NAA-SR-3969
COPY

PRELIMINARY TEST OF NATURAL-CIRCULATION DOUBLE-TUBE STEAM GENERATOR

AEC Research and Development Report

ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.

Price $1.00
Available from the Office of Technical Services
Department of Commerce
Washington 25, D. C.

NAA-SR-3969
REACTORS-POWER
35 PAGES

PRELIMINARY TEST OF NATURAL-CIRCULATION DOUBLE-TUBE STEAM GENERATOR

By
R. D. WELSH

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. BOX 309 CANOGA PARK, CALIFORNIA

CONTRACT: AT(11-1)-GEN-8
ISSUED: DEC 1 1959

DISTRIBUTION

This report has been distributed according to the category "Reactors-Power" as given in "Standard Distribution Lists for Unclassified Scientific and Technical Reports" TID-4500 (15th Ed.), August 1, 1959. A total of 620 copies was printed.

CONTENTS

TABLES

FIGURES

ABSTRACT

Testing of the Natural Circulation Steam Generator has been conducted with the Sodium Reactor Experiment. A necessary modification of the original feedwater control system was utilized. Transient and steady-state tests were run at power levels up to 19.5 Mwt. All tests indicate that the steam generator responds well and is stable.

Overall heat transfer coefficients for the boilers and superheater are, respectively, 507 and 180 Btu/hr-°F-ft^2. These values are within 11.5% of the theoretical values, 449 and 169 Btu/hr-°F-ft^2. Reverse sodium flow occurred in the lower tubes of the boilers, as predicted, when the temperature gradient across the boilers was 94°F and the total sodium flow for the two boilers was 35 gpm. Recommendations are given for further steam generator tests and improvements.

I. INTRODUCTION

The steam generator was designed and built for possible use in the Navy program. The lack of tube sheets gave the steam generator a novel design. The superheater and boiler shells resisted flexing because they were circular; thus, this steam generator design might be scaled up for a large, stationary, power plant without the pressure and thermal stresses usually found at tube sheets.

The steam generator was made available to AI by the U.S. Atomic Energy Commission. It has been tested by operation with the Sodium Reactor Experiment (a 20 Mwt reactor) as the heat source. During initial operation it was found necessary to modify the steam generator controls as described in Section III.

The steam generator was first operated with the SRE on March 13, 1959. It was tested to determine the operating characteristics under both steady-state and transient conditions. The testing to date was intended primarily to assure satisfactory operation and develop operating techniques prior to more intensive testing. The information gained from this and future studies will be used to make recommendations for better steam generator designs taking full advantage of the excellent heat transfer capabilities of sodium. Future transient test data will be used to define operating parameters for the SRE-Edison plant control system which will be used to check out the control system for the Hallam Nuclear Power Facility. The latter also has a drum-type steam generator.

II. DESCRIPTION OF EQUIPMENT

The steam generator (Figures 1, 2, and 3) is a natural-circulation type with separate superheater, steam drum, and two parallel-operating boilers. Operation with a single boiler is also possible. Mercury is used as the third fluid in the annulus of the double-walled tubing inside the boilers and superheater. Feedwater is introduced into the steam drum and then flows down to the boilers where steam is generated. Saturated steam and water flow up from the boilers to the steam drum where the water is separated; then the saturated steam continues to the superheater.

The steam generator was designed to operate with a maximum sodium inlet temperature to the superheater of 850°F and with a maximum steam pressure of 600 psi. A more complete description of this equipment may be found in NAVSHIPS-351-0506,[1] and a cutaway view of the boiler (with the manifolds upside down) may be found in the Liquid Metals Handbook.[2]

At the SRE the steam generator was valved in parallel with the existing once-through steam generator (Figure 4) and was operated separately. The SRE was used as the heat source, and the Southern California Edison Company Electrical Power Generation Facility (with the once-through steam generator valved off for this test) was used as the heat sink.

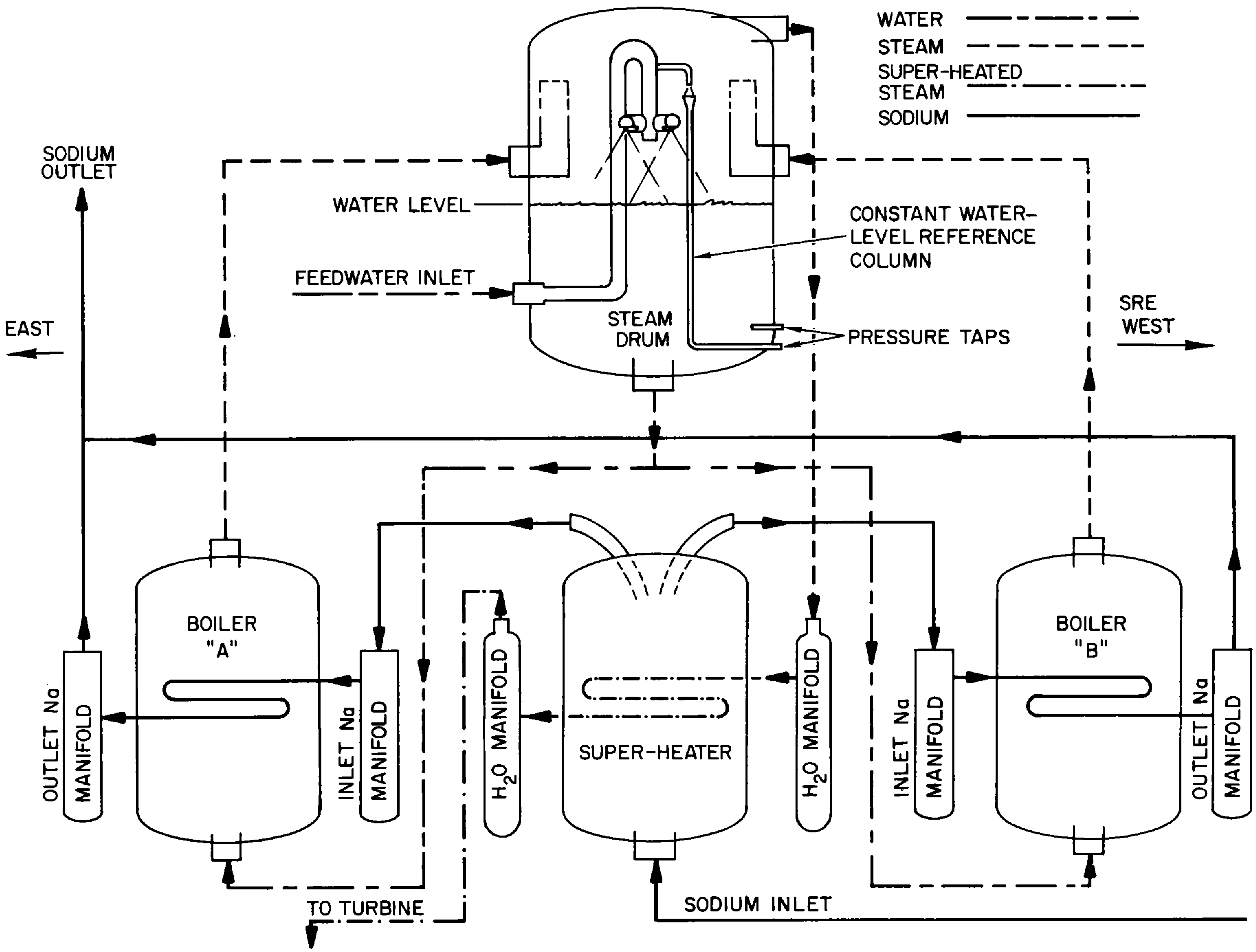

Figure 1. Natural Circulation Steam Generator Schematic

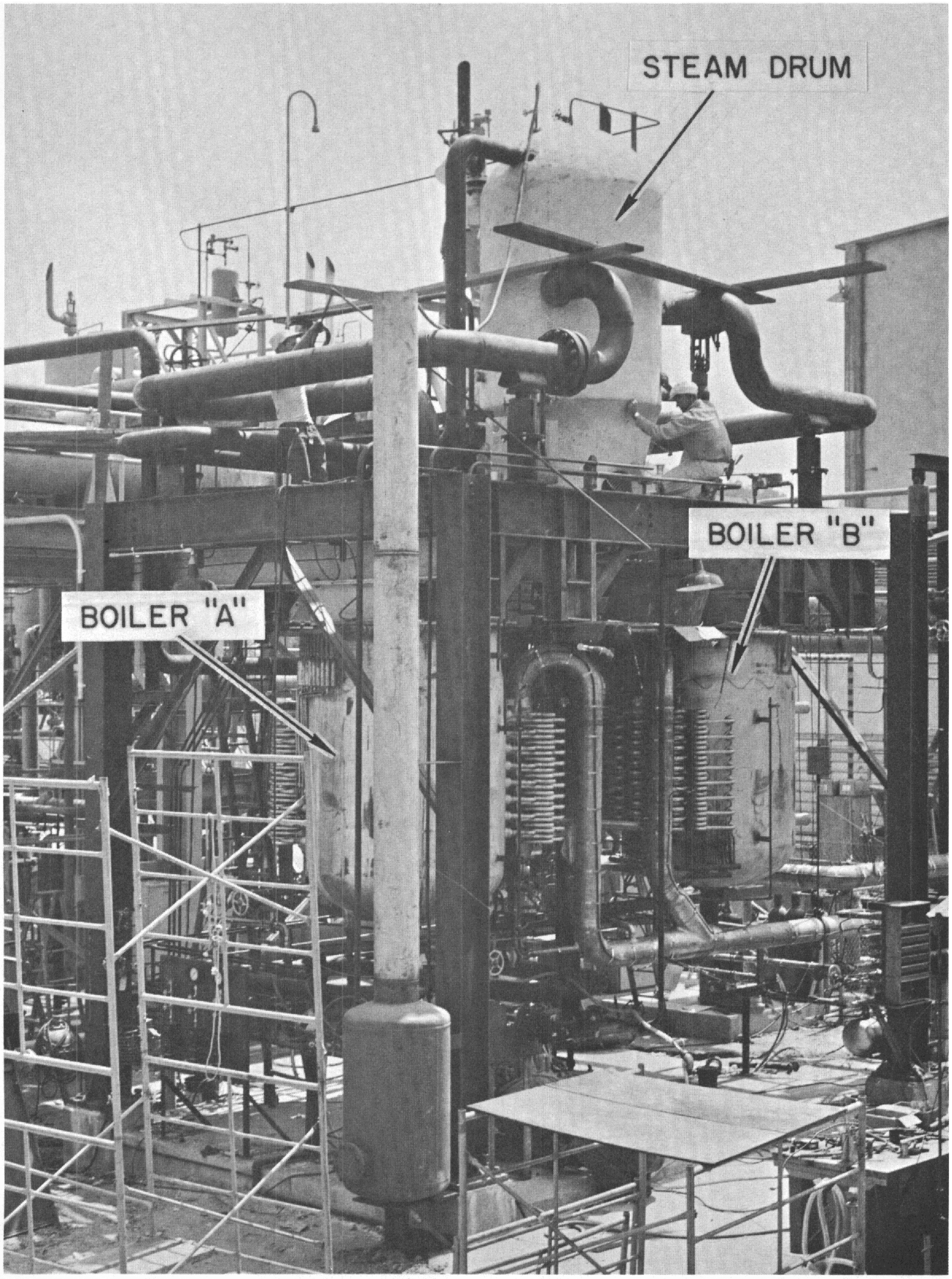

Figure 2. Steam Generator

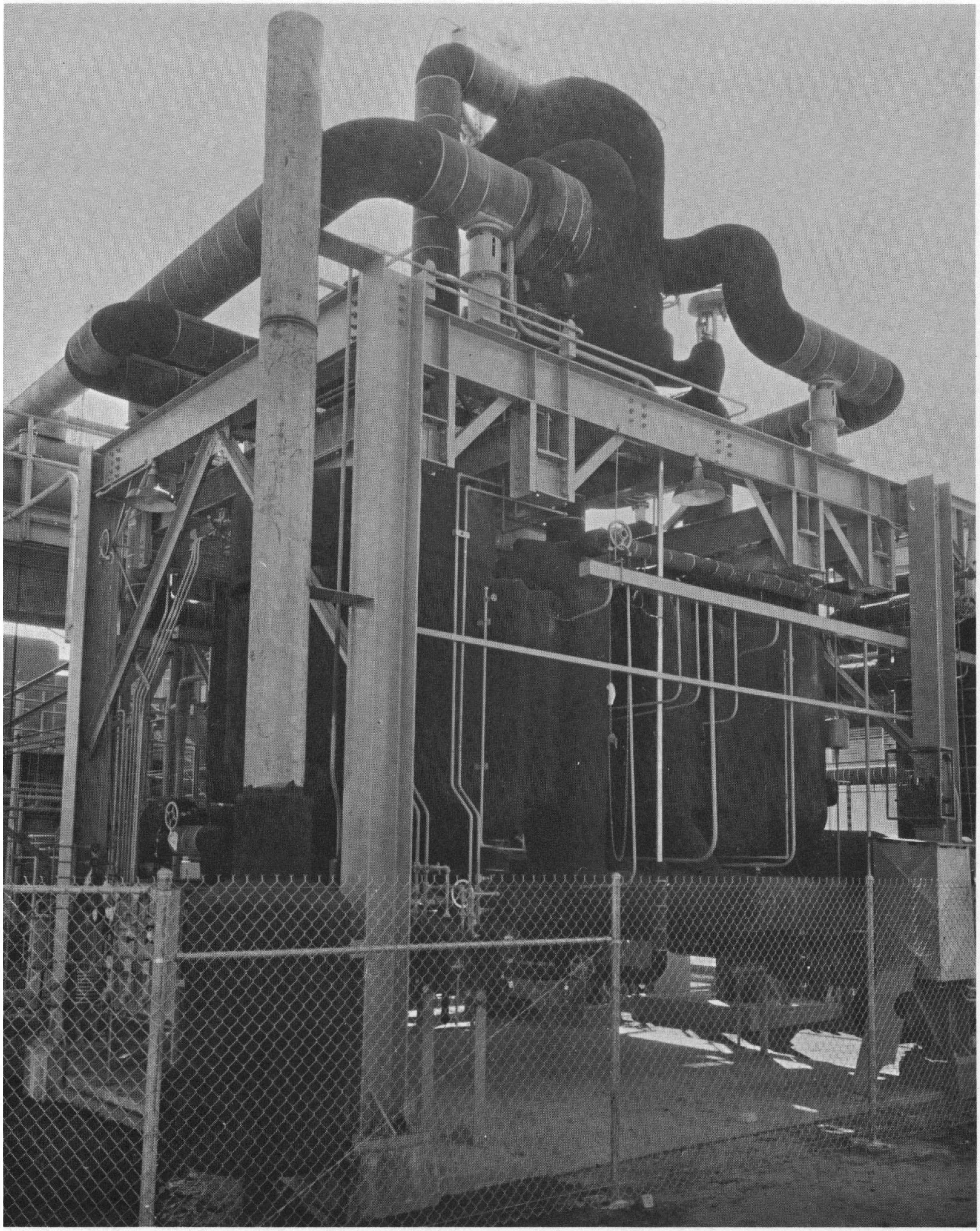

Figure 3. Steam Generator Insulated

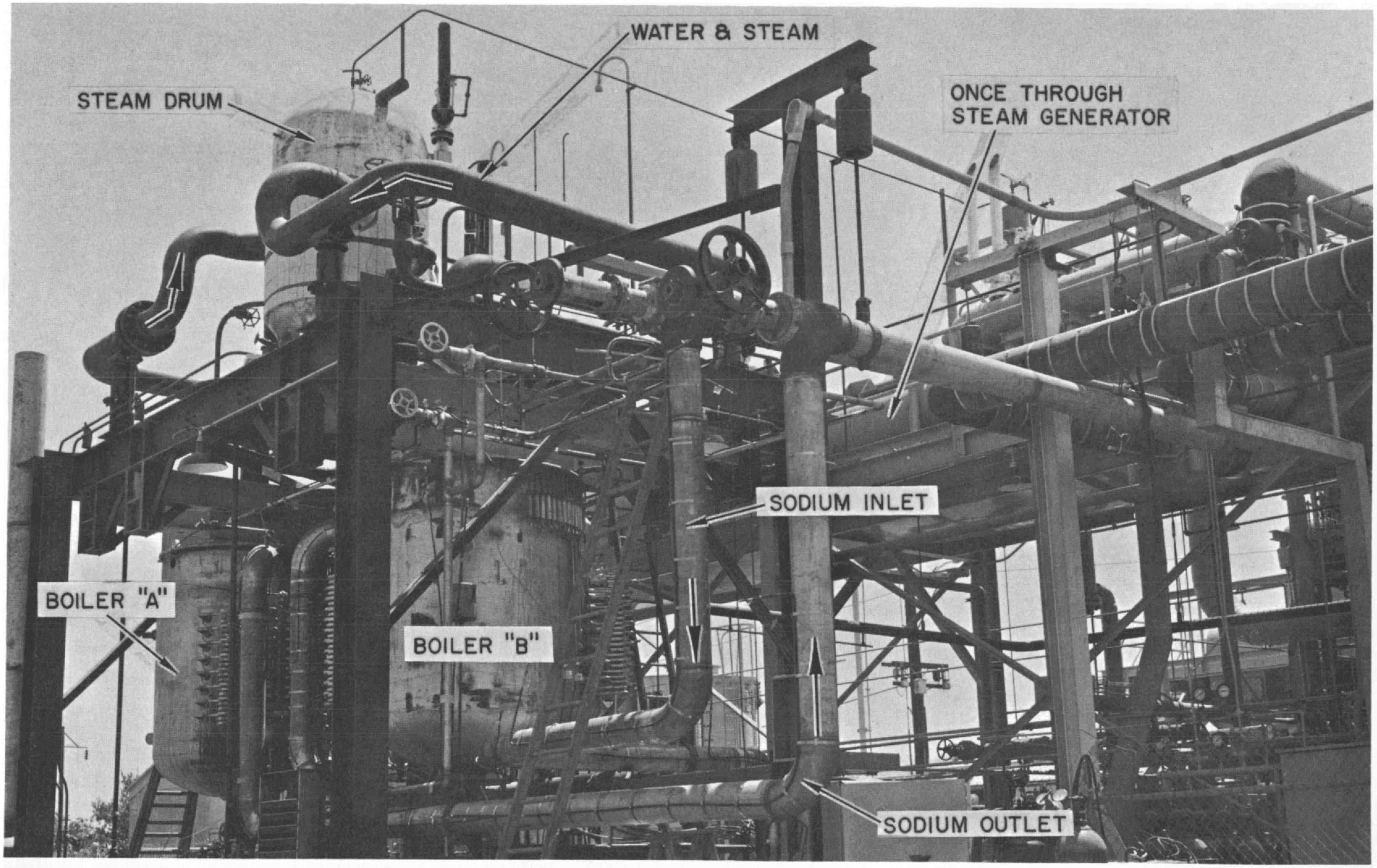

Figure 4. Natural Circulation and Once-through Steam Generator

III. INITIAL OPERATION

A. WATER CONDITION

The steam generator was washed out with raw water and rewashed with demineralized water before startup. The boilers were blown down immediately after startup. During initial operation of the steam generator, black, flocculent material was discharged from the boilers, at irregular intervals, with the blowdown water. Blowing down was continued at a high rate, limited only by the rate that demineralized makeup water could be supplied, until the blowdown water cleared. Then blowing down was continued intermittently for the remainder of the operating time.

The material in the blowdown water was examined by the Steam Division Laboratory of the Southern California Edison Company at the Redondo Steam Station and was found to be asbestos, impregnated with black iron oxide. This material, probably insulating material, apparently entered one of the boilers or the steam drum when the steam generator was being disassembled and shipped to AI.

B. STEAM-DRUM WATER-LEVEL CONTROL

One of the initial tests of the steam generator was that on the three-element automatic feedwater controller (drum level, steam flow, and feedwater flow). It was observed that the indicated water level would decrease when the feedwater flow rate was increased. A test was conducted in which feedwater flow rate was rapidly increased from approximately 6000 lb/hr to various rates up to 100, 000 lb/hr. The actual level changes (as observed in the steam-drum sight gage) were negligible during the short duration of these tests. The level controller indicated an apparent level change in excess of 5 ft at the highest flow rate. The error in the indicated level varied approximately with the square of the feedwater flow rate. The water level was indicated from the calibration of a differential pressure measurement between the static pressure at the bottom of the inside of the steam drum and the base of a constant water-level reference column (Figure 1). The reference column was kept at a constant level by a small jet of water. The velocity head of the jet of water from the main feedwater inlet line was found to add to the static head of the column, thus producing the error in indicated water level. A calculation to verify this hypothesis indicated an apparent level change of approximately 8 ft for the maximum feedwater flow rate. The experimental value of the indicated water-level error was in excess of 5 ft for the same feedwater flow rate.

A new water-level reference column was installed on the outside of the steam drum. This location is standard design practice on stationary steam generators. The external reference column was kept full by bleeding steam from the upper sight-gage connection and using a condensing chamber. This reference column was found to be free of feedwater flow-rate effects. At the same time, the range of the level transmitter was extended from its original value of 30 in. to a new value of 60 in.[3] As a result of these modifications, the control system operated very satisfactorily, with good response and stability.

IV. STEADY STATE PERFORMANCE

A. EXPERIMENTAL VALUES OF OVERALL HEAT TRANSFER COEFFICIENTS

Comparison of experimental and theoretical values of overall heat-transfer coefficients indicated that the steam generator performed better than expected. The test data of Table I were used to calculate the overall heat-transfer coefficients.

TABLE I

TEST DATA FROM APRIL 1, 1959

Sodium Temperature Into Superheater (°F)	795
Sodium Temperature Into Boilers (°F)	760
Sodium Temperature Out of Boilers (°F)	495
Sodium Flow (lb/hr)	712,000
Feedwater Temperature (°F)	280
Feedwater (or Steam) Flow (lb/hr)	62,000
Boiler Inlet Water Temperature (°F)	480
Steam Temperature (°F)	660
Steam Pressure (psia)	565
Thermal Power (Mw)	19.5

The experimental value of the overall heat-transfer coefficient for the boilers was calculated from the equation $Q = U_b A_b \Delta T_L$ where Q was the heat transferred in the boilers.* Similarly, the value of U was calculated for the superheater. The superheater was treated as a single-pass, cross-flow heat exchanger with one fluid (sodium) mixed and the other unmixed for the calculation of the log mean-temperature difference.

The values of U, calculated from the above test data, were 507 and 180 Btu/hr-°F-ft^2 for the boilers and superheater respectively.

*Nomenclature at end of report

B. THEORETICAL VALUE OF BOILER OVERALL HEAT-TRANSFER COEFFICIENT

The theoretical values of the overall heat-transfer coefficients of the boilers and superheater were calculated from the sum of the individual thermal resistances referred to the outer surface of the double-tube assembly ($A_b R_i$).

$$U_b = \frac{1}{\sum A_b R_i}$$

where

$$\sum A_b R_i = \frac{b}{a h_a} + \frac{b \ln c/a}{K_1} + \frac{b \ln d/c}{K_{Hg}} + \frac{b \ln b/d}{K_2} + \frac{1}{h_s} + \frac{1}{h_b}$$

The thermal conductivities and film heat-transfer coefficients were calculated for the flows and temperatures shown in Table I. The theoretical values of the individual heat-transfer resistances and the temperature drops were as shown in Table II.

TABLE II

BOILER HEAT-TRANSFER RESISTANCE VALUES

	$A_b R_i$ $\left(\frac{\text{hr-ft}^2\text{-°F}}{\text{Btu}}\right)$	% of Total	ΔT (°F)
Sodium Film	0.000332	14.9	14.2
Inside Tube Wall (0.065 in. thick, Type 347 SS)	0.000609	27.3	26.1
Mercury (0.030 in. thick)	0.000353	15.9	15.1
Outside Tube Wall (0.095 in. thick, carbon steel)	0.000358	16.1	15.3
Scale	0.000400	18.0	17.1
Nucleate Boiling	0.000176	7.8	7.5
Totals	0.002228	100.0	95.3

The theoretical value of the overall heat-transfer coefficient calculated for the two boilers was 449 Btu/hr-°F-ft^2.

The sodium-film heat-transfer coefficient was calculated from the Labarsky and Kaufman equation [4]

$$D_a\left(\frac{h_a}{K}\right)_{Na} = 0.625\left(\frac{GD_a C_p}{K}\right)_{Na}^{0.4} .$$

The nucleate-boiling heat-transfer coefficient was calculated from the Levy equation [5]

$$h_b = \frac{(KC_p\rho^2)_l\ (\Delta T)^2}{\sigma T_s(\rho_l - \rho_v)\ B} .$$

C. THEORETICAL VALUE OF SUPERHEATER OVERALL HEAT-TRANSFER COEFFICIENT

The theoretical values of the individual heat-transfer resistances and the temperature drops were as given in Table III.

TABLE III

SUPERHEATER HEAT-TRANSFER RESISTANCE VALUES

	$A_b R_i$ $\left(\frac{\text{hr-ft}^2\text{-°F}}{\text{Btu}}\right)$	% of Total	ΔT (°F)
Steam Film	0.003660	61.7	127.2
Scale	0.000300	5.1	10.4
Inside Tube Wall (0.065 in. thick, Type 347 SS)	0.000591	10.0	20.5
Mercury Gap (0.040 in. thick)	0.000463	7.8	16.0
Outside Tube Wall (0.085 in. thick, Type 347 SS)	0.000608	10.2	21.0
Sodium Film	0.000307	5.2	10.6
Totals	0.005929	100.0	205.7

The steam-film heat-transfer coefficient was calculated from the Dittus-Boelter equation [6*]

$$\frac{h_a D_a}{K} = 0.023\left(\frac{D_a G}{\mu}\right)^{0.8}\left(\frac{C_p \mu}{K}\right)^{0.4} \quad .$$

Because the following equation was for liquid-metal flowing over staggered tubes while the superheater tubes were in line, 80% of the value of the sodium-film coefficient calculated from this equation was used.[7]

$$\frac{h_b D_b}{K_{Na}} = 1.17\left(\frac{D\ G_{max} C_p}{K}\right)_b^{0.8}\left(\frac{C_p \mu}{K}\right)_b^{1/3}$$

The theoretical value of the overall heat-transfer coefficient calculated for the superheater was 169 Btu/hr-°F-ft^2.

D. DISCUSSION

The calculated steam film ΔT in the superheater was 61.7% of the total ΔT radially across the superheater tube walls. Since the heat-transfer resistances across the tubes were in series, the superheater overall heat-transfer coefficient could most effectively be increased by decreasing the resistance due to the steam film.

All of the radial temperature drops calculated for the boilers were about equal. Therefore, no one of the series resistances could limit the heat transfer in the boilers.

Since sodium has excellent heat-transfer qualities, the calculated sodium film ΔT in the boilers and superheater was very low. It was 14.9 and 5.2% of the total ΔT radially across the boiler and superheater tubes respectively. The sodium film ΔT in the boilers was higher than that in the superheater because of the higher heat flux in the boilers (calculated at the outer surface of the double tubes), and because the sodium film was on the inside of the double tubes in the boilers while it was on the outside of the double tubes in the superheater.

*All physical properties are evaluated at the bulk steam temperature.

E. EFFECT OF SUPERHEATER DESIGN CHANGES

1. Increase in Steam Mass-Velocity

Differentials of the Dittus-Boelter equation indicate that a 10% increase in the steam mass velocity would cause an 8% increase in the steam-film coefficient.

$$\frac{hD}{K} = 0.023 \left(\frac{DG}{\mu}\right)^{0.8} \left(\frac{C_p \mu}{K}\right)^{0.4}$$

$$\frac{dh}{h} = -0.2\,\frac{dD}{D} + 0.6\,\frac{dK}{K} + 0.8\,\frac{dG}{G} - 0.4\,\frac{d\mu}{\mu} + 0.4\,\frac{dC_p}{C_p}$$

Then, with everything constant (number and diam of tubes, etc.) except G,

$$\frac{dh}{h} = 0.8\,\frac{dG}{G} \quad .$$

As was previously shown, the steam film represented the major heat-transfer resistance in the superheater, 62% of the total. Therefore, the overall heat-transfer coefficient should increase by 5% for a 10% increase in mass velocity.

$$U_1 = \frac{1}{\sum A_b R_i} = \frac{169}{0.62 + 0.38} = 169\ Btu/hr\text{-}ft^2\text{-}^\circ F$$

$$U_2 = \frac{169}{\dfrac{0.62}{1.08} + 0.38} = 177\ Btu/hr\text{-}ft^2\text{-}^\circ F$$

Therefore,

$$\frac{dU}{U} = 0.05.$$

If, however, the mass velocity increased by 10% (with other parameters constant) the power level of the superheater would increase by 10% if the heat-transfer area were increased only 5% by increasing the tube lengths.

$$Q = U_b A_b \Delta T_L$$

$$\frac{dQ}{Q} = \frac{dU}{U} + \frac{dA}{A} + 0$$

$$\frac{dA}{A} = 0.10 - 0.05$$

$$= 0.05$$

Therefore, only a 5% change in heat-transfer area of the superheater would be required for a 10% increase in power level with constant ΔT_L and ΔT_{Na}.

At the 10% higher flow rates, the sodium-film heat-transfer coefficient would be higher, but this change was neglected because the sodium-film resistance was only 5.2% of the total thermal resistance.

The effect on the steam properties of the increased steam-pressure drop across the superheater was negligible. However, there would be an economical limit to which velocity (or turbulence) could be utilized to increase the rate of heat transfer. The friction pressure loss would increase across the superheater with the velocity squared which would cause the superheated steam pressure to be slightly lower. The optimum design compromise, between heat transfer and superheated steam pressure, could only be made for each particular application.

2. Elimination of Double Tube

The chemical reaction between steam and sodium is mild compared to that between water and sodium. Therefore, it may not be necessary to use a double-tube assembly in the superheater. If the mercury gap and outer tube were eliminated, the theoretical values of the individual heat-transfer resistances and the temperature drops would be as shown in Table IV.

TABLE IV

MODIFIED SUPERHEATER THERMAL RESISTANCE VALUES

	$A_b R_i \left(\frac{\text{hr-ft}^2\text{-°F}}{\text{Btu}}\right)$	% of Total	ΔT (°F)
Steam Film	0.003050	73.2	106.0
Scale	0.000300	7.2	10.0
Tube Wall (0.065 in. thick, Type 347 SS)	0.000475	11.4	16.5
Sodium Film	0.000343	8.2	11.9
Totals	0.004168	100.0	144.0

The theoretical value of the overall heat-transfer coefficient of the superheater would then be 240 instead of 169 Btu/hr-ft^2-°F, an increase of 42%. The heat transfer area of the superheater could be decreased by 30% with no change in the power level.

V. TRANSIENT TEST PROGRAM

A. LOW POWER TRANSIENT TEST

SRE power run 11, with the steam generator in operation, started on March 13, 1959. The first scram tests were initiated on March 14, 1959 with an 85°F total temperature gradient across the steam generator. The steady-state steam generator conditions before the test are given in Table V.

TABLE V

CONDITION BEFORE SCRAM TEST ON MARCH 14, 1959

Sodium Inlet Temperature (°F)	546
Sodium Outlet Temperature (°F)	461
Total Sodium Flow (gpm)	800
Steam Flow (lb/hr)	9000
Steam Temperature (°F)	530
Steam Pressure (psia)	475
Thermal Power (Mw)	2.8

After the scram, the total sodium flow rate was reduced to 60 gpm and was held at this rate for 8 min. At this time, to verify a suspected tendency to reverse flow, the sodium flow rate was changed to 35 gpm and held constant for the remainder of the test. Most of the sodium inlet temperatures on the steam generator were observed to change at a constant rate of about 15°F in 30 min. When the sodium flow rate was reduced to 35 gpm, the temperature of one of the lower tubes from the boiler sodium inlet manifold to the boiler dropped 82°F in 6 min (when the temperature gradient across the boiler was 94°F). The locations of the thermocouples were as shown in Figure 5, and the temperature changes as shown in Figure 6.

The boiler configurations and temperatures, shown in Figures 5, 6, and 7, revealed convective pressure heads due to the density differences of the sodium in the vertical manifolds. The natural convection pressure would tend to drive sodium in a direction opposite to that of the normal flow direction. During normal operation with high sodium flow rates, the pressure drop across the boiler tubes would be considerably in excess of the available convective head, and normal flow patterns would be maintained. However, at the low flow rates and

THE NORMAL SODIUM FLOW DIRECTION IS FROM THE LEFT HAND TO THE RIGHT HAND MANIFOLD. THE SODIUM FLOW REVERSES DIRECTION IN THE LOWEST TUBE WHEN THE NATURAL CONVECTION PRESSURE (PROPORTIONAL TO THE SODIUM DENSITY DIFFERENCE) BECOMES GREATER THAN THE FRICTION PRESSURE DROP (PROPORTIONAL TO VELOCITY SQUARED).

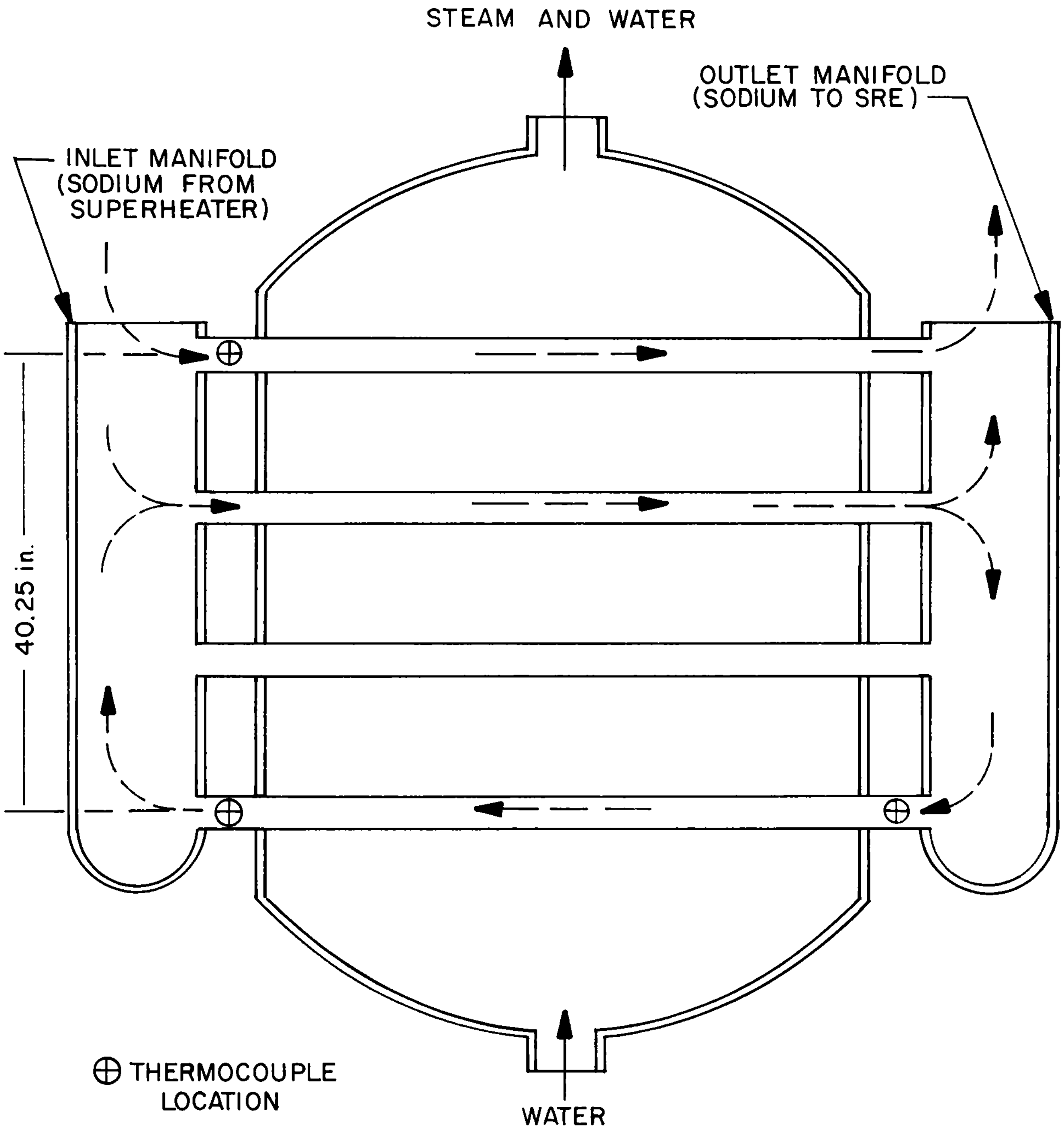

Figure 5. Schematic of Boiler "B" After Sodium Flow Reversal
(24 Pancake Layers of Coiled Tubes in Each Boiler; Only 4 Tubes Shown)

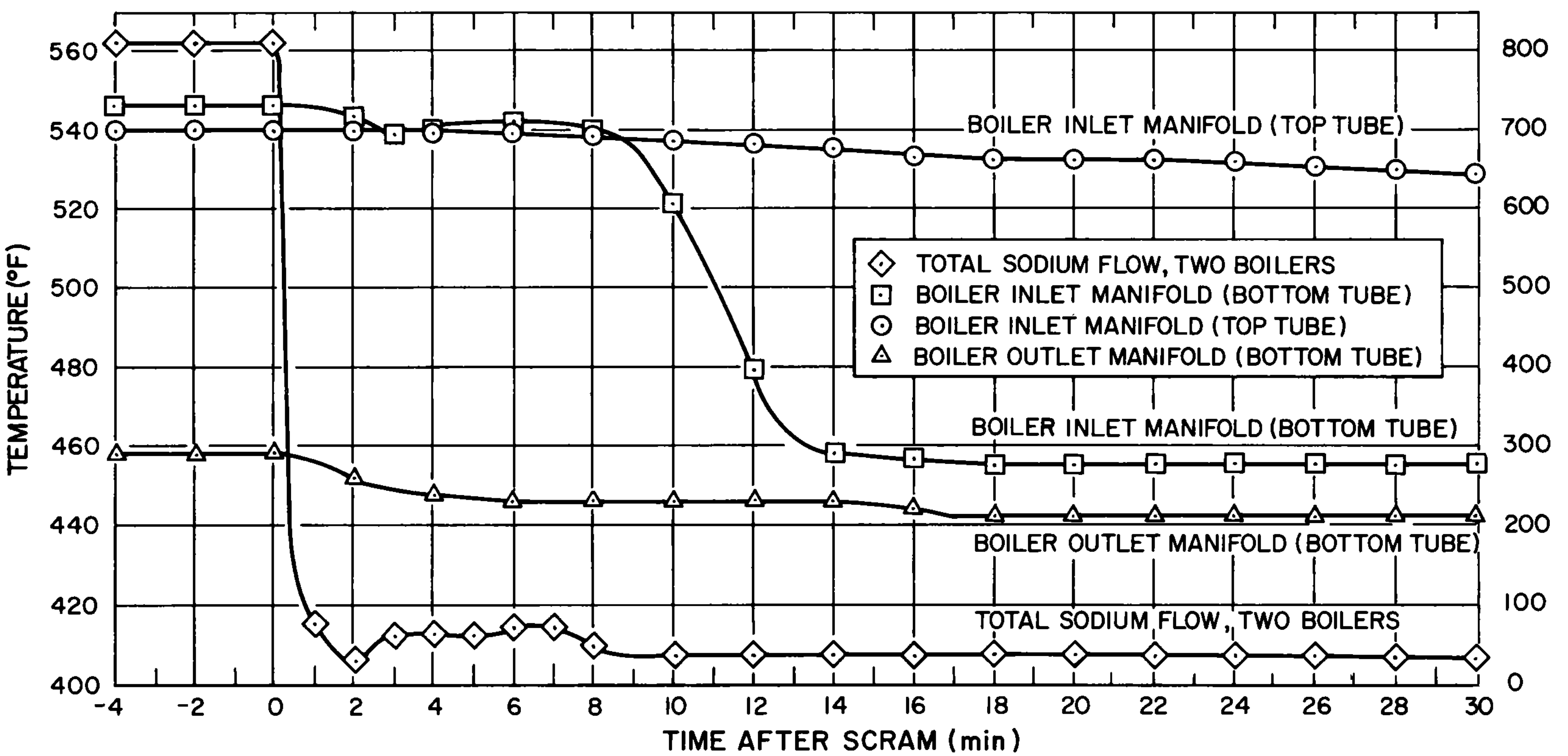

Figure 6. Sodium Flow Rate and Boiler Sodium Temperatures vs Time After Scram on March 14, 1959

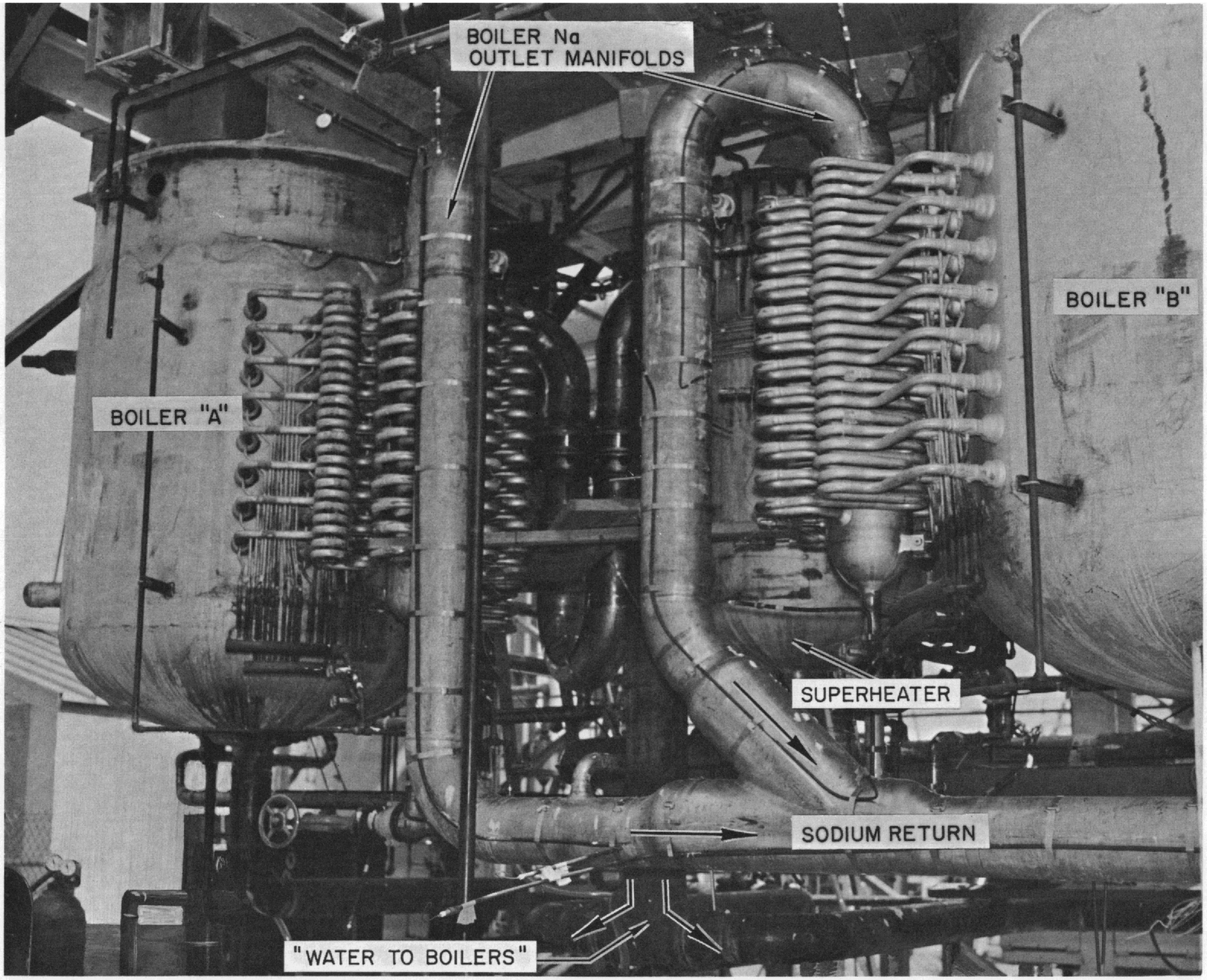

Figure 7. Boiler Sodium Outlet Manifolds

high temperature gradients which may occur after a scram, the friction pressure drop across the lower tubes would fall below the convection head. The result would be a flow reversal within the lower tubes of the boilers as shown in Figure 5.

The temperature drop caused by flow reversal, together with the drawings of the weld joint of the boiler tubes connected to the sodium inlet manifold, indicates that there would be a thermal stress at this location. The cold sodium flowing in the reverse direction in the lowest tube would drop the temperature of the tube manifold weld joint suddenly. The thick manifold section could not follow the rapid temperature change of the tube. The stress would be concentrated on the inside edge of the weld where the tube contacts the thick-walled manifold.

The sudden temperature change shown in Figure 6 indicated that flow reversal occurred in the lower portion of the boiler manifolds at 35 gpm but not at 60 gpm total sodium flow when the sodium temperature gradient across the boilers was 94°F.

At a total sodium flow rate of 35 gpm, the Reynolds number for the flow in the boiler tubes was 2420, which indicated that the flow would be in the transition range between turbulent and viscous flow. At slightly higher flow rates the flow would be turbulent, and the pressure head resisting flow reversal would be proportional to the total sodium flow rate squared. The pressure head causing flow reversal was due to the sodium density difference and was directly proportional to the sodium ΔT across the boilers. Since flow reversal will not occur at a flow rate of 60 gpm with a boiler ΔT of 94°F, then the following relationship indicated that flow reversal would not occur at a flow rate of 100 gpm with a boiler ΔT of 260°F.

$$\left(\frac{X\ GPM}{60\ GPM}\right)^2 = \frac{260°F}{94°F}$$

The low-power transient test was conducted to get this reverse flow data. Flow reversal at low post-scram flow rates was expected because of previous experience with this phenomenon.[8]

B. HIGH POWER TRANSIENT TEST

On March 27, 1959 the reactor was scrammed when the sodium total temperature gradient was 275°F. The steady-state conditions before the scram were as shown in Table VI.

TABLE VI

CONDITIONS BEFORE SCRAM TEST ON MARCH 27, 1959

Sodium Inlet Temperature (°F)	775
Sodium Outlet Temperature (°F)	500
Total Sodium Flow (gmp)	1620
Steam Temperature (°F)	650
Steam Pressure (psia)	574
Thermal Power (Mw)	18.0

After the scram, the total sodium flow rate was reduced to about 100 gpm and was momentarily as low as 50 gpm (Figure 8). None of the boiler sodium inlet temperatures dropped more than 55°F in the first hour after the scram, while the sodium temperature gradient across the boiler was approximately 260°F. Flow reversal was prevented in the manifold tubes due to the high post-scram flow rate specified in a part of the test procedure.

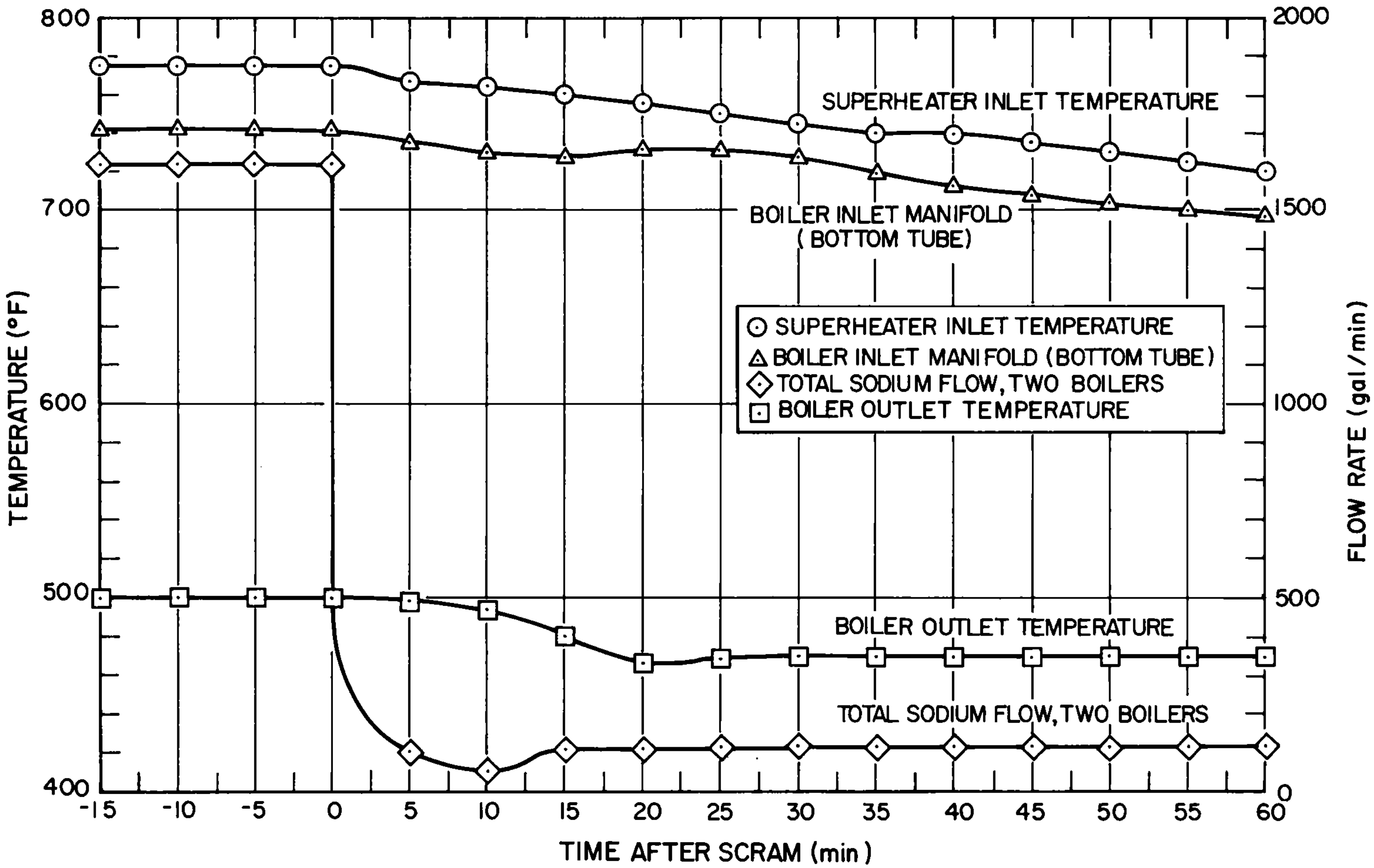

Figure 8. Sodium Flow Rate and Steam Generator Sodium Temperature
vs Time After Scram on March 27, 1959

VI. CONCLUSIONS

A. OPERATING STABILITY

The three-element feedwater controller now functions properly over the complete range of operating conditions. As a result of the modifications, the controller has good response and stability.

B. PREVENTION OF REVERSE FLOW

Sodium flow reversal can be prevented in the boiler manifolds after a scram from 275°F total sodium temperature gradient when the post-scram flow rate is programmed to a value of 100 gpm (the normal SRE post-scram flow rate). Therefore, high thermal stresses could not occur.

Sodium flow reversal cannot be induced in the superheater because the flow direction before a scram is the same as the natural convection flow direction.

C. OVERALL HEAT-TRANSFER COEFFICIENTS

The experimentally determined value of the overall heat-transfer coefficient for the two boilers and superheater were 507 and180 Btu/hr-°F-ft^2 respectively, while the theoretical values were 449 and 169 respectively. The experimental and theoretical values were in reasonable agreement. The theoretical values were 11.4 and 6.1% lower than the experimental values for the boilers and superheater, respectively.

If a single tube were used in the superheater instead of the double-tube assembly, the overall heat-transfer coefficient would be 42% higher than the theoretical value shown above. The heat transfer area of the superheater could be decreased by 30% with no change in the power level.

Most of the resistance to heat transfer in the superheater (61.7%) is due to the steam film inside the inner tube. Analysis of the steam-film heat-transfer coefficient equation indicated that, for small changes in the steam velocity, the steam-film coefficient changes by 80% of the steam velocity change. Since the steam-film resistance is 61.7% of the total heat-transfer resistance, increasing the power level of the superheater by 10% (by increasing the mass flow rates by 10%) would require only a 5% increase in the heat-transfer area for a constant ΔT_L and ΔT_{Na}.

VII. RECOMMENDATIONS

A. CLEANING OF STEAM GENERATOR

The quantity of impurities that would not dissolve in water and that are still in the steam generator is not known. Chemical cleaning of the steam generator may improve the overall heat transfer coefficients or prevent them from becoming lower. Therefore, it is planned to chemically clean the steam generator before it is used again. This will probably be done with phosphoric acid, which is used on other stainless steel steam generators designed for nuclear power use.[9]

B. THERMAL STRESS TEST ON MOCKUPS

A thermal stress would occur at the weld joint of the lower tubes to the sodium inlet manifold of the boilers if a flow reversal were experienced. There would be a stress concentration at the inside edge of the weld at the point where the tubes join the thick-walled sodium manifold. It is not possible to calculate the maximum stress from temperature gradient and strain data taken on the outside of the tubes and manifold because of the type of stress concentration. Therefore, transient tests should be made on a mockup of this weld joint.

C. IMPROVEMENT OF INSTRUMENTATION

Thermocouples and resistance thermometers should be added to the steam lines and sodium lines in order to get more accurate heat transfer information. To obtain data on the temperature gradients which occur when flow reversal is purposely induced, thermocouples should be added to the lower tubes connected to the boiler inlet manifold.

After the steam generator is chemically cleaned, the steam quality should be determined by the ratio of the silica concentration in the steam to that in the boilers and checked by the ratio of the total dissolved solids in the steam to that in the boilers.

D. TRANSIENT TESTS

Transient tests should be conducted on the steam generator to give time-constant data needed for the SRE-Edison plant control system, the type which will be used to check out the control design for the HNPF. During the previous steam generator test, the superheated steam pressure was held constant by controlling the turbine steam valve position from the pressure error signal. This type of control, not normally used on a drum-type steam generator, was used because it was already installed this way for the once-through steam generator. Normally, the turbine steam valve position is controlled either manually or from a load distribution substation, and the superheated steam pressure error controls thermal power into the steam generator. In the case of sodium-cooled reactors, the total temperature gradient of the sodium will be held constant, and the sodium flow rate will be controlled to vary the thermal power into the steam generator. Transient tests should be made with this type of control system, since it will be used for the HNPF.

VIII. NOMENCLATURE

A_b = Outside surface area of outer tube (ft^2)

B = Empirically determined coefficient shown in Figure 4 of reference 5, nondimensional

C_p = Specific heat of fluid (Btu/lb-°F)

G = Mass velocity per unit cross-section area (lb/hr-ft^2), equals ρV

K = Thermal conductivity (Btu/hr-ft-°F)

K_1 = Thermal conductivity of inside tube (Btu/hr-ft-°F)

K_2 = Thermal conductivity of outside tube (Btu/hr-ft-°F)

Q = Heat-transfer rate (Btu/hr)

R_i = Individual heat-transfer resistance (hr-°F/Btu)

ΔT = Temperature gradient (°F)

ΔT_L = Log mean temperature difference (°F)

D = Tube diameter (ft)

T_s = Saturation temperature (°R)

U_b = Overall heat-transfer coefficient referred to outer surface of double-walled tube assembly (Btu/hr-ft^2-°F)

V = Velocity (ft/hr)

a = Inside radius of inside tube (ft)

b = Outside radius of outside tube (ft)

c = Outside radius of inside tube (ft)

d = Inside radius of outside tube (ft)

h_a = Film heat-transfer coefficient of fluid inside double tubes (Btu/hr-ft^2-°F)

h_b = Film heat-transfer coefficient of fluid outside double tubes (Btu/hr-ft^2-°F)

h_s = Heat-transfer coefficient of scale (Btu/hr-ft^2-°F)

Subscripts

Hg = Mercury

Na = Sodium

a = Inside of inside tube

b = Outside of outside tube

c = Outside of inside tube

d = Inside of outside tube

l = Liquid

v = Vapor

ln = Natural logarithm

Greek

ρ = Density (lb/ft^3)

σ = Surface tension (Btu/ft^3)

μ = Viscosity (lb/hr-ft)

$\sum$ = Summation

REFERENCES

1. "Natural Circulation Double Tube Steam Generator," NAVSHIPS 351-0506 (November 1956)

2. Liquid Metals Handbook, Sodium NaK Supplement, TID 5227, 3rd Edition, 2nd Printing (U.S. Government Printing Office, July 1, 1955), p 283

3. Supplied by E. D. Bilbao and R. W. Madsen in a conversation with R. D. Welsh, March 1959

4. B. Lubarsky and S. Kaufman, "Review of Experimental Investigations of Liquid-Metal Heat Transfer," NACA 1270 (1956)

5. S. Levey "Generalized Correlation of Boiling Heat Transfer," ASME-AIChE Heat Transfer Conference, Paper No. 58-HT-8, Chicago, Ill., (August 18-21, 1958)

6. W. H. McAdams, Heat Transmission, (3rd Edition, McGraw-Hill Book Co., Inc., 1954) p 219

7. H. Etherington (ed) Nuclear Engineering Handbook (McGraw-Hill, 1958), pp 9-63

8. K. W. Foster, "Thermal Performance and Structural Evaluation of the SRE MIHE," NAA-SR-3775 (to be published)

9. Supplied by I. Bielek, Dowell Division of Dow Chemical Co., in conversation with R. D. Welsh, AI, June 1, 1959

ATOMICS INTERNATIONAL • A DIVISION OF NORTH AMERICAN AVIATION, INC.