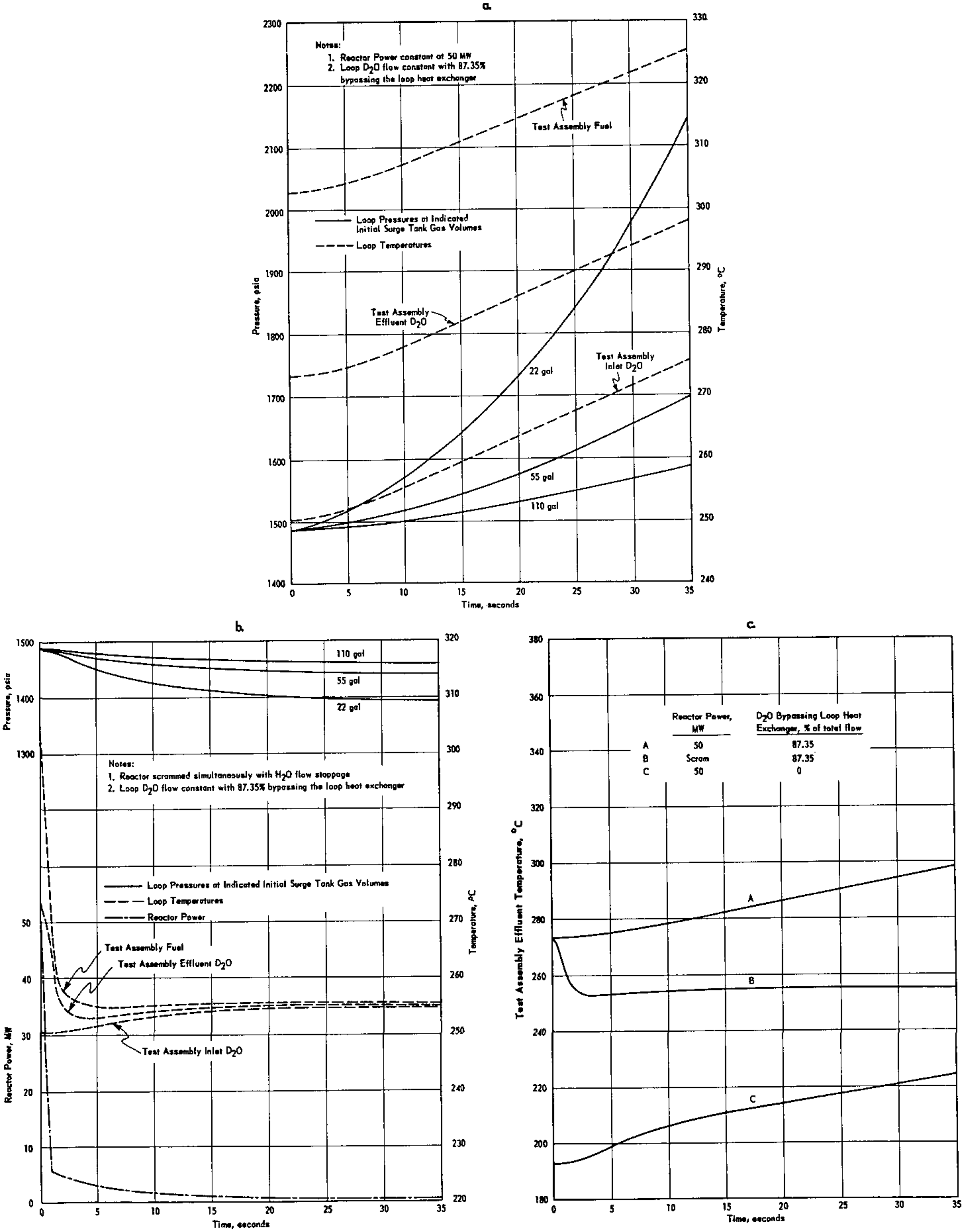


FIG. 4-1 LOSS OF H_2O COOLANT TO LIQUID LOOP HEAT EXCHANGER

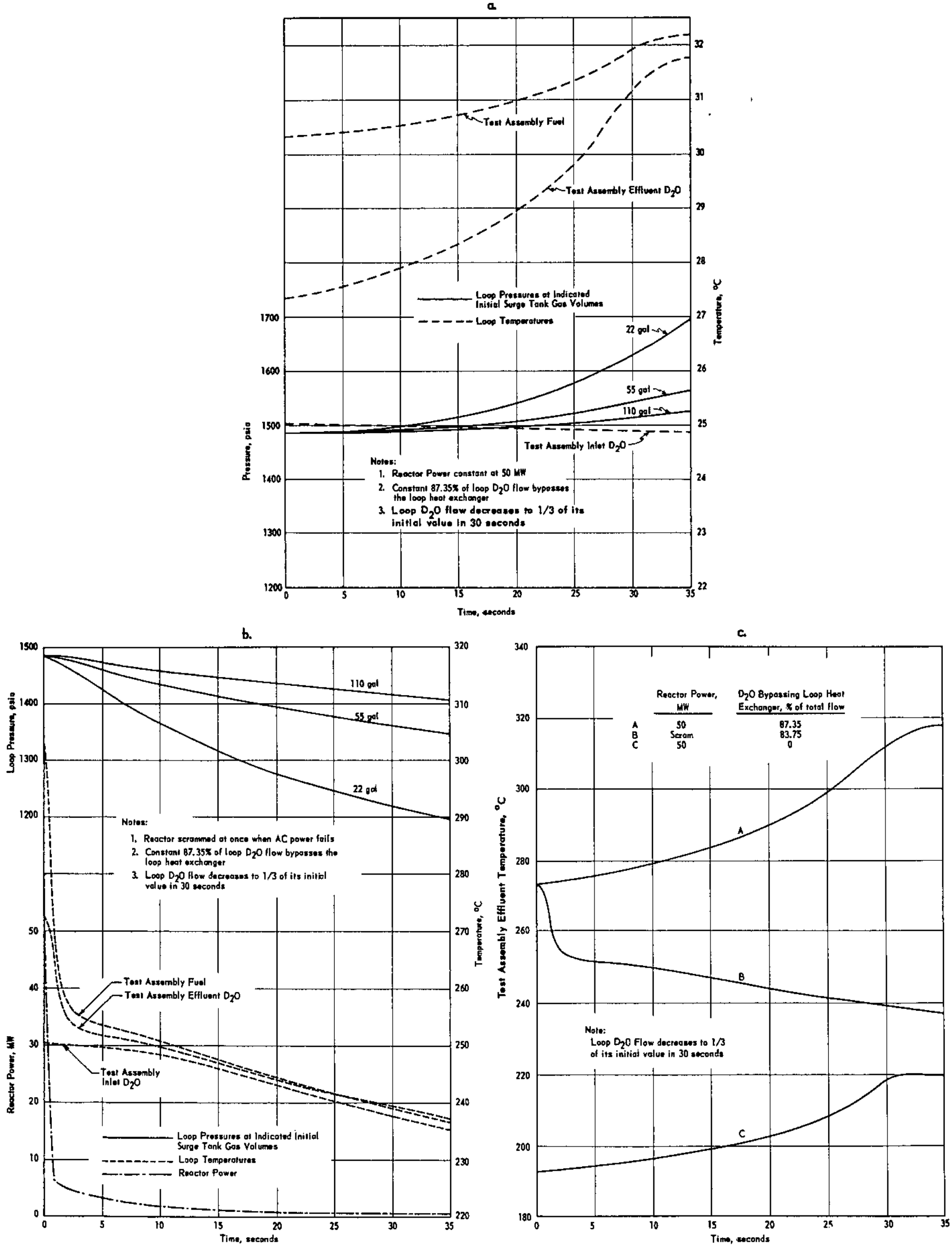


FIG. 4-2 AC POWER FAILURE TO LIQUID LOOP D_2O PUMP

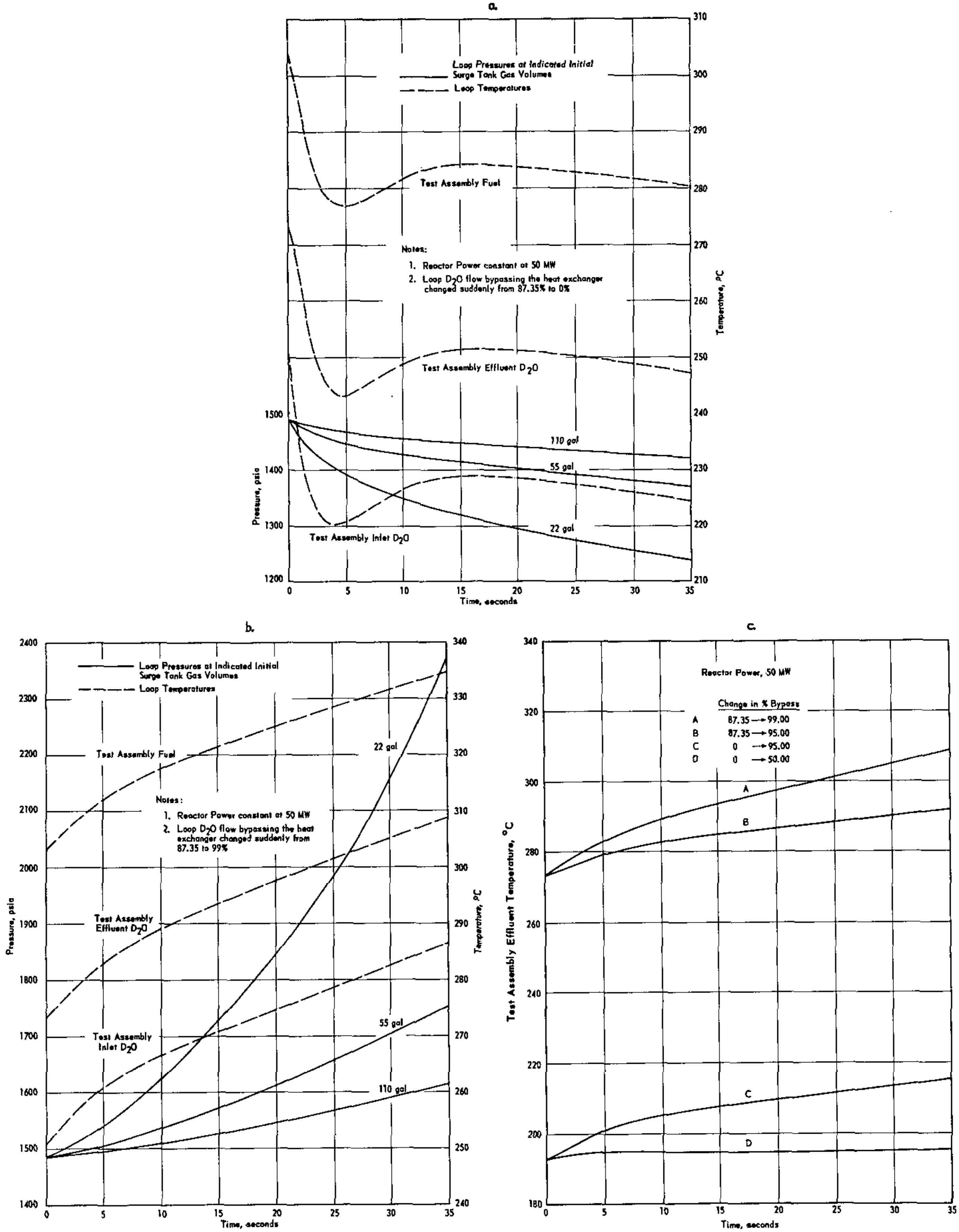
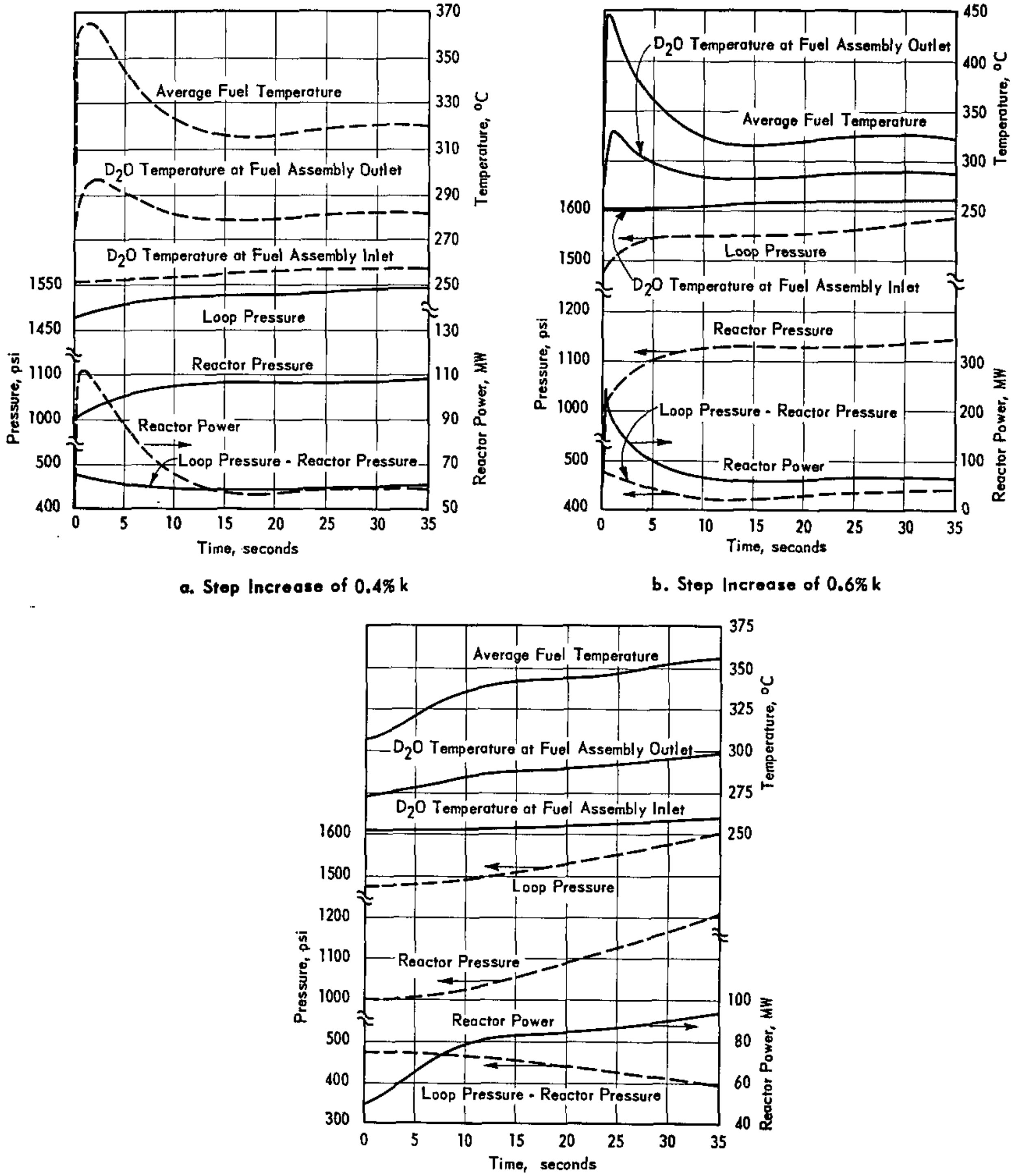


FIG. 4-3 STEP CHANGE IN COOLING IN LIQUID LOOP

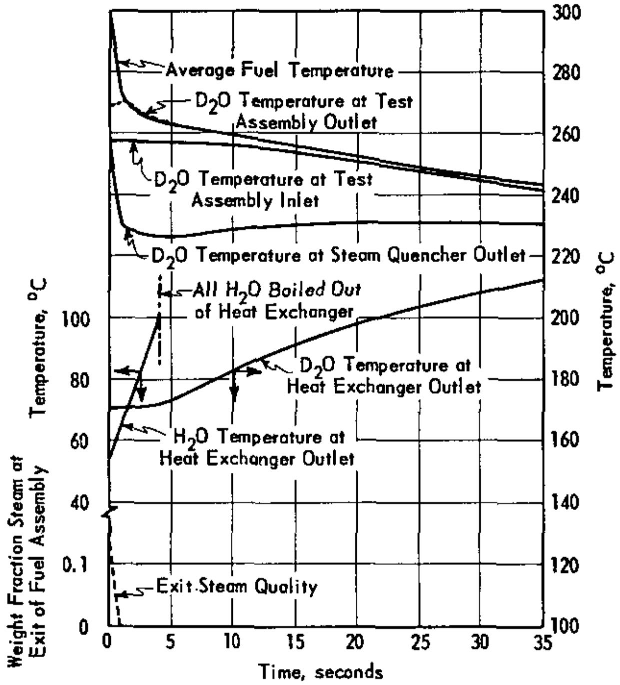


a. Step Increase of 0.4% k

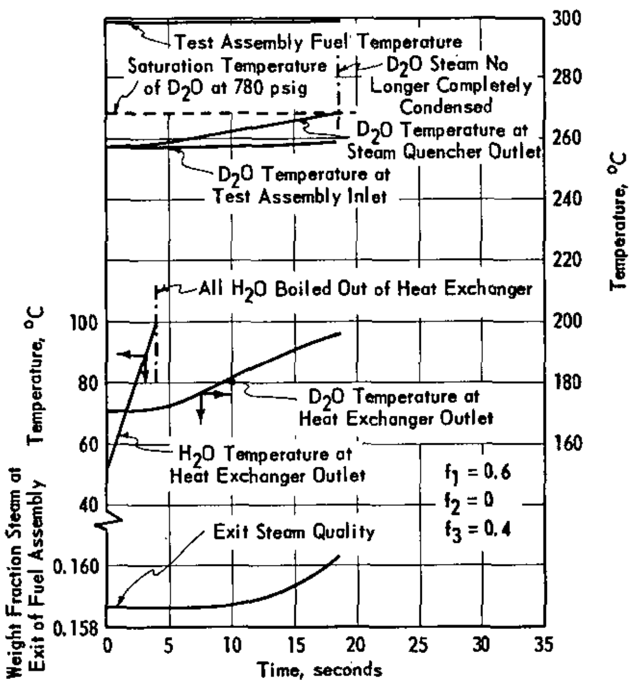
b. Step Increase of 0.6% k

c. Ramp Increase of 0.03% k/sec

FIG. 4-4 RESPONSE OF THE LIQUID LOOP TO REACTOR TRANSIENTS

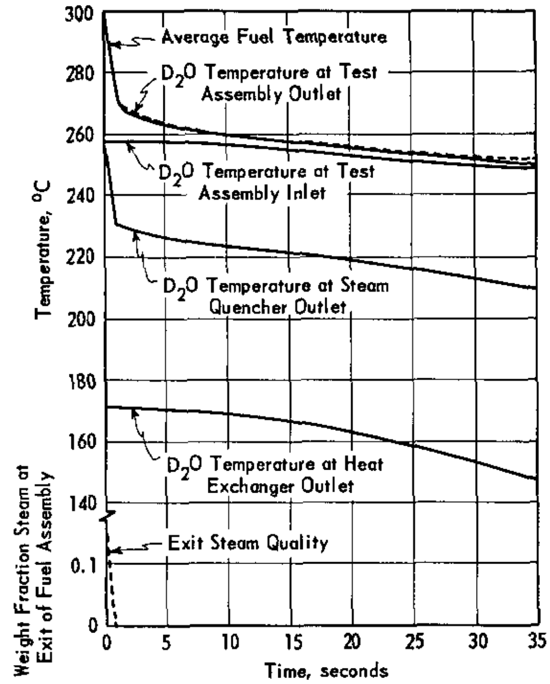


a. With Scram

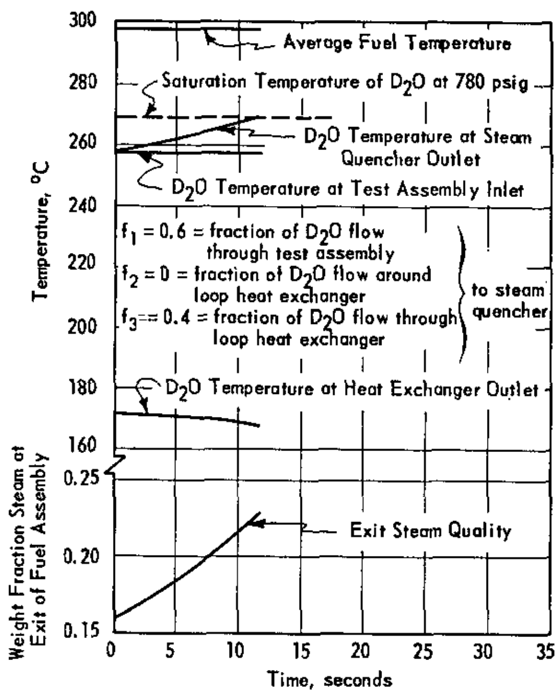


b. Without Scram

FIG. 4-5 LOSS OF H_2O COOLANT TO BOILING LOOP HEAT EXCHANGER



a. With Scram



b. Without Scram

FIG. 4-6 AC POWER FAILURE TO BOILING LOOP D_2O PUMP

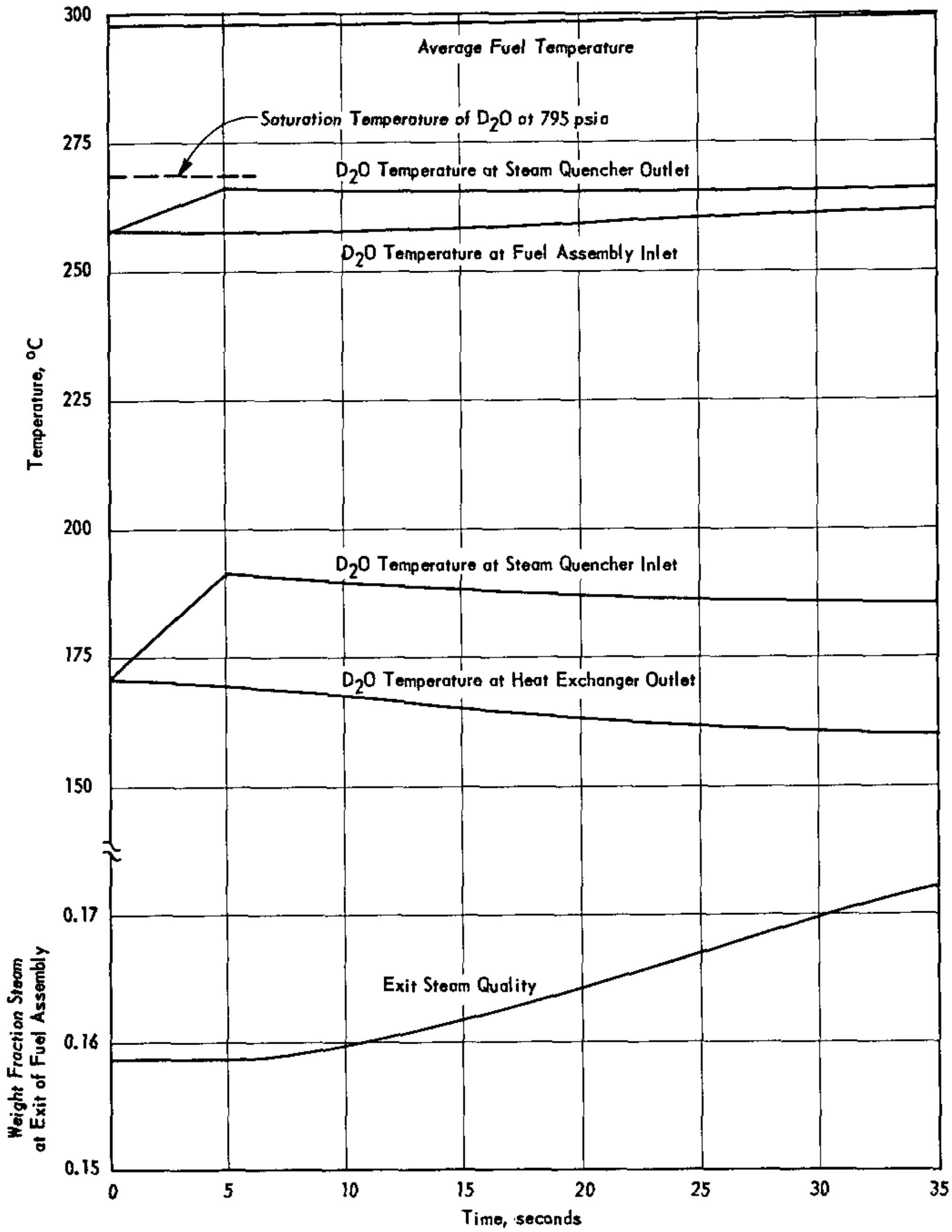
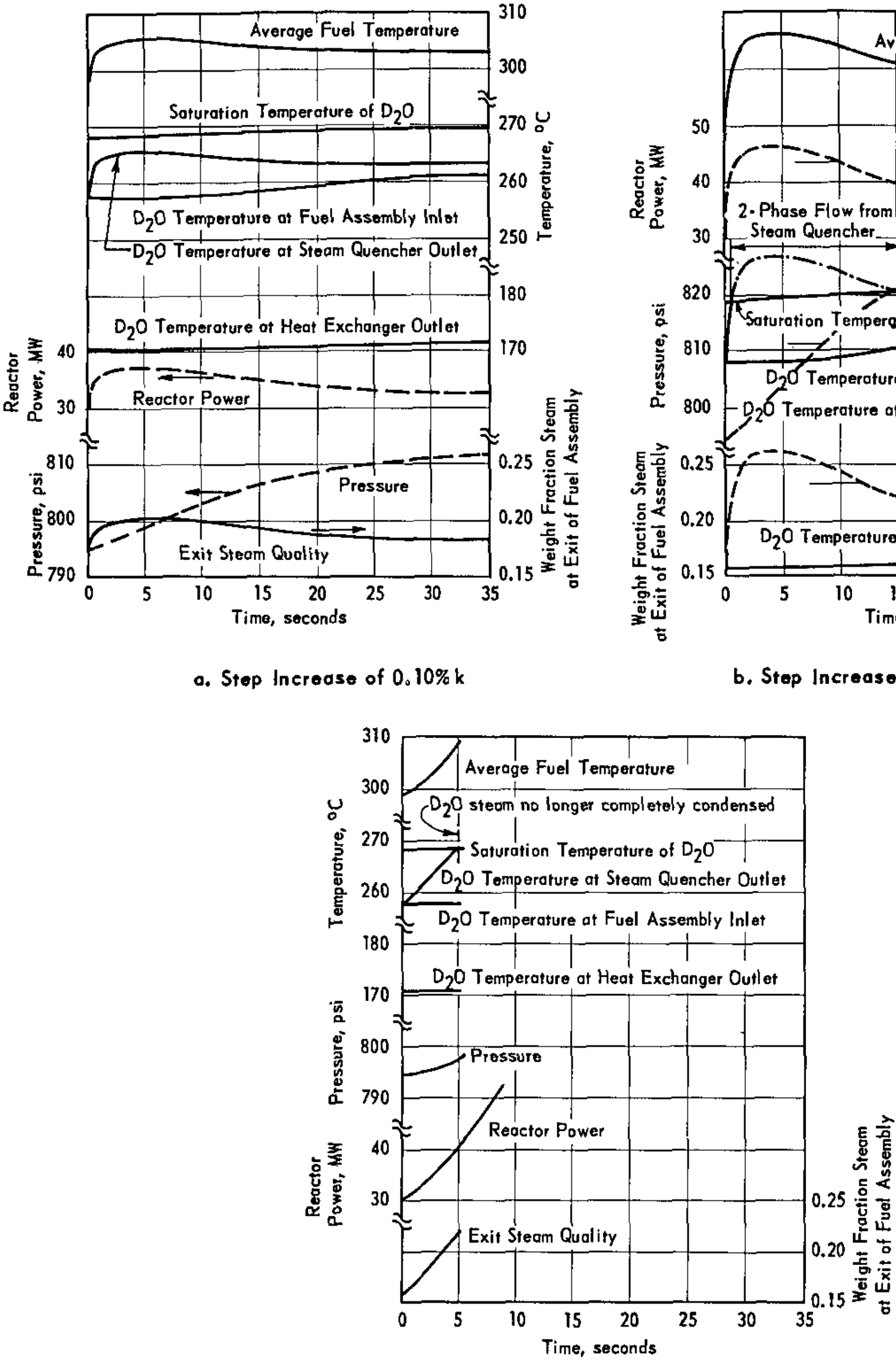


FIG. 4-7 TRANSIENTS IN THE BOILING LOOP FOLLOWING PARTIAL BYPASSING OF THE HEAT EXCHANGER
Fraction of D_2O Flow through the Heat Exchanger Reduced from 0.4 to 0.3 in 5 seconds



a. Step Increase of 0.10% k

b. Step Increase of 0.20% k

c. Ramp Increase of 0.03% k/sec

FIG. 4-8 RESPONSE OF THE BOILING LOOP TO REACTOR TRANSIENTS

5. MAXIMUM CREDIBLE ACCIDENT

The maximum credible accident was defined and discussed in the previous report(1). This accident follows a major break in the high pressure system. The operation of the loops in conjunction with the reactor increases the probability of this accident only to the extent that they increase the size of the high pressure D_2O or gas system. The loops are designed to pressure specifications equal to or greater than those for the main reactor. Each loop is protected against overpressure by the relief devices on the main reactor. The gas space in the boiling D_2O loop is connected directly to the gas space in the main reactor. The gas space of the liquid D_2O loop is separated from the main reactor system by rupture discs. The ratio of the gas space in the liquid loop to that in the main reactor is about 0.1 so that the rupture of a disc at +700 psi differential raises the reactor pressure only slightly. Thus, the addition of the loops does not add to the severity of the maximum credible accident and can, at most, increase only slightly the probability of its occurrence.

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