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"OPERATING EXPERIENCE

of the

EBR-II INTERMEDIATE HEAT EXCHANGER

and the

STEAM GENERATOR SYSTEM"

CONF-830905--12

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INTRODUCTION

Initial Purpose of EBR-II

Experimental Breeder Reactor-II (EBR-II) is an experimental liquid metal fast breeder reactor located at the Idaho National Engineering Laboratory. It consists of an unmoderated, heterogeneous, sodium-cooled reactor with a nominal thermal power output of 62.5 MW; an intermediate closed loop of secondary sodium coolant; and a steam plant that produces 20 MW of electrical power through a coventional turbine generator.

EBR-II was originally designed as an engineering facility to demonstrate the feasibility of fast reactors for central station power plant applications. It was also intended to prove that a breeding ratio greater than unity could be obtained in a power producing reactor. The EBR-II facility was also designed to prove the feasibility of a completely integrated plant in which fuel could be irradiated in the reactor, reprocessed in the Fuels and Examination Facility, and returned to the reactor without being removed from the site. The thermal performance of the reactor and the size of the system components were intended to be amenable to direct extrapolation to central station application. The plant was designed to permit a maximum of experimental flexibility by separation of the plant systems, and yet permit extrapolation to a commercial plant which would not require this same degree of separation.

Evolution of EBR-II Mission

Experience with the reactor during early operation indicated that it would also be useful as a high temperature fast neutron irradiation facility. Since the long-range national emphasis had shifted to much larger plants using ceramic fuel, the purpose of the facility was redirected in 1965 to provide irradiation services for the development of fuels and structural materials for the Liquid Metals Fast Breeder Reactor (LMFBR) program.

Recently, the mission of EBR-II has expanded into a program of Operational-Reliability Testing (ORT) which includes fuel-performance testing and operational safety testing wherein the heat transport system and its components are required to accommodate the consequences of transient power and duty-cycle events imposed upon advanced fuel-element designs.

General Layout of EBR-II

The EBR-II reactor is a pool-type design where all primary system components are located in a large sodium filled tank. The general arrangement of the system is shown in Fig. 1. The primary pumps are in the cold leg piping and take their suction from the pool. The flow is directed through the reactor, where it is heated by nuclear fission. From the reactor, the hot sodium flows to the IHX and then returns to the pool. In the IHX, heat from the radioactive primary system is transferred to the secondary sodium. The secondary sodium system is essentially nonradioactive and is used to transfer heat from the radioactive primary system located inside a containment building to the steam generating equipment located outside the containment building.

The EBR-II steam generator system uses heat from the reactor by way of the primary and secondary liquid-sodium cooling systems. The steam generator system consists of seven (formerly eight) natural circulation evaporators, two once-through superheaters, and a single steam drum. The steam drum is located horizontally above the evaporators and superheaters. The evaporators are arranged in two rows and are connected to the steam drum by individual risers and downcomers. Primary and secondary steam separation takes place within the drum and dry-saturated steam is routed from the top of the drum, through the parallel connected superheaters to a common header to the turbine. Feedwater is supplied to the drum, where it mixes with the saturated steam-water mixture before entering the downcomers. Blowdown is taken from a collection header located within and near the bottom of the drum. Figures 2 and 3 are schematic of the general layout of the system.

PERFORMANCE OF THE EBR-II HEAT TRANSPORT SYSTEM

Operating Parameters (See Table 1)

Secondary sodium flows from the intermediate heat exchanger (IHX) located in the primary-sodium tank to the steam generator at about 299.5 kg/s (5400 gpm) and 466°C (870°F), then passes through the superheaters, where it raises 30.9 kg/s (245,000 lb/hr) of 304°C (580°F) saturated steam to 438°C (820°F) superheated steam. The secondary sodium then passes on to the evaporator inlet headers at about 427°C (800°F), and then passes through the evaporators, heating steam-drum water in an almost isothermal process to 304°C (580°F) wet steam, which returns to the drum. The sodium then returns at about 307°C (585°F) to the IHX.

Feedwater is normally supplied to the steam drum at 33.4 kg/s (265,000 lb/hr) and 288°C (550°F); this accounts for a continuous blowdown flow of 2.5 kg/s (20,000 lb/hr), which is extracted from the steam drum. To inhibit corrosion of the systems, the feedwater is chemically treated by injection of hydrazine and morpholine. Hydrazine (N_2H_4) is used to scavenge dissolved oxygen in the feedwater. Morpholine (C_4H_9NO) is used to maintain feedwater pH in the range of 8.6 to 9.4. In addition, the morpholine acts as a neutralizing agent for dissolved CO_2 or other acid-forming compounds. The results of the feedwater treatment are shown in Table 2.

Operating History

The EBR-II heat transport system continues to operate satisfactorily after 18 years. This represents about 89,000 hours of steaming, which results in a total integrated thermal power production of about 215,000 MWd. In this time, the steam generator has experienced over 580 plant startups and 349 reactor scrams. The plant capacity factor for the past five years has been in excess of 70%, and in fact has averaged almost 60% over the last thirteen years. This excellent record is partly attributable to the trouble-free operation of the steam generator which, aside from an initial construction tube-to-tubesheet weld defect, has had a plant availability of 100%.

THE INTERMEDIATE HEAT EXCHANGER

Configuration

The IHX (see Fig. 4) consists of three basic structures: (1) Well casing, (2) Tube bundle, and (3) Shield plug.

The well casing is a cylindrical Type 304 stainless steel structure, approximately 18.5 ft (5.64 m) long and 6 ft (1.83 m) in diameter. This structure is an extension of the heat-exchanger nozzle of the primary-tank cover. It provides the support structure for the primary-flow inlet diffuser and neutron shielding that surround the heat-exchanger tube bundle. The tube bundle and shield plug form an integral unit that slides into the well casing from the top of the primary tank.

If heat-exchanger maintenance is ever necessary, the tube bundle and shield plug can be removed from the well casing. Removal is accomplished by draining the secondary sodium, cutting the secondary inlet and outlet piping, breaking the upper mounting flange, and lifting the tube bundle and shield plug out of the well casing. Since an inert-gas blanket must be maintained at all times, a caisson or similar mechanism must be used during the removal procedure. After removal, the tank nozzle must be closed with a temporary plug.

The heat exchanger was designed with a low length-to-diameter ratio, which is compatible with the philosophy of a low-pressure-drop heat exchanger. The pressure drop was further reduced by maintaining axial flow to the maximum practical extent. No provisions were made for cross flow, and the support-baffle flow areas were maximized by the use of convoluted-ribbon-type supports, rather than the more conventional drilled-plate-type support. A maximum-pressure-drop criterion of 5 psi (34.4 kPa) for both primary and secondary coolant was easily achieved. Values at full power operation are approximately 2.1 psi (14.5 kPa) and 3.5 psi (24.1 kPa) respectively.

Because of an unusually low length-to-diameter ratio for the heat exchanger, a situation was created in which the primary flow could readily become imbalanced. With imbalanced flow, much of the primary sodium would not penetrate the tube bundle and would therefore bypass the center tubes. This situation could cause a loss of overall efficiency and produce excessive thermal stresses in the tubes and tube-to-tubesheet welds. To achieve balanced primary flow, the heat exchanger was designed to provide an equal static pressure drop, with proper flow for every possible flow path. Since good thermal-convection characteristics were a requirement, cross-flow baffles were considered to be unacceptable for use in the heat exchanger. As an alternative to baffles, the belt diffuser and two orifice plates were used to achieve equal pressure drops for all possible flow paths.

The secondary side of the heat exchanger was also required to have balanced flow and good thermal-convection characteristics. The physical arrangement of the secondary side was also designed to promote natural convection flow. The secondary sodium enters the heat exchanger through an insulated pipe and flows down to the lower ellipsoidal head. Within the head, the flow must make a 180-degree turn before flowing up through the tubes. A semi-torus-shaped diffuser is enclosed within the lower head to turn the flow the required 180 degrees; the diffuser also distributes the coolant to provide a balanced secondary flow.

IHX Performance

Thirty-four thermocouples were installed to provide temperature data at various locations on the primary-sodium side (shell side) of the heat exchanger. Eight of these thermocouples were installed at various locations just below the top orifice plate. Eighteen were at various locations below the bottom orifice plate. Four each were positioned to monitor the primary-sodium inlet and outlet temperatures.

These thermocouples provided little useful information because most of them had failed early in life (prior to raising power above 45 MW). Data apparently were not systematically recorded and/or reported early in life from these thermocouples, so perfomance data are available only from indirect plant data. No instrumentation was provided to directly measure secondary-sodium temperatures, or pressures, or pressure drops within the assembly.

Based upon plant instrumentation external to the IHX, the overall heat-transfer performance of the EBR-II IHX has been in agreement with the design correlation. Considering the unknown channeling of hot sodium on the inside and outside of the unit, which would lower the overall performance, the system condition was adequately described by the design correlations initially used.

Temperature measurements from the installed instrumentation indicated that some of the primary sodium is short-circuiting the tube bundle and traversing the unit essentially uncooled. This occurs in the open areas in the tube bundle next to the center pipe and at the outer periphery next to the shell. This uncooled sodium is not forced to mix with the cooler sodium until the flow streams reach the lower orifice plate. Temperatures have been measured near the inner and outer peripheries of the tube bundle below the lower orifice plate, after some mixing has occurred; these temperatures are of the order of 820°F (438°C). This compares with an average outlet temperature of 700°F (371°C). This measurement was made at full power when the hot primary inlet temperature was 883°F (473°C). This bypass flow lowers the performance of the exchanger and would explain why the measured performance is less than design, as previously discussed.

One other observation, which has caused some minor operational concern but has not resulted in any real problem, is worth mentioning. When primary flow is established through the tube bundle, the pressure in the belt diffuser is equal to the pressure drop through the shell side. This causes primary sodium to rise in the annulus between the shield plug and nozzle casing. With a pressure drop of 2.1 psi (14.48 kPa), the sodium rises as much as 5 ft (1.52 m) up the annulus. With flow changes, this causes a washing action in this annulus as the level moves up and down. As a result, higher than normal temperature and radiation levels have been observed in and above the primary-tank cover in the vicinity of the IHX. Another concern is the thermal stress cycling that occurs as a result of this washing action at the weld joining the 1-in.-thick (25.4-mm) well casing to the 2-in.-thick (50.8-mm) bottom plate of the reactor-tank cover.

IHX Maintenance Events

Except for a minor problem in November 1970, when it was necessary to remove the permanently installed evacuation tube, service has been trouble-free. The investigation of the noise caused by the evacuation tube and the activities involved in the repair are reported in detail in Reference (1). The abstract from this reference adequately describes, for the purpose of this presentation, the problem and subsequent repair.

> "On the night of November 14, 1970, a loud banging noise was heard in the vicinity of the EBR-II Intermediate Heat Exchanger (IHX). Indications were that the noise source was within the IHX inlet pipe. A port for access to the IHX internals was installed on the inlet-pipe elbow. Visual examinations using both a periscope and a remote TV system revealed that of the two supports clips holding a 1-in. (25.4-mm) diameter evacuation tube in place, the top clip was loose and the bottom clip was missing. This condition allowed the evacuation tube to move because of the secondary sodium flow stream and vibrate against the wall of the 12-in. (3.24-mm O.D) diameter inlet pipe. Evidence of wear on both the 12-inch (324-mm O.D.) pipe and the 1-in. (25.4-mm) tube was found.
>
> The upper clip was removed; the evacuation tube was cut at the top and bottom and removed. The lower clip was not found.
>
> The section cut out of the inlet elbow was rewelded in place and the secondary system was restored to operational status. Quiet operation of the IHX verified that the repair was successful."

THE STEAM GENERATORS (EVAPORATORS)

Configuration

The evaporators are shell-and-tube-type heat exchangers. The basic material used in fabrication is 2-1/4 Cr - 1 Mo ferritic steel. This material was selected because of its favorable high temperature properties, established use (ASME coded), availability, cost, fabricability, and resistance to caustic attack. Baffle plates are Type 304 stainless steel. The evaporators are designed to minimize the possibility of interaction between sodium and water/steam by using bonded duplex tubes and double tubesheets (see Figs. 5, 6, and 7). One tubesheet seals the outer of the duplex tubes to form the sodium cavity, and the other seals the inner of the duplex tubes to the water/steam header. Two types of duplex tubing were used in the fabrication of the units. Four evaporators contain mechanically bonded duplex tubes, and four units contain metallurgically bonded tubes. Fabrication of both types of tubes consisted of placing the outer tube over the inner tube, drawing the two tubes together through a die and over a pin. This was followed by expanding the duplex tubes by drawing a pin through the I.D. without restraining the O.D. Metallurgical bonding required a final operation of heating to flow the nickel-nickel phosphorous alloy between the tubes to produce a brazed tube-to-tube-bond. The heating operation annealed out the prestress, which was introduced during the drawing operation, the mechanically bonded tubes were left in the stressed condition of outer tubes in tension and inner tubes in compression.

Maintenance Events

Initial power of EBR-II was attained in July 1964. On February 7, 1965, during a shutdown period with the steam system at ambient temperature, the operating crew reported that liquid water could be observed in the space between the steam and sodium tubesheets at the upper end of evaporator 702. The source of water was traced to a crater crack in one of the tube-to-tube-sheet welds. It was obviously a birth defect that was not detected by the original helium leak test because it was at least partially plugged with slag. The defect was repaired by manual welding. Access was gained simply by removing a section of the steam riser from the evaporator. Up to the present time, no additional leaks have been detected on any of the steam generators.

The evaporator that experienced a leak in 1965 has been subjected to periodic inspections of the steam side. The section of the riser that was removed for leak repair was reinstalled with removable flanges to permit reasonably easy access. Inspections were made in 1969, 1970, 1972, 1973, 1974, 1976, and 1978. Until 1974 all inspections were visual. Tube internals were viewed with a borescope, and photographs were taken for comparison with previous examinations. Surfaces were found to be covered with brown-red magnetic-iron oxide varying in thickness from light to 1.59 mm (1/16-in.). The outer layer was a light porous coating, easily removed by brushing; the inner deposit was a more adherent, dense film. There was no evidence of pitting or metal loss when the deposits were removed. After the evaporator tubes were brushed to remove the light deposits, axial and circumferential fabrication marks could be seen through the borescope. The top 381 mm (15-in.) of the evaporator tube was covered with a thin, fine-grained deposit of iron oxide. The next 2.44-2.74 m (8-9 ft) from the top appeared crystalline as seen through the borescope. The crystalline appearance disappeared gradually down the tube length. Samples of the deposits on the tubesheet have been analyzed and found to be primarily iron, copper, and nickel. Starting in 1974, ultrasonic techniques have been employed to examine the duplex tubes of EV-702. These tubes are metallurgically bonded. The techniques and results of these examinations are reported in Reference (2). The results of these examinations indicate a complete lack of any indications that are detrimental to the integrity of the evaporator.

In 1980, evaporator EV-706 was removed from the system and coverted to a superheater for later replacement of one of the superheaters. Ultrasonic inspection of all the steam tubes revealed no defects. The steam tube-to-tubesheet welds were examined using the liquid penetrant method. The tube I.D.'s were measured and found to be within manufacturing tolerance. Except for the same deposits observed in inspections of evaporator EV-702 and described above, the unit was in excellent condition and considered acceptable for conversion to and installation as a superheater.

THE SUPERHEATERS

Performance

In 1974 the superheater containing mechanically bonded tubes began exhibiting anomalous thermal behavior. The anomalous behavior is typified by a sudden decrease in outlet steam temperature occurring just as full power is approached (Fig. 8). The magnitude of the decrease in outlet temperature varies from startup to startup and has not been correlated with any specific plant parameters. Detailed analyses and examinations are in process to better define the mechanisms contributing to this behavior. The magnitude of the decrease generally increased with time, and the power level at which the drop occurs seems to be decreasing. This trend is shown in Figure 9, which shows the difference in steam outlet temperatures between the two superheaters as a function of calendar time. Measurments have shown that individual tubes are exhibiting sudden drops in steam temperature (Fig. 10) and that the magnitude of the average outlet temperature is dependent on the number of individual tubes so performing. This behavior is explained by an increased thermal resistance of the duplex tubes caused by a reduction in the contact pressure, and in some cases, actual separation. Tube separation occurs when the differential thermal expansion between the inner and outer tube is sufficient to overcome the effect of residual tube prestress combined with steam system pressure within the inner tube. The increase in thermal resistance caused by reduced contact pressure or tube separation results in a larger temperature difference between the two tubes, which leads to further separation. This process continues, along with a decrease in the superheater power level, until stable heat transfer conditions exist.

Maintenance Events

An in-service inspection of the superheater containing mechanically bonded duplex tubes, SU-712, was made in April 1979. This examination was prompted by the observed anomalous thermal behavior reported herein. The main activities of the inspection included visual and ultrasonic inspection of the steam-side internals and the removal and destructive examination of three of the core tubes. Straightness and I.D. measurements of three steam tubes were made.

Ultrasonic inspection of these three tubes indicated that the inner and outer tubes were in intimate contact. Although many interesting observations and measurements were made, the overall assessment of the examination results is that the unit is surprisingly clean and appeared to be "like new" and that there are no unusual conditions or any unusual wear. The details of the techniques used and the results are reported in Reference (3).

Superheater SU712 Disassembly and Examination

In April 1981, the superheater was removed from the EBR-II steam system for destructive examination. The sodium side was sealed and maintained in an inert argon atmosphere until it was cleaned of residual sodium. The steam side was dried and sealed until examination was started in June 1981. Details of the disassembly examinations are found in Reference 4.

The sodium side was cleaned with ethanol prior to disassembly. After cleaning, the exposed surfaces were free of sodium; but restricted interfaces retained unreacted sodium along with some reaction products.

The superheater was disassembled in a sequence that would progressively yield examination results prior to their obliteration by subsequent disassembly. The inlet and outlet steam reducers were first removed (Fig. 11) to allow access to the steam tube inside surfaces for visual examination, I.D. measurements, and straightness measurements. The steam surfaces contained lightly scattered corrosion pitting. The pits were less than 0.25 mm (.010-in.) in depth and appeared to have been formed early in the life of the tubing as evidenced by the oxide coating. The inside diameter measurements were within 0.05 mm (.002-in.) of the nominal fabricated diameter of 27.05 mm (1.065-in.). The results of the straightness measurements indicated that some tubes were bowed beyond the maximum measurable offset of 20.6 mm (0.810-in.), that the peripheral tubes were bowed more than the central tubes, and that the bowing was consistently in the same direction, i.e., the tube bundle was twisted in one direction then returned in the opposite direction (Fig. 12).

Because the superheater was fabricated with a prestress imposed on the shell and tubes (shell in tension and tubes in compression) the disassembly was conducted in a manner that permitted measurement of the residual prestress by two methods. Strain gages were placed on the shell and on the outside surface of one accessible tube. Benchmarks were applied to the shell on either side of a lateral cutting plane. The shell was cut and the springback, and change of shell strain, were measured. The measured springback and the strain gage results closely correlated. The final prestress was 55% of the design prestress.

The tube-to-tubesheet welds were a matter of concern since they were at the location of highest stress. Liquid Penetrant examination located discontinuities in some weld craters (Fig. 13). These discontinuities are being evaluated by metallographic examination.

The superheater was essentially an evaporator installed in an inverted position. The baffle nest, which thus was suspended from inadequate welds, had broken away from its support ring. It was found displaced about 22 inches, but had "floated" in the flow stream during operation within 2-in. of its as-built location (Fig. 14).

In general, the superheater was found to be in remarkably "like new" condition with even the original soap stone marks clearly visible on the baffle nest.

Materials Examinations

Materials examinations of the duplex tubing are in process; some results are reported below. Examinations of the remainder of the subcomponents, i.e., the shell, the baffle nest, the tubsheets, and the inlet and outlet reducers revealed no deleterious effects of operation.

Duplex Tubing Examinations

To determine the contact pressure at the interface (radial stress) between the tubes, the Sachs method[1] was used. This method requires the installation of strain gauges on either the tube bore or outside surface. These strain gauges measure the change in circumferential and longitudinal strain as either the bore or outside surfaces are removed in small increments. The use of this method allows the calculation of radial stress (contact pressure) tangential stress, and longitudinal stress. The testing program required specimens from three different archive tubes and four superheater tubes. Two of those tubes were known to have reduced steam temperature. One had normal steam temperature, and one was both reduced and normal at various times in life. Test specimens, 153 mm long, were cut from four locations within the tube; near the steam inlet, near the steam outlet, the midpoint, and 7880 mm from the steam inlet. Additional specimens (153 mm) were cut from the three archive tubes. Two strain gauges were mounted in the circumferential direction, 180° apart, and two strain gauges in the longitudinal direction, 180° apart. The specimen was placed in a fixture and 0.25 mm increments were bored from the inside with measurements obtained after each increment. The residual stress (radial, tangential, and longitudinal) distributions were then calculated by means of the Sachs equations.

The contact pressures obtained from the tubes in the superheater were lower than those obtained from the archive tubes. Also the contact pressures were highest at the steam inlet/sodium outlet end of the unit, the end with the lowest operating temperature (Fig. 15). The steam outlet/sodium inlet had the lowest contact pressure, and was the end with the highest operating temperature. It appears the contact pressures were temperature sensitive. Thus, it is feasible that some creep or stress relaxation of the material has occurred in the tubes. All of the tubes examined reveal the same trend in contact pressure relaxation.

Tensile test specimens were taken from the tube wall adjoining the Sachs specimens. The results of these tests, indicative of a stress relaxation anneal, confirm the trends obtained from the Sachs specimen.

The tensile test specimens indicated a reduction in ultimate tensile strength and increase in elongation as the distance from the steam inlet/sodium outlet increased.

ACKNOWLEDGEMENTS

The work reported herein was performed under the auspices of the United States Department of Energy with the support of the Electric Power Research Institute.

REFERENCES

(1) Buschman, H. W., Cerutti, B. C., and Clark, A. F., Noise Investigation and Repair of the EBR-II Intermediate Heat Exchanger," Argonne National Laboratory report ANL-7834, August 1971.

(2) K. J. Longua, et al, Second Mechanical In-Service Inspection of EBR-II Steam Generator, Argonne National Laboratory report ANL-78-86, November 1978.

(3) Penney, Wm. H., et al, In-Service Inspection of Superheater 712 at EBR-II, Argonne National Laboratory report ANL-80-2, March 1980.

(4) Penney, Wm. H., Buschman, H. W., and Washburn, R. A., Disassembly and Phase 1 Examination of EBR-II Superheater SU-712, Argonne National Laboratory report ANL-82-16, May 1982.

TABLE 1

EBR-II HEAT TRANSPORT SYSTEM OPERATING PARAMETERS

Power	62.5 MWt
Secondary Sodium Flow	299.5 kg/s (5400 gpm)
Secondary Sodium Temp.	
From IHX	466°C (870°F)
To Evap.	427°C (800°F)
To IHX	307°C (585°F)
Feedwater Temperature	288°C (550°F)
Steam Drum Pressure	9.14 MPa (T_{sat} = 304°C)
	(1325 Psi, Tsat = 580°F)
Superheated Steam Temperature	438°C (820°F)
Blowdown	2.5 kg/s (20,000 lb/h)
Steam Flow	30.9 kg/s (245,000 lb/h)
Circulation Ratio(c)	10

TABLE 2

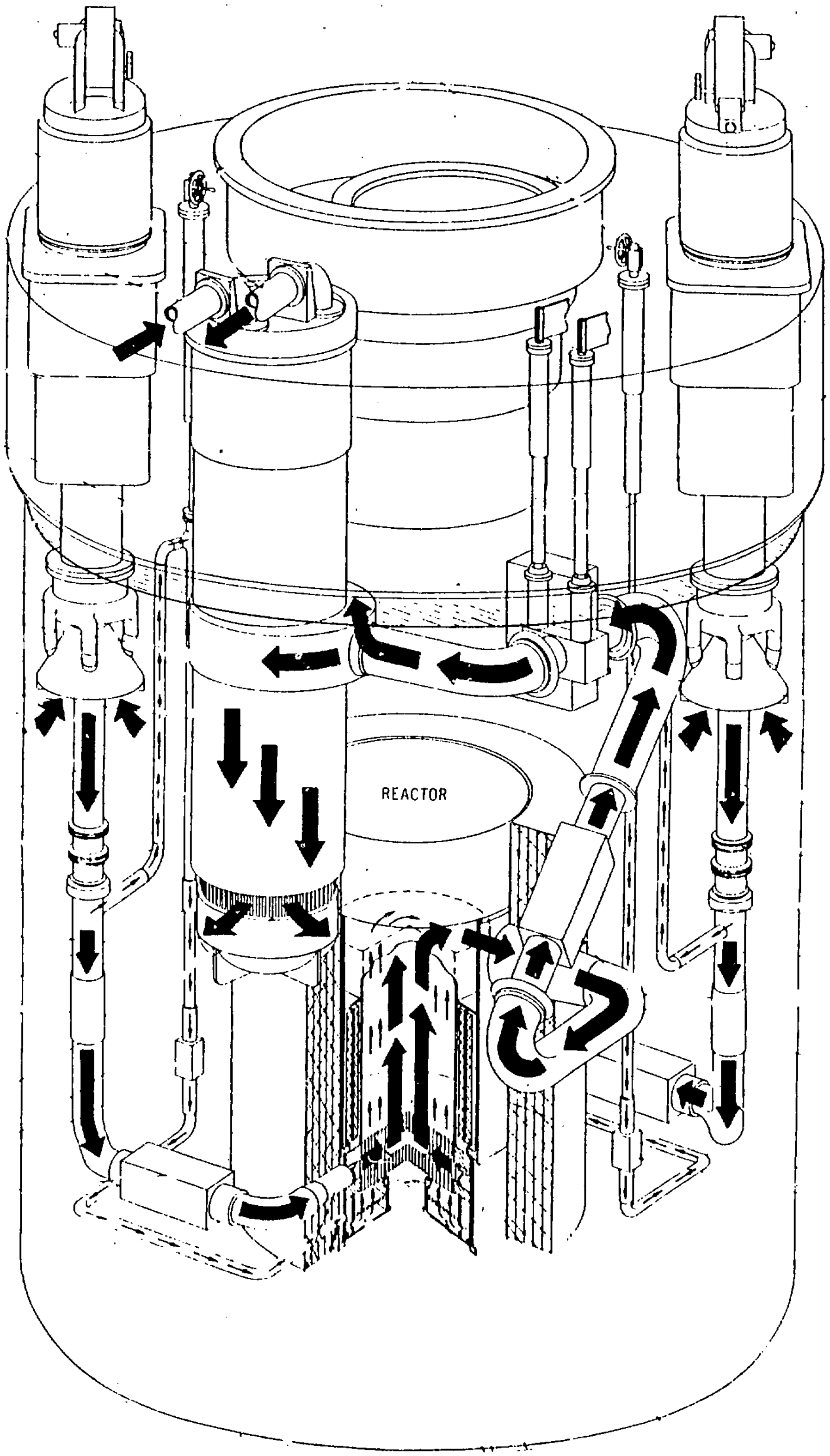
FEEDWATER QUALITY DURING POWER OPERATION

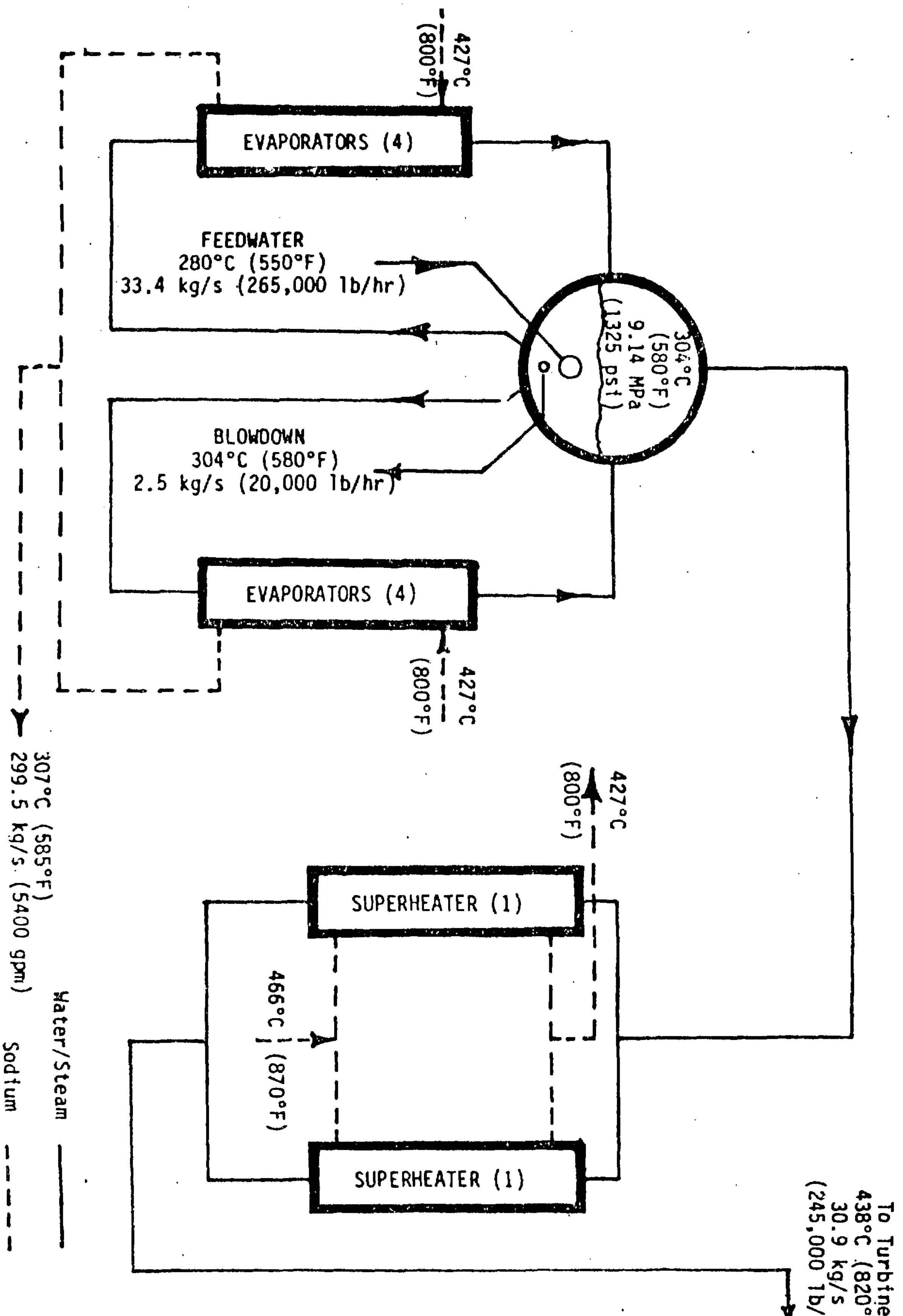
	EBR-II[a]	LMFBR Conceptual Design Study[b]
Total Dissolved Solids	∿0	50 ppb (max.)
Dissolved Oxygen	< 5 ppb	5 ppb (max.)
Silica	<10 ppb	10 ppb (max.)
Iron	<10 ppb	10 ppb (max.)
Copper	<20 ppb	< 2 ppb
pH	8.6 - 9.2	8.5 - 9.3
Hydrazine (Residual)	10 - 20 ppb	20 - 100 ppb
Sodium	<0.1 ppb	< 3 ppb
Chlorides	< 20 ppb	< 3 ppb

[a]Concentrations shown as < are lower than measurement capability.

[b]Unpublished data

Figure 1





427°C (800°F)
EVAPORATORS (4)
FEEDWATER
280°C (550°F)
33.4 kg/s (265,000 lb/hr)
304°C (580°F) 9.14 MPa (1325 psi)
BLOWDOWN
304°C (580°F)
2.5 kg/s (20,000 lb/hr)
EVAPORATORS (4)
427°C (800°F)
307°C (585°F)
299.5 kg/s (5400 gpm)
427°C (800°F)
SUPERHEATER (1)
466°C (870°F)
SUPERHEATER (1)
Water/Steam
Sodium
To Turbine
438°C (820°F
30.9 kg/s
(245,000 lb/h

Figure 2

Figure 3

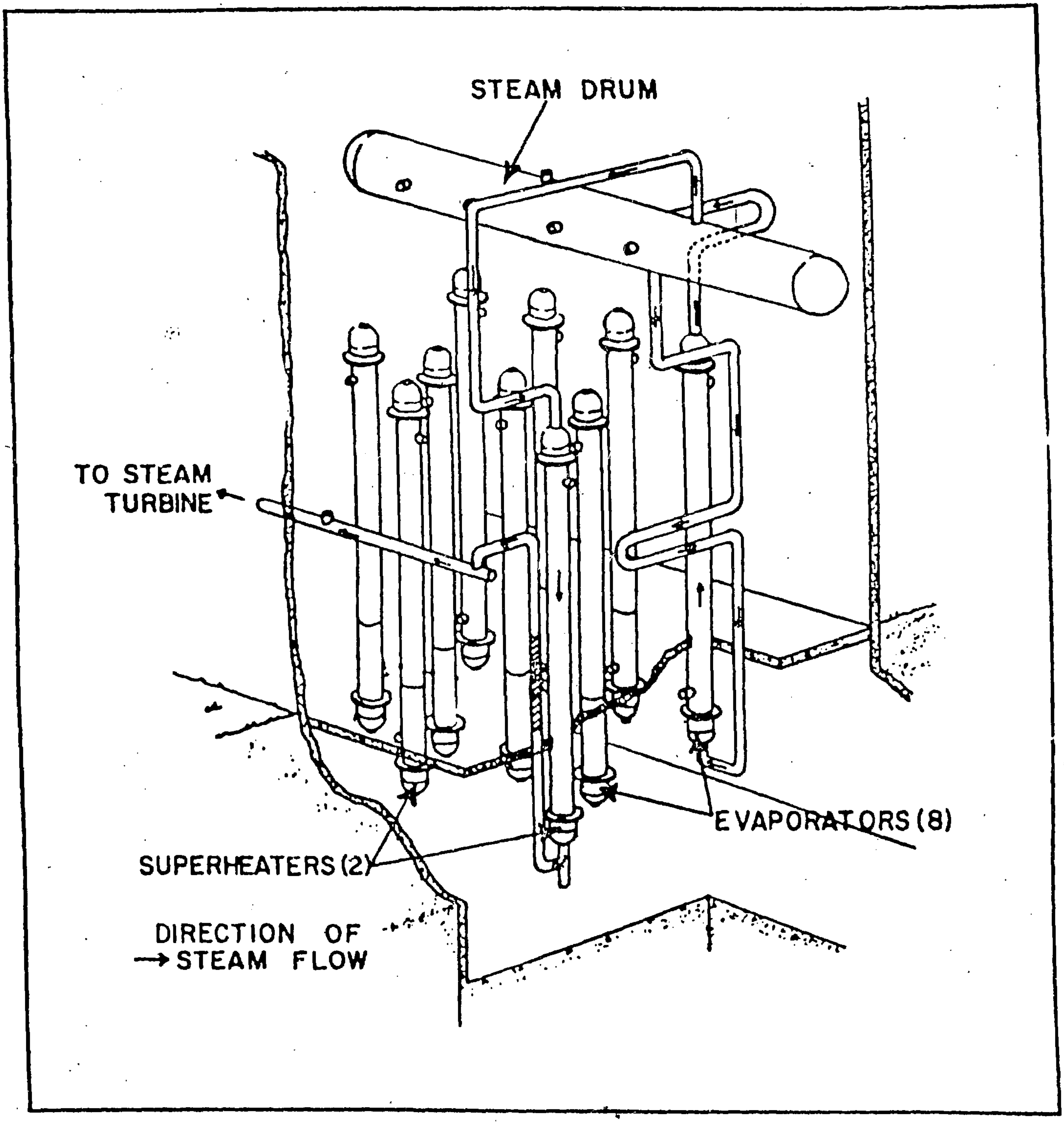
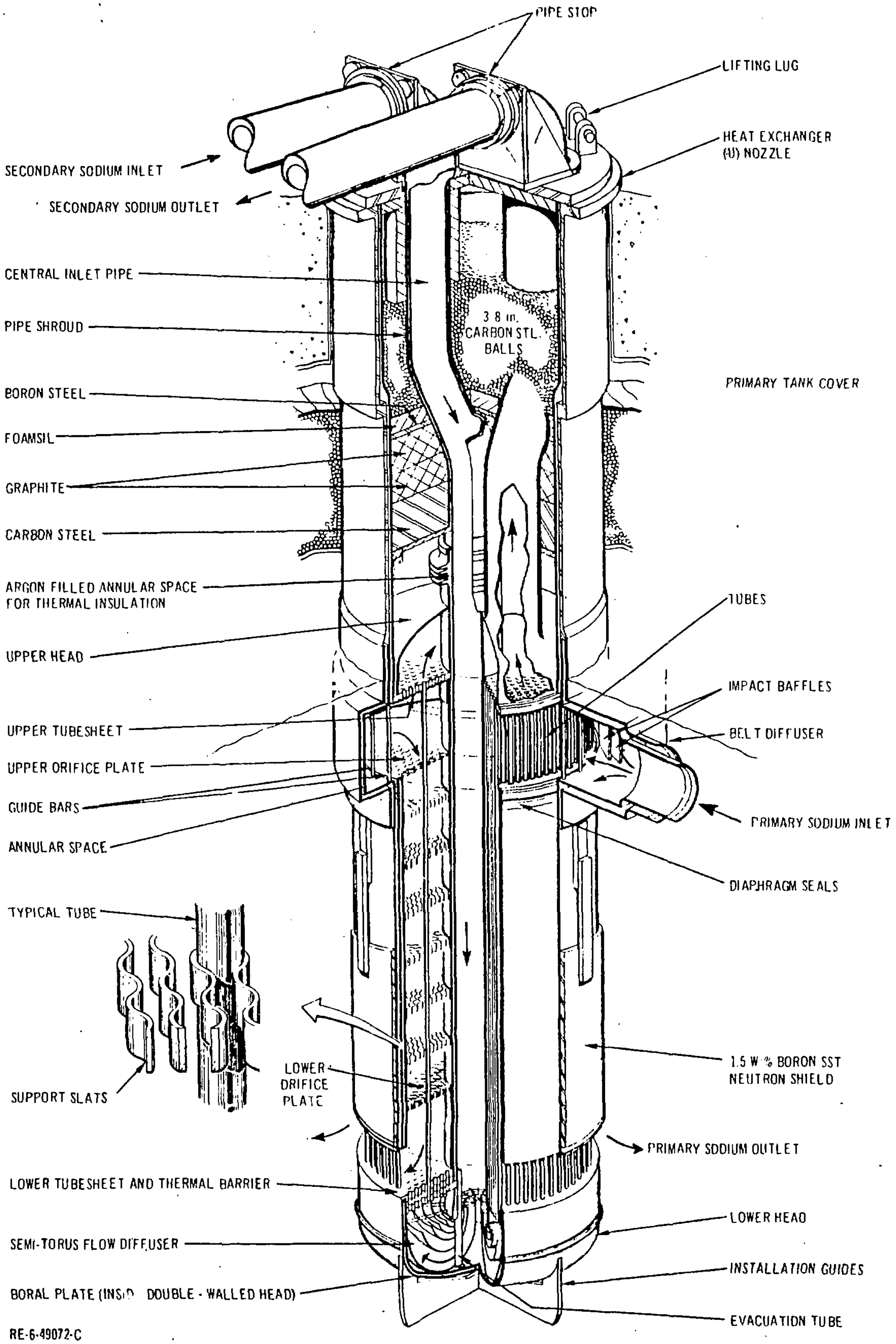

STEAM DRUM
TO STEAM TURBINE
SUPERHEATERS (2)
EVAPORATORS (8)
DIRECTION OF
→ STEAM FLOW

Figure 4



Intermediate Heat Exchanger

Figure 5

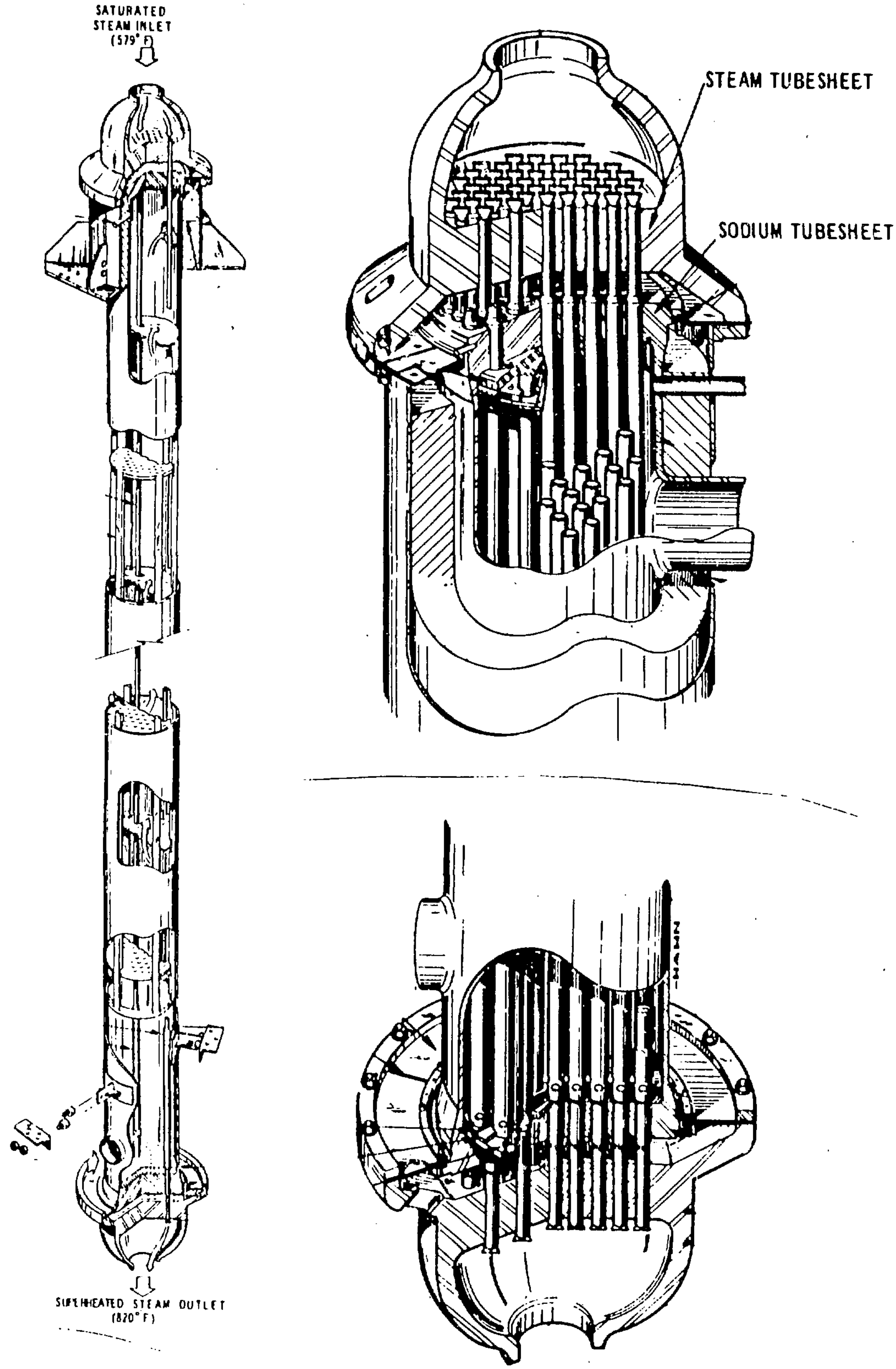

SATURATED STEAM INLET (579° F)
STEAM TUBESHEET
SODIUM TUBESHEET
SUPERHEATED STEAM OUTLET (820° F)

Figure 6

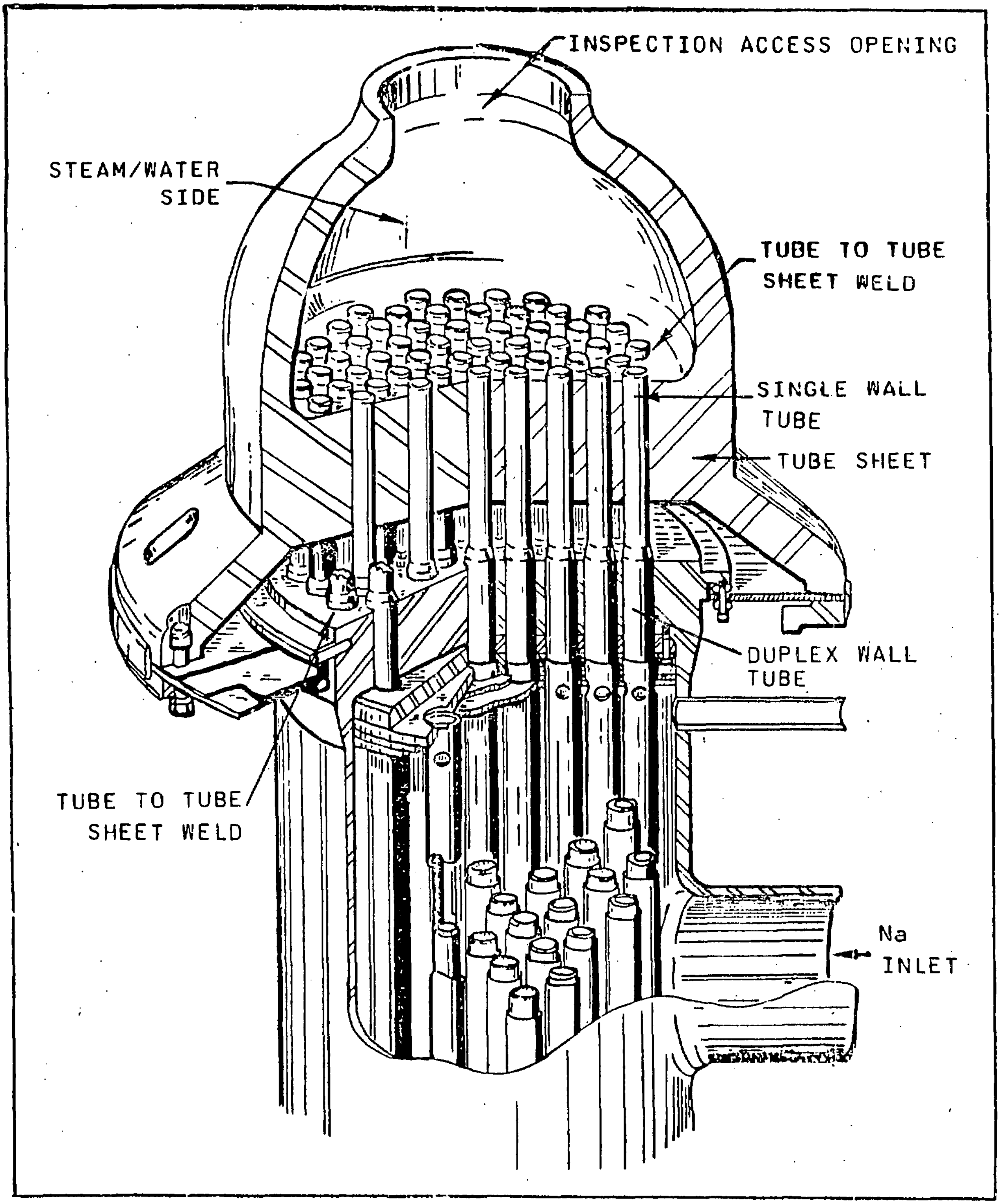
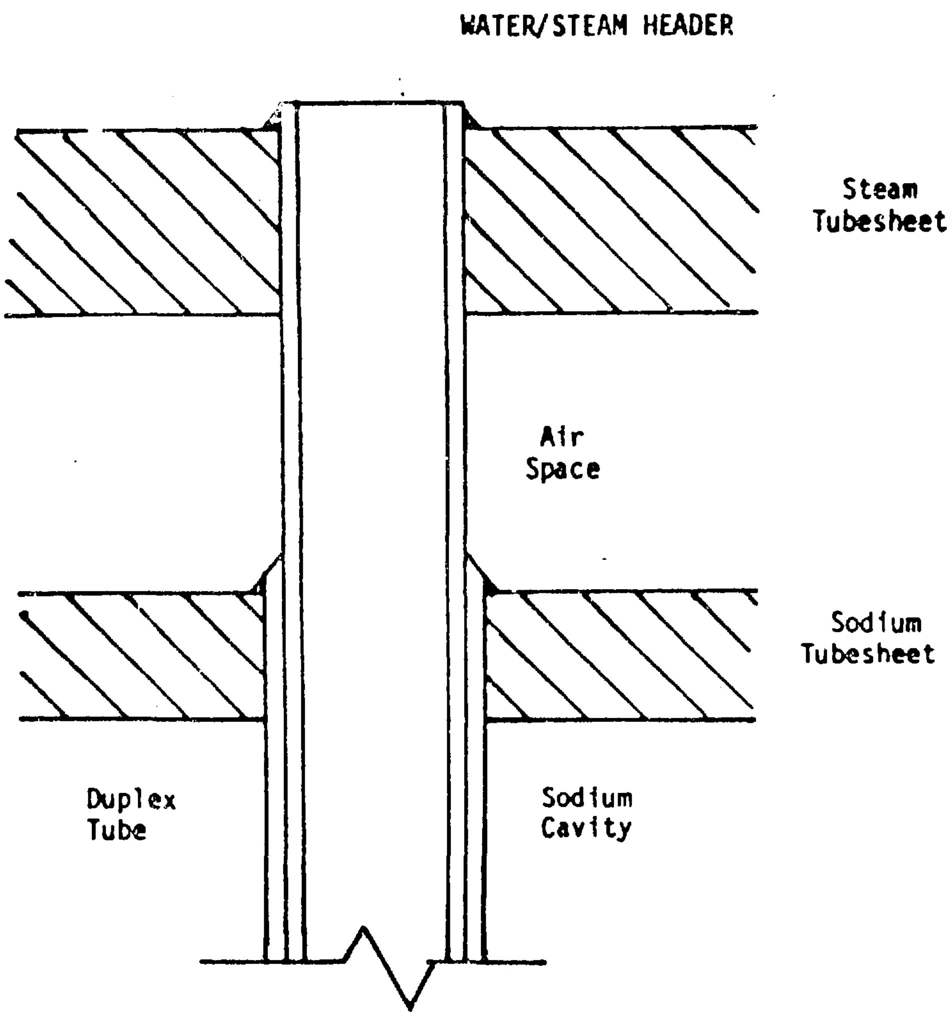

INSPECTION ACCESS OPENING
STEAM/WATER
SIDE
TUBE TO TUBE
SHEET WELD
SINGLE WALL
TUBE
TUBE SHEET
DUPLEX WALL
TUBE
TUBE TO TUBE
SHEET WELD
Na
INLET

Figure 7



TUBE DIAMETERS

Inner Tube	Outer Tube
I.D. - 27.05 mm (1.065 in.)	I.D. - 31.75 mm (1.250 in.)
O.D. - 31.75 mm (1.250 in.)	O.D. - 36.53 mm (1.438 in.)

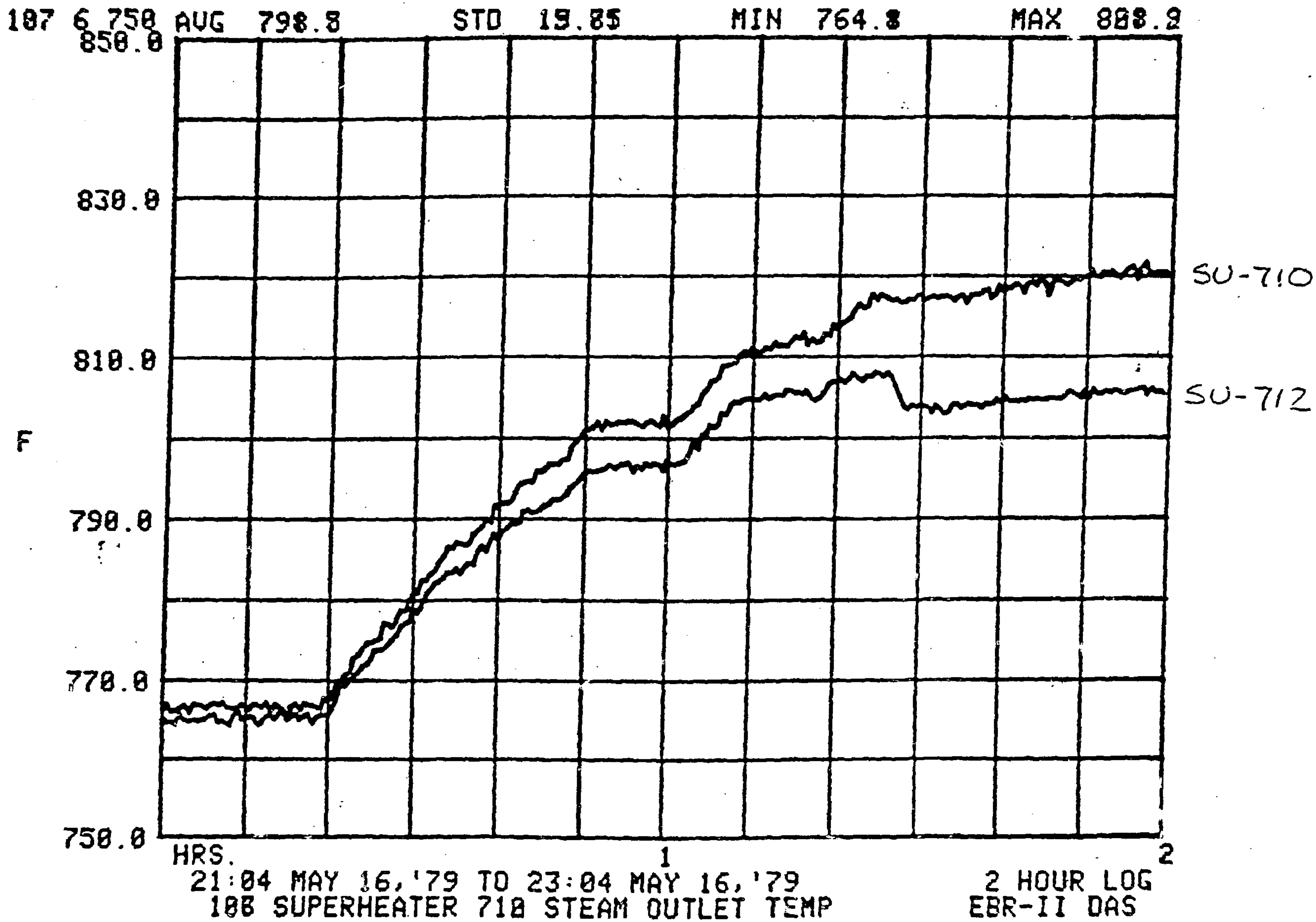
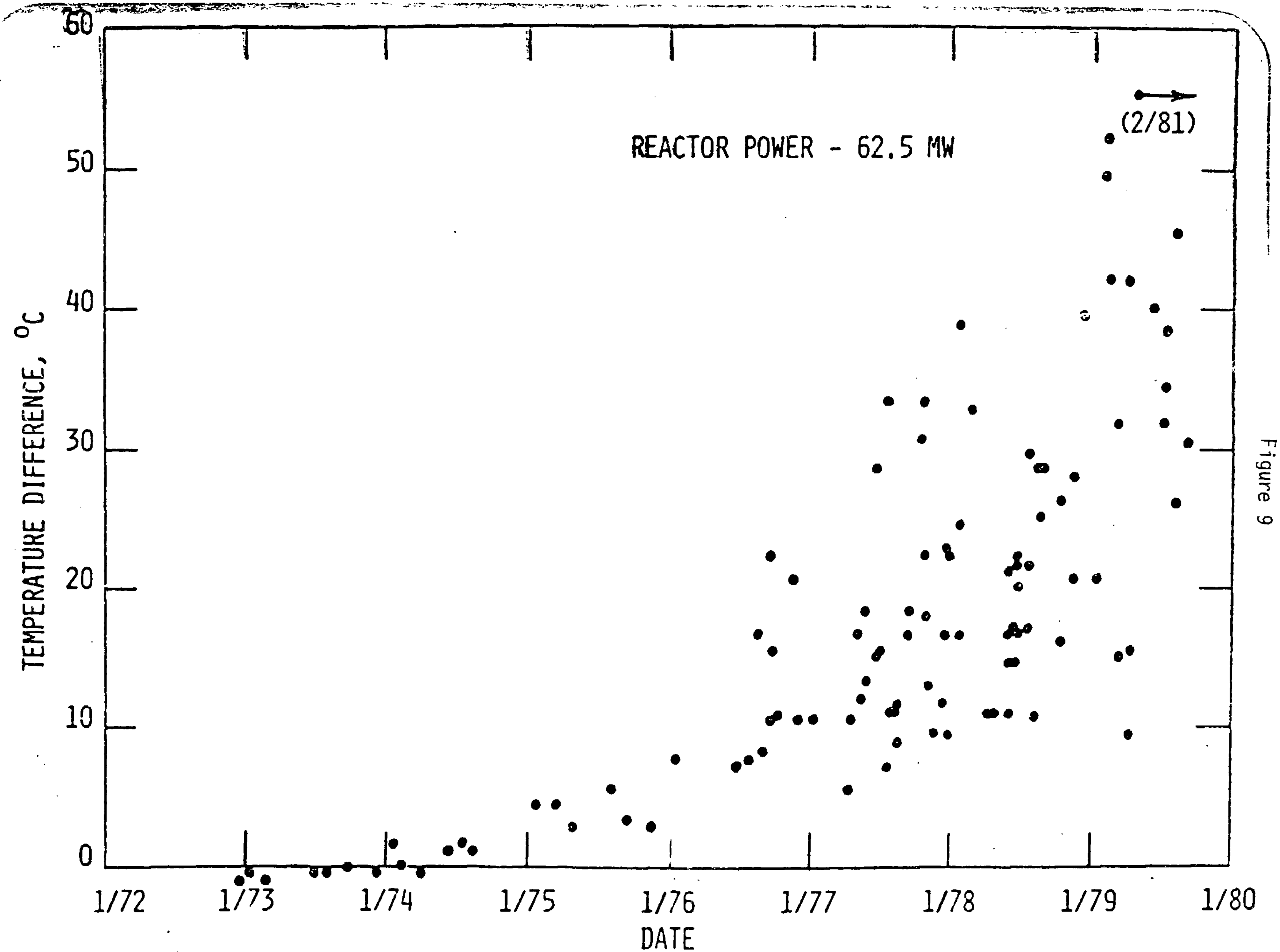

187 6 750 AVG 790.8 STD 19.85 MIN 764.0 MAX 800.2
850.0
830.0
810.0
790.0
770.0
750.0
F
SU-710
SU-712
HRS.
1
2
21:04 MAY 16,'79 TO 23:04 MAY 16,'79
2 HOUR LOG
100 SUPERHEATER 710 STEAM OUTLET TEMP
EBR-II DAS

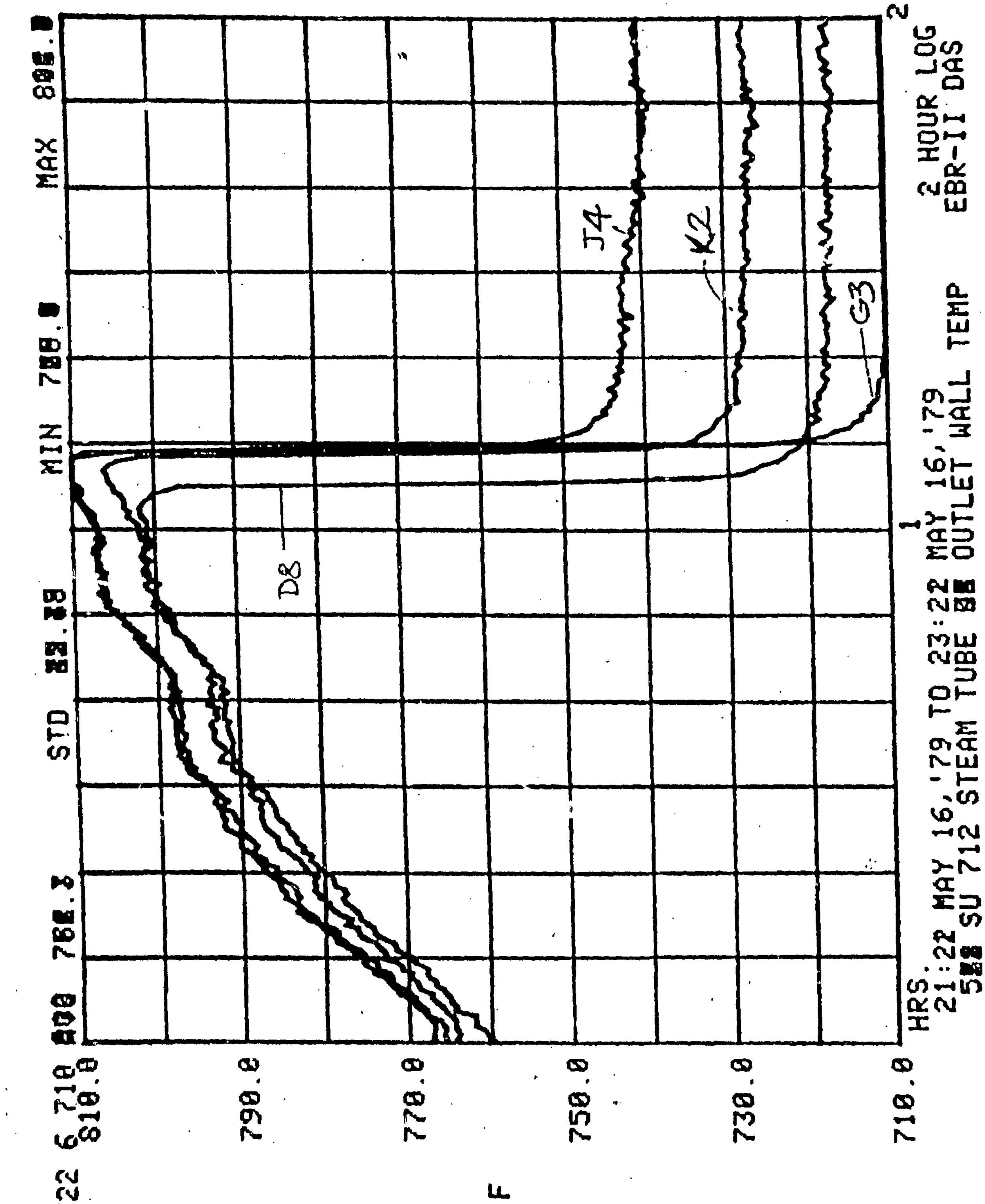
Figure 8

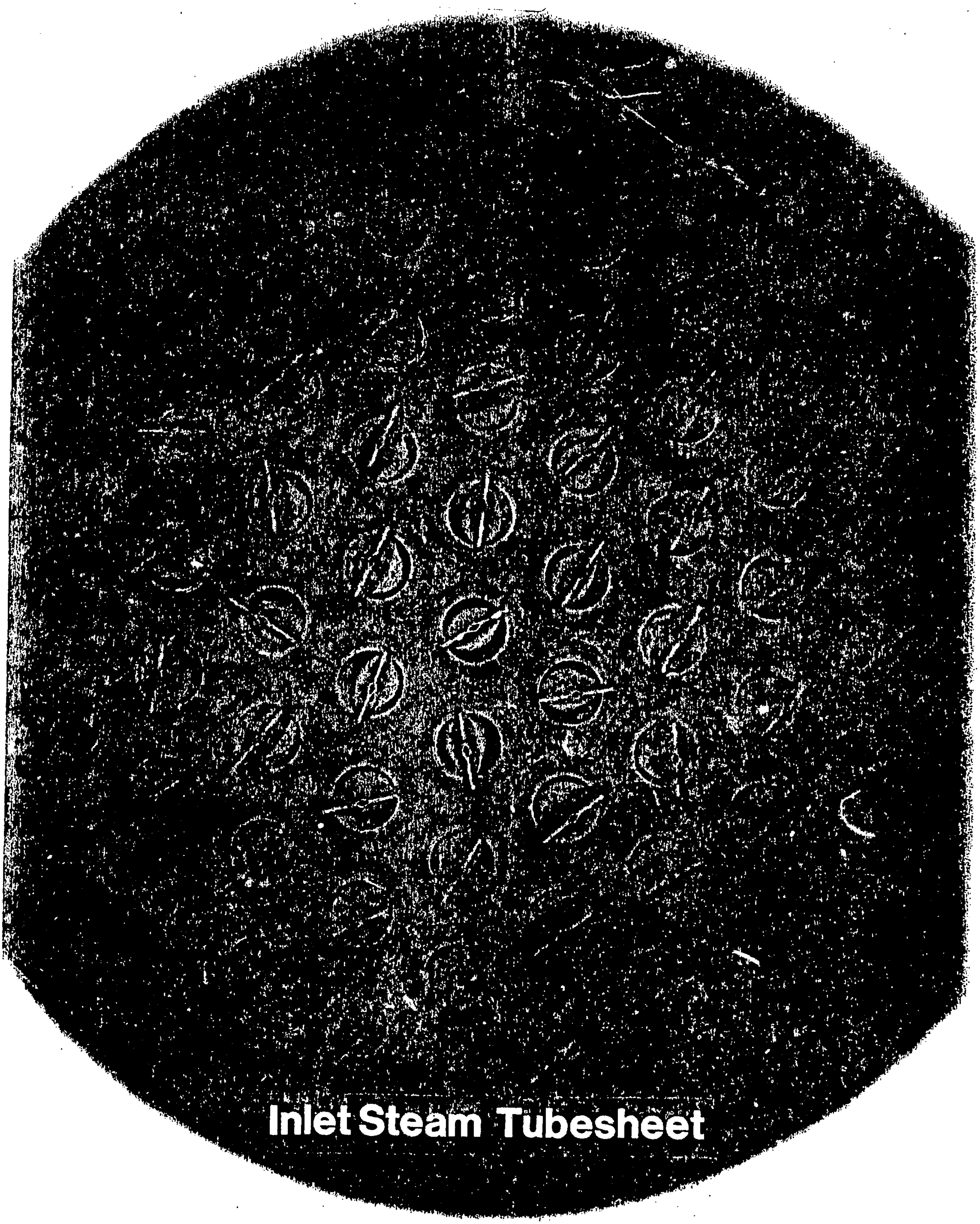


SUPERHEATER STEAM OUTLET TEMPERATURE DIFFERENCE

Figure 9

Figure 10


MAX
MIN
STD
2
1
HRS.
21:22 MAY 16,'79 TO 23:22 MAY 16,'79
SU 712 STEAM TUBE
OUTLET WALL TEMP
2 HOUR LOG
EBR-II DAS
J4
K2
G3
D8
522
6.710
810.0
790.0
770.0
750.0
730.0
710.0
F


Inlet Steam Tubesheet

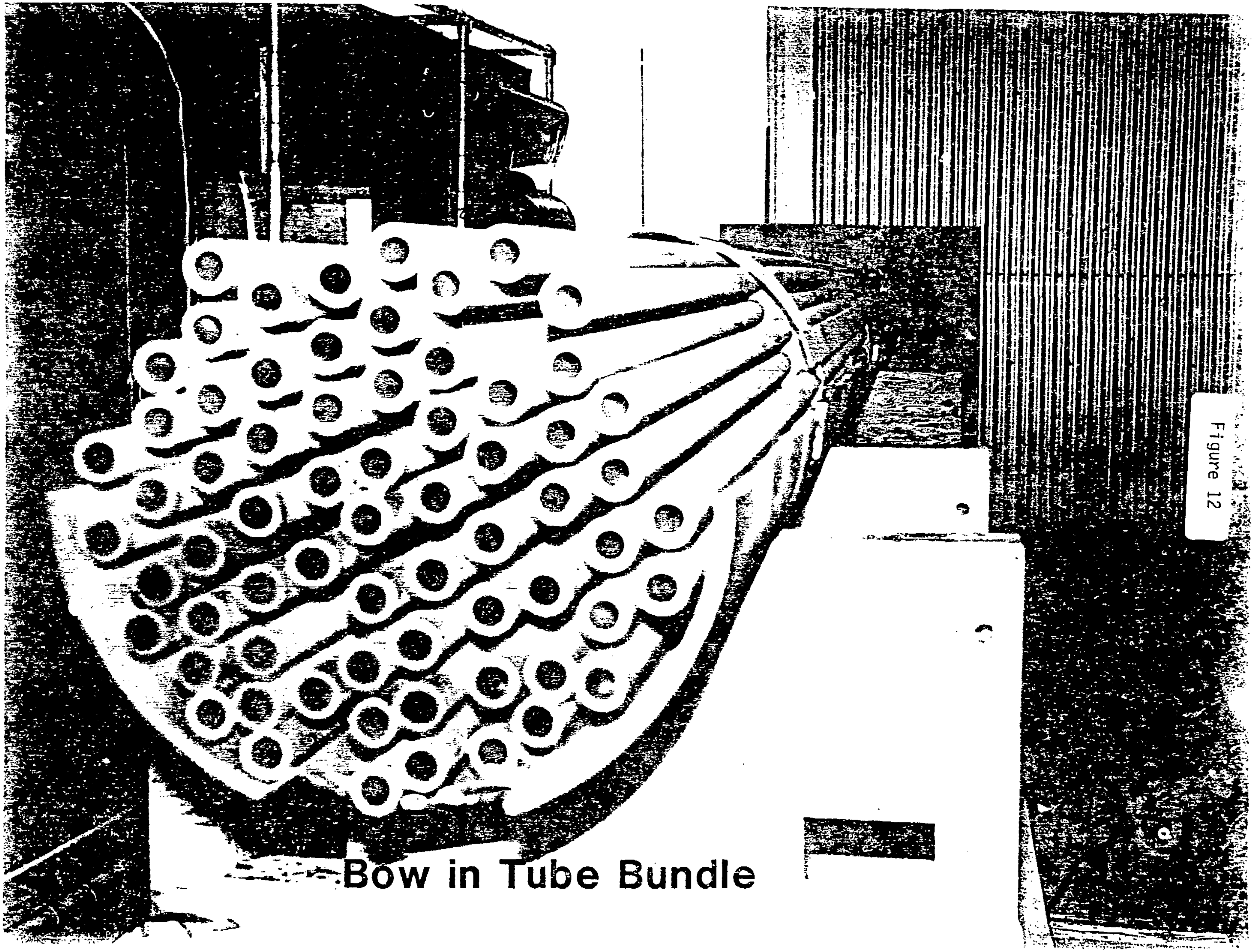
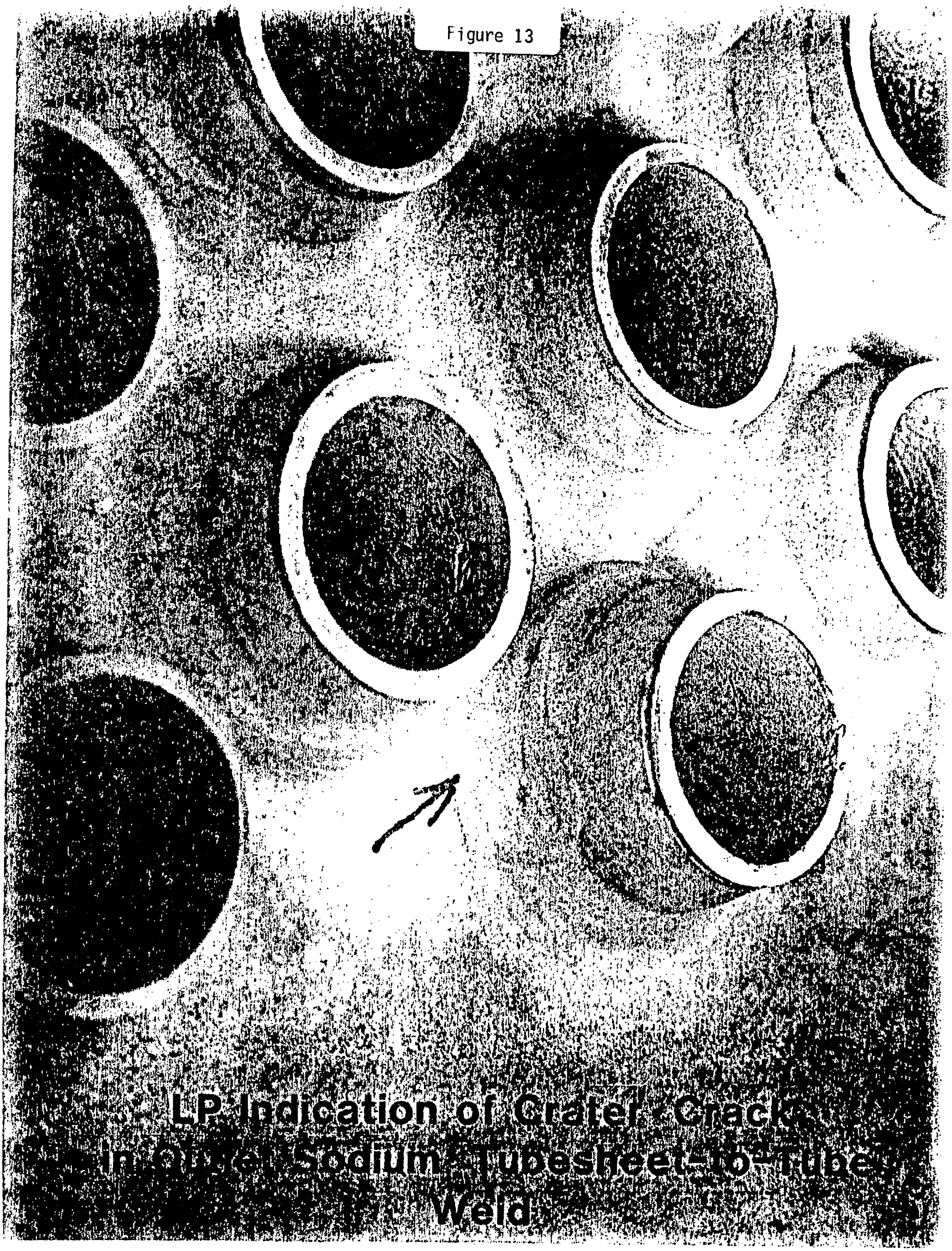


Figure 12

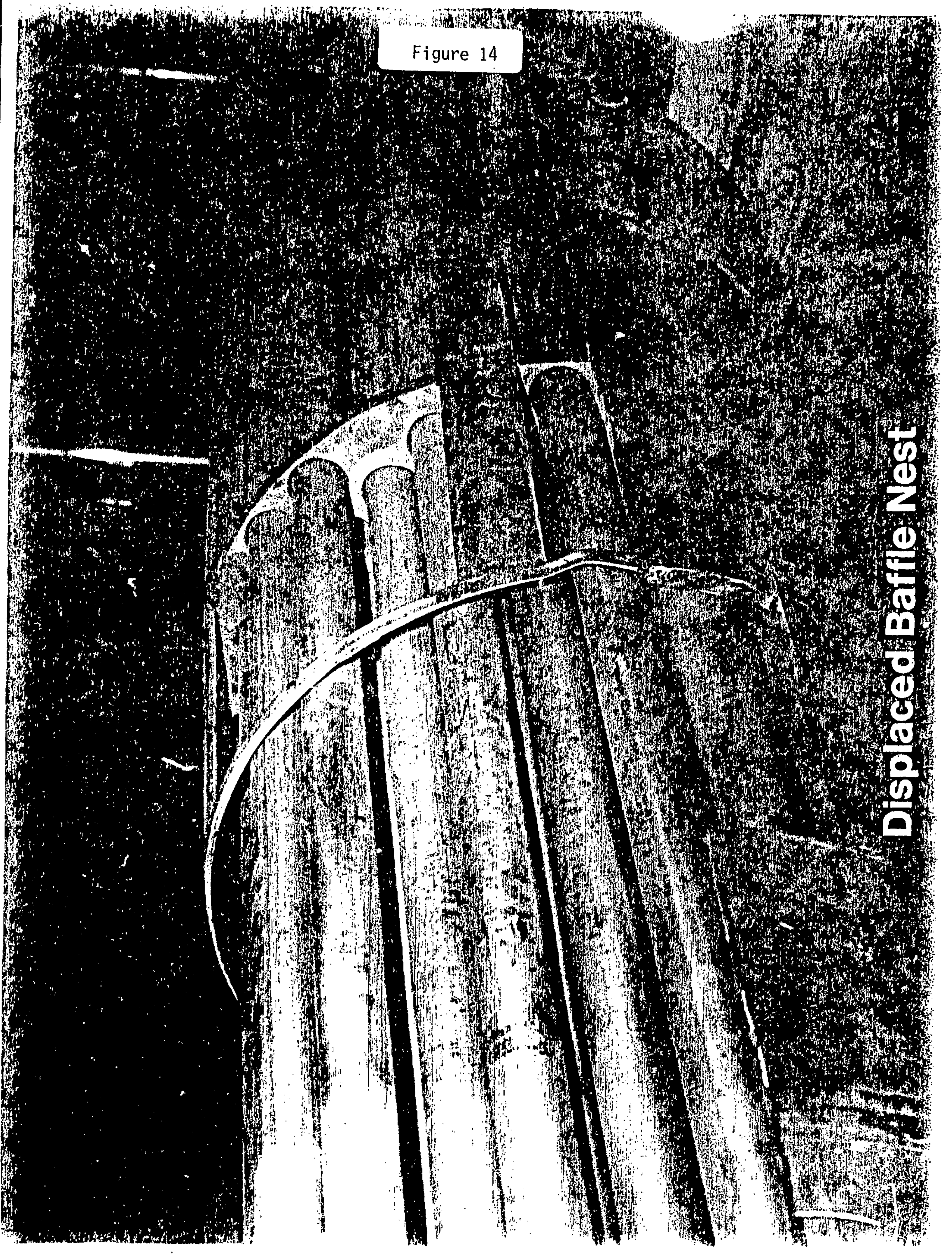
Bow in Tube Bundle

Figure 13



LP Indication of Crater Crack In Outer Sodium Tubesheet-To-Tube Weld

Figure 14



Displaced Baffle Nest

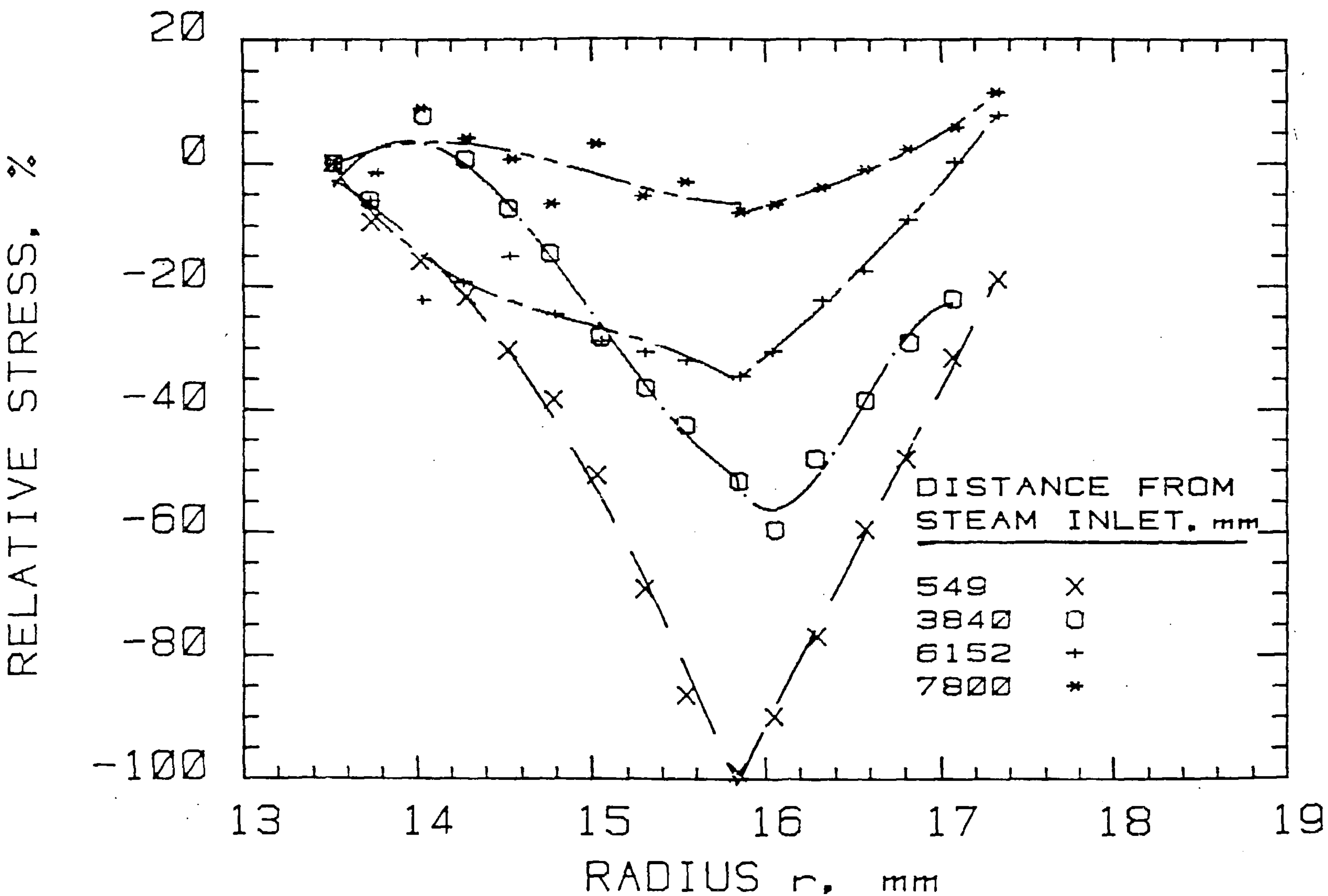

20
0
-20
-40
-60
-80
-100
RELATIVE STRESS, %
13
14
15
16
17
18
19
RADIUS r, mm
DISTANCE FROM
STEAM INLET, mm
549 X
3840 O
6152 +
7800 *

Figure 15